

Management and Disposition of Excess Weapons Plutonium

Reactor-Related Options

Panel on Reactor-Related Options for the
Disposition of Excess Weapons Plutonium

Committee on International Security and Arms Control

National Academy of Sciences

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**PANEL ON REACTOR-RELATED OPTIONS
FOR THE DISPOSITION OF
EXCESS WEAPONS PLUTONIUM**

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Preface

With the end of the Cold War, the United States and the republics of the former Soviet Union have undertaken arms control on an unprecedented scale. What to do with the fissile materials from the tens of thousands of nuclear weapons to be dismantled has become a pressing problem for international security. Limits on access to these materials are the primary technical barrier to acquisition of nuclear weapons in the world today.

In 1992 the U.S. government asked the Committee on International Security and Arms Control (CISAC) of the National Academy of Sciences (NAS) to study alternative approaches for dismantling nuclear weapons, and for storing and eventually using or disposing of the plutonium they contain. To support CISAC's work, the NAS formed the Panel on Reactor-Related Options for the Disposition of Excess Weapons Plutonium in November 1992. The panel consists of three members of CISAC and four additional members selected for their relevant expertise on issues related to reactors and reactor wastes (see list of panel members on p. iii).

The official U.S. government sponsor of the project is the Office of Nuclear Energy of the U.S. Department of Energy. Additional support was provided by the John D. and Catherine T. MacArthur Foundation and National Research Council funds. The Carnegie Corporation of New York and the MacArthur Foundation provide core support for CISAC, including its policy reports.

The panel's report served as input to the deliberations of CISAC in its broader charge, which included consideration of disposition options not related to nuclear reactors, as well as issues of preliminary storage and management of

the weapons plutonium.¹ The responsibility for the content of the panel's report, which has been subjected separately to the Academy's review process, rests solely with the members of the panel; similarly, the non-CISAC members of the panel bear no responsibility for the conclusions that CISAC drew, in its 1994 report, from this and other inputs.

Like the main committee study, the panel report proved to be an immense undertaking, requiring hundreds of hours of research, drafting, and discussion by the panel members. The panel's basic analysis and conclusions were completed in late 1993, in time to be an essential ingredient of the full CISAC report. It required an additional 18 months, however, to complete the drafting, editing, and review of the panel's report to its satisfaction. The consensus achieved in the fall of 1993 has not changed over that time, but the analysis is now laid out in full detail and documented. It provides substantial additional information and analysis on various reactor-related options beyond that contained in the committee report.

Every member of the panel contributed to the work of the group, with each person responsible for drafting the description and assessment of particular options. Panel chair John P. Holdren, who is also the chair of CISAC, wrote major sections of the report and undertook the formidable task of comparing the various options. The depth and richness of the report reflects his prodigious efforts.

The CISAC staff provided invaluable assistance throughout the course of the panel's work. Study Director Matthew Bunn somehow managed to oversee the work of both the main committee and the panel. He was an essential liaison between the two groups and provided significant intellectual input to the work of both. He drafted major portions of the CISAC report and edited the panel report to harmonize the work of the individual panel members. The project could not have been completed without him.

CISAC's research associate, Lois Peterson, and research assistant, Monica Oliva, provided crucial substantive and administrative support, including the preparation of the manuscript for publication as part of the new National Academy Press program in desktop publishing. Ms. Peterson also served as an additional staff liaison for the panel once Mr. Bunn was burdened with a new assignment. The entire CISAC staff received a group staff award in recognition of its exceptional efforts on this project.

The issue of management and disposition of plutonium from arms reductions has a long history and a voluminous literature, stretching back almost to the beginning of the nuclear age. In recent years these issues have been studied by a wide variety of groups and individuals in the United States, including those associated with the U.S. Department of Energy and other agencies of the U.S.

¹ The CISAC report (National Academy of Sciences, Committee on International Security and Arms Control. *Management and Disposition of Excess Weapons Plutonium*. Washington, D.C., National Academy Press, 1994) was released prior to the report of this panel.

government, the Office of Technology Assessment, the Natural Resources Defense Council, the Federation of American Scientists, the Center for Energy and Environment Studies at Princeton University, the Center for Science and International Affairs at Harvard University, the Institute for Energy and Environmental Research, several Department of Energy laboratories, and a variety of private companies. Groups and individuals in Russia, Europe, Japan, and elsewhere have also examined the problem. In carrying out their studies, CISAC and the panel benefited greatly from this substantial body of prior work, and extensive communications with many of those involved in it, for which the committee and the panel are profoundly grateful.

In addition, the panel was fortunate to receive help from many parts of the Department of Energy. Staff members from the Department of Energy headquarters and facilities, including Hanford, Savannah River, Los Alamos, and Lawrence Livermore generously gave time to help clarify and resolve technical issues, as well as providing access to relevant experts and materials. The Idaho National Engineering Laboratory merits particular recognition for its significant effort to analyze several aspects of the reactor disposition options, such as non-fertile reactor fuels, carried out without charge to the Academy. Without this assistance, it would have been impossible for the panel to examine the issues in the depth required with the time and personnel it had at its disposal.

As the main CISAC report concludes, there are no easy answers to the problems posed by the fissile materials that are part of the legacy of the Cold War arms competition between the United States and the former Soviet Union. The issues addressed and the options outlined and evaluated will be of critical importance for the future prospects for nonproliferation and arms reduction. Action is urgently needed; in CISAC's words, "The existence of this surplus material constitutes a clear and present danger to national and international security. None of the options yet identified for managing this material can eliminate this danger; all they can do is to reduce the risks."

Bruce Alberts
President, National Academy of Sciences

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Management and Disposition of Excess Weapons Plutonium

Reactor-Related Options

Executive Summary

With the end of the Cold War, many tens of tons of weapons plutonium (WPu) and hundreds of tons of highly enriched uranium (HEU) are expected to be declared surplus to U.S. and Russian military needs. These materials are the essential ingredients of nuclear weapons, and limits on access to them are the primary technical barrier to nuclear proliferation. The existence of large excess stocks of these materials poses a clear and present danger to national and international security.

The report of this panel's parent committee (NAS 1994) recommended an array of steps to reduce this danger, including: verified declarations of total stocks of warheads and fissile materials; bilateral monitoring of warhead dismantlement; secure interim storage of the nuclear-explosive materials resulting from dismantlement, under bilateral and (as quickly as possible) international safeguards; and disposition steps beyond interim storage, designed to make it more difficult to reuse these materials in weapons. The steps beyond interim storage are much less challenging for HEU, which can be "denatured" by blending with natural or depleted uranium, than for WPu, for which no counterpart to this straightforward denaturing process exists. The work of this panel, which supported the parent committee and was reflected in the committee's report, has been confined to disposition of WPu beyond interim storage and, more specifically, to measures that involve irradiation of the plutonium in nuclear reactors or its immobilization with reactor wastes.

The primary motivation of the U.S. government in its search for the best approach to disposition of excess WPu is to minimize the security risks posed

by this material. As long as this material remains in readily weapon-usable form, it will:

- continue to be vulnerable to theft (by parties other than the possessor state) and diversion (by the possessor state) for use in nuclear weapons (which we call the direct risks);
- send the signal that a reversal of current arms reductions remains possible, with negative consequences for arms reduction and nonproliferation efforts (the indirect risks).

The timing of disposition options is crucial to minimizing both types of risks. Minimizing the time until the start of operations to transform the surplus WPu into forms less easily used for weapons, and minimizing the time until this transformation is completed, are of obvious value in reducing the direct risks of theft and diversion. An expeditious approach reduces the indirect risks, moreover, by signaling commitment to irreversible arms reductions and seriousness in addressing proliferation hazards.

While the direct risks are significantly greater in the former Soviet Union, under current economic and political circumstances, than in the United States, the indirect risks apply equally to both countries. To reduce the risks in both categories, the two countries should proceed expeditiously, and more or less in parallel, with programs of WPu disposition that move beyond the status quo—guarded interim storage of plutonium “pits” in the form in which they emerge from weapon dismantlement—to make it significantly more difficult for this plutonium to be reused in weapons by the original possessor state or by others.

In considering the options for using irradiation in nuclear reactors or immobilization with reactor wastes to provide such barriers, the panel has addressed, on a comparative basis, technological readiness, institutional requirements, economics, and environment, safety, and health. We have given the greatest weight, however, to the security characteristics of the various options—their capacity to reduce rapidly the direct and indirect security risks posed by prolonged storage of the plutonium as pits, while minimizing any new security risks arising from the disposition options themselves. (The task of comparing those long-term disposition options within the panel’s purview with other options, as well as the task of comparing all these options with indefinite storage, was left to the parent committee.)

THE SPENT FUEL STANDARD

The panel recommends that the goal for WPu disposition operations to be undertaken in the next few decades should be to meet what this panel and the parent committee call the “spent fuel standard.” This means making the excess WPu roughly as inaccessible for weapons use as the much larger and growing

quantity of plutonium in spent fuel from commercial nuclear-power reactors.¹ The “reactor-grade” plutonium found in commercial spent fuel, while it could be used to make nuclear bombs (NAS 1994, pp. 32–33), poses much smaller risks than separated plutonium in this regard because of the mass, bulk, and intense radiation field of the spent fuel assemblies and because of the additional technical sophistication and resources required for chemical separation of the spent fuel plutonium from the accompanying fission products and uranium.

Options for the disposition of WPu that leave it more accessible than the plutonium in spent reactor fuel would mean that the WPu would continue to pose a unique safeguards problem indefinitely. Conversely, accepting substantial costs, complexities, risks, and delays in order to go beyond the spent fuel standard to make the WPu significantly less accessible for weapons use than the plutonium in commercial spent fuel would not be justified unless the accessibility of the global stock of plutonium in spent fuel were to be similarly reduced.

Promising and Unpromising Options for Meeting the Spent Fuel Standard

Two options have emerged from the panel’s investigations as the most promising ones for the timely disposition of excess WPu to the spent fuel standard:

- the current-reactor/spent-fuel option would use light-water reactors (LWRs) or Canadian deuterium-uranium (CANDU) reactors of currently operating types or evolutionary adaptations of them, employing mixed-oxide (MOX) fuel in a once-through mode, to embed the WPu in spent fuel similar to the larger quantity of such fuel that will exist in any case from ordinary nuclear electricity generation;
- the vitrification-with-wastes option would immobilize the WPu together with intensely radioactive fission products in heavy glass logs of the type planned for use in the immobilization of defense high-level wastes.

Under current U.S. policy, the ultimate fate of either of these waste forms is expected to be a geologic repository. Our conclusion that these two options are the most attractive ones in our purview does not depend, however, on emplacement of the waste forms in a particular repository or by a particular time, or

¹ When currently planned arms reductions are complete, the United States and Russia are each expected to have approximately 50 metric tons (MT) of excess weapons plutonium. At the end of 1990 there were about 260 MT of plutonium in military inventories worldwide and about 650 MT of plutonium that had been produced in commercial power reactors. Of this commercial reactor plutonium, about 530 MT were in irradiated spent fuel and 120 MT were in mixed-oxide (MOX) fuel or stored as plutonium oxide. New production of commercial reactor plutonium during the first half of the 1990s was about 70 MT per year.

even on emplacement in a repository at all: the key point is that once the plutonium is embedded in spent fuel or waste-bearing glass logs of suitable specifications, it will be approximately as resistant to theft or diversion as the larger quantities of plutonium in commercial spent fuel and will represent neither a unique security hazard nor a large addition to the radioactive waste management burdens that the spent fuel and immobilized defense wastes would pose in any case.

All other options in the panel's purview were found to have significant disadvantages compared to the preceding two. Specifically:

- Some of the options *fail to meet* the spent fuel standard, in that the barriers they provide against the reuse of the plutonium for weapons are substantially lower than those of typical commercial spent fuel. This is so for "spiking" options that irradiate WPu-MOX quickly to levels much lower than those typical of commercial spent fuel, and for vitrification of the plutonium in a glass matrix that does not include large quantities of fission products.
- Some of the options *meet* the spent fuel standard but require more time and expense to do so than do the current-reactor/spent-fuel and vitrification-with-wastes options. This is so for building advanced LWRs or advanced liquid-metal reactors (ALMRs) for use in a once-through mode, and for building high-temperature gas-cooled reactors (HTGRs) for use in a once-through mode with moderate burnup.
- Some of the options could *exceed* the spent fuel standard by destroying a larger fraction of the plutonium or by reducing the proportion of the plutonium-239 isotope that is most attractive from a bomb-maker's viewpoint, but doing this would take longer and cost more than the preferred options, and it would only bring a significant security gain after the investment of the still larger amounts of time and money needed to apply the approach to civilian as well as military plutonium. This is the case with use of an HTGR in a high-burnup, once-through mode, with development of nonfertile fuels for LWRs or LMRs, and with use of either current or advanced reactor types (including accelerator-based convertors) in a multiple recycle mode to achieve high destruction of plutonium.

Some of the advanced nuclear technologies considered and rejected by the panel for the WPu disposition mission in the decades immediately ahead might be appropriate for a later campaign to move beyond the spent fuel standard for military and civilian plutonium alike, if society decides to do that, and continuing study of such possibilities at the conceptual level is certainly warranted. The choices eventually made about this will be tightly intertwined with society's choices about how much of its energy will come from nuclear sources and with what technologies this nuclear energy will be provided; these are important

questions, but the panel was not asked to address them and did not need to do so in order to address the narrower and shorter-term question of how to reduce expeditiously the security risks from excess WPu.

The Leading-Candidate Options: Specific Cases and Their Timing

In terms of timing, which as argued above is a crucial element of security, the current-reactor/spent-fuel option and the vitrification-with-wastes option are roughly comparable to each other and superior to the other options in the Reactor Panel's purview. The limiting ingredients on when the current-reactor/spent-fuel approach in the United States could begin processing plutonium would be providing the needed MOX fuel fabrication capacity (no such capacity is currently operational in the United States) and obtaining the necessary approvals and licenses (use of MOX fuel in U.S. power reactors is not now licensed). The limiting ingredients on when the vitrification-with-wastes option could begin processing plutonium would be building or modifying the vitrification facilities to handle plutonium and determining a combination of melter configuration and log composition that provides adequate assurance against accidental criticality in the melter and the repository.

Although the timing of all disposition options is subject to influences that are not entirely predictable, the panel judges that with a prompt decision to proceed in this direction—and given high national priority assigned to the task—fabrication of WPu-MOX fuel could begin in the United States as soon as the year 2001. This timing would entail bringing to operability the partly completed MOX fabrication facilities at the Fuel Materials and Examination Facility (FMEF) at the U.S. Department of Energy site in Hanford, Washington, with a capacity of 50 metric tons of heavy metal (uranium plus plutonium) in MOX fuel per year. For the range of 3.0- to 6.8-percent plutonium in heavy metal contemplated for use in LWRs, this would correspond to 1.5–3.4 metric tons of WPu per year, so that a nominal 50 metric tons of WPu could be fabricated into MOX in 15–33 years of operation. If it were decided instead to build a new MOX fabrication facility at a government site from scratch—perhaps with a capacity of 100 metric tons of heavy metal or 3.0–6.8 metric tons of WPu per year—fabrication of WPu-MOX fuel in the United States might, optimistically, begin as soon as 2003, and 50 metric tons of WPu could be fabricated thereafter in 7–17 years of operation.

Most if not all of the 109 commercial LWRs operating in the United States in 1994 would be capable, without significant modification, of operating with at least one-third WPu-MOX fuel in their cores. (The remainder of the core would contain ordinary low-enriched uranium—LEU—fuel.) Only a small fraction of these reactors, representing about 7 percent of the 1994 U.S. LWR capacity, would be needed in order to load 50 metric tons of WPu, fabricated at the FMEF

into MOX with 4.0-percent plutonium in heavy metal, over a period of 25 years—say, starting in 2002 and finishing in 2026—assuming one-third MOX cores.² If instead of FMEF, a new MOX fabrication plant with twice FMEF's capacity were used and went into operation in 2003, then loading its output into 14 percent of U.S. LWR capacity using one-third MOX cores would lead to completion of loading of the WPu-MOX by about 2016.

There is, of course, a security trade-off associated with increasing the number of reactors using MOX in order to reduce the duration of the campaign. The number of reactors needed to load a given amount of WPu-MOX in a given period can be reduced if the fraction of MOX in the core can be increased. Three 1,221-megawatt-electric (MWe) pressurized-water reactors (PWRs) operating at Palo Verde, Arizona, and one 1,240-MWe PWR that is 75-percent complete in Satsop, Washington, were designed to use 100-percent MOX cores, and U.S. reactor manufacturers have indicated that a significant number of the other operating U.S. reactors could use 100-percent MOX cores safely without major modification. Assuming this capability is confirmed by regulatory review, just two 1,200-MWe-class PWRs using 100-percent MOX cores with 4.0-percent plutonium in heavy metal could load 50 metric tons of WPu in 25 years; with WPu-MOX from FMEF, this job could be finished as soon as 2026. Given twice the WPu-MOX output from a new plant that started in 2003, four reactors with these specifications could finish this job by about 2016. Alternatively, if a WPu loading of 6.8-percent plutonium in heavy metal in a 100-percent MOX core passed safety review, two 1,200-MWe-class PWRs using the output of FMEF could load 50 metric tons of WPu in 15 years, finishing around 2016.

If the use of 100-percent MOX cores in order to hold down the required number of reactors is desired but the existing U.S. reactors suitable for this turn out to be unavailable for the WPu disposition campaign, modifications to one or more of the other operational or under-construction U.S. LWRs would make it possible, at tolerable cost, to use these in the 100-percent MOX mode (again, assuming favorable safety review). The modifications would entail addition of more control absorbers and corresponding changes to the hardware at the top of the core, and if applied to reactors already operating would require a substantial shutdown period to complete. This could be done well within the period that will be required in any case to bring a U.S. MOX fabrication capability into operation; the extra cost of this approach is, rather, the loss of electricity supply and associated revenues if a working power reactor needs to be shut down for an extended period.

² For example, six 1,250-megawatt-electric pressurized-water reactors using MOX containing 4.0-percent plutonium in heavy metal in 33 percent of their cores, operating at a capacity factor of 75 percent, and irradiating their fuel to 42 thermal megawatt-days per kilogram of heavy metal, would load among them 2.0 metric tons of WPu per year and thus would complete the loading of 50 metric tons of WPu in 25 years.

If, for some reason, no combination of currently operating and partly completed U.S. LWRs was deemed attractive for the WPu disposition mission, it would be possible at the cost of a few years' delay to construct a new dual-purpose (WPu disposition/electricity-generation) reactor or reactors on a government site. The logical choice of reactor type for this function, given adoption of the spent fuel standard and the desirability of minimizing the delay, would be an evolutionary LWR. Such reactors are technically mature, have achieved final design approval from the Nuclear Regulatory Commission, and are under construction overseas; they would offer the smallest increases in time and cost, compared to using currently operating reactors or finishing partly completed ones, of any new reactor type.³ Given a timely decision to proceed, fuel loading in a new evolutionary LWR could begin, optimistically, as soon as 2005.

Another way to accomplish the current-reactor/spent-fuel option would be through the use of some of the heavy-water-moderated CANDU reactors in commercial operation in Canada. These reactors appear to be capable of using 100-percent MOX fuel without physical modification. Given favorable regulatory review and Canadian interest in participating in nuclear arms reductions in this way, it would be possible for two currently operating CANDU reactors of 769 MWe each to process 50 metric tons of WPu into spent fuel somewhat less radioactive than that from U.S. LWRs in about 24 years of operation. (Canada has 22 CANDU reactors totaling about 15,000 MWe.) Analysis by AECL Technologies⁴ indicates that MOX CANDU fuel in the needed quantities for this scenario could be fabricated in the United States at the FMEF. Assuming FMEF operation starting in 2001, the last of the WPu-MOX could be loaded into the two CANDUs in 2025.

The panel believes that the timing of the current-reactor/spent-fuel option in Russia could be similar to that in the United States or Canada, given a Russian decision to proceed in this direction. For safety reasons, if this option is selected the only currently operating Soviet-designed reactors that should be used are the 950-MWe VVER-1000 LWRs, which are similar to Western PWR designs. Russia has six such reactors in operation and Ukraine has nine. Depending on the results of a safety analysis of high plutonium loadings in this reactor type, and on the acceptability of high plutonium content in the spent fuel, it might be necessary to bring into operation some of the additional VVER reactors currently standing unfinished in Russia if the goal is to process 50 metric tons of WPu in 30 years of operation within that country alone.⁵ In the most straight-

³ Advanced light-water reactors (ALWRs) of the passive type are based on proven technology, but they have not yet reached the stage of licensing approval and thus would entail more time and cost to bring into operation than would the evolutionary LWRs. HTGRs, ALMRs, and other advanced reactor types would entail still greater development costs and delays.

⁴ AECL Technologies is the U.S. branch of Atomic Energy of Canada, Limited.

⁵ As a technical matter, the VVER-1000 reactors in Ukraine could be used along with those in Russia, but sending Russian WPu to Ukraine could raise difficult political issues. Based on one-

forward scenario, the WPu-MOX fuel for the VVER-1000s would be fabricated at a partly completed facility at Chelyabinsk-65, which could be finished for this purpose.

If WPu-MOX fuel were produced not in the United States or Russia but in the existing or planned MOX fabrication plants in Europe and Japan, and if agreements on institutional arrangements for doing this could be reached expeditiously, a current-reactor/spent-fuel disposition campaign might be able to start 2–3 years earlier (by avoiding licensing and plant-completion delays in the United States and Russia). The campaign might also be able to be completed more rapidly once started if enough European or Japanese MOX fabrication capacity were available and it were used to make WPu-MOX for more reactors than in the fabrication-limited cases described above. A shortened campaign would entail the security liability of using MOX in a larger number of reactors than otherwise, however. If MOX fabrication capacity beyond civilian needs is not available, this accelerated approach would also have the shortcoming of increasing the stock of civilian separated plutonium (because of its displacement in MOX fabrication by WPu), unless civilian plutonium separation were slowed down until WPu disposition was complete.

The earliest possibility for implementing the vitrification-with-wastes option in the United States would seem to be at the U.S. Department of Energy's Savannah River plant, where vitrification operations to immobilize the defense high-level wastes (HLW) at that site are scheduled to begin at its Defense Waste Processing Facility (DWPF) in 1996. Although some technical issues require resolution before a decision to proceed can be confidently made, the panel believes that the necessary preparations for adding WPu to this process could be completed by 2005. It would then be possible to incorporate 50 metric tons of surplus U.S. WPu, at 1.3-percent WPu by weight, in the 2,200 logs of 1,700 kilograms each that are scheduled to be produced during the last eight years of the currently planned DWPF campaign, that is, from 2006 through 2013.

Groundbreaking for a vitrification facility similar to Savannah River's DWPF is expected sometime this year at the U.S. Department of Energy's Hanford site. It is to be used to vitrify the military high-level waste now stored at that location, which is roughly comparable in quantity to that at Savannah River. By virtue of its later time schedule, the Hanford facility might be more readily and economically modifiable than the DWPF to accommodate WPu in the vitri-

third MOX fuel, 4.0-percent plutonium in heavy metal, average burnup of 40 megawatt-days per kilogram heavy metal, and 75-percent capacity factor, five VVER-1000 reactors would need just over 30 years to load 50 metric tons of surplus WPu. One official of the Russian Ministry of Atomic Energy has said that only four of that country's six operating VVER-1000 reactors are suitable for MOX fuel. If this problem were to be addressed by completing some of Russia's unfinished VVER-1000 reactors, modifications could be made during the completion of construction to permit use of 100-percent MOX cores in these reactors, which would reduce the number of reactors required for the WPu disposition campaign.

fication process, all the more so if criticality considerations prove to require extensive changes to the current DWPF design.

The timing of a WPu vitrification campaign in Russia probably would be similar, if an early decision were made to proceed in that direction. A waste-vitrification facility with a nominal output of 1 metric ton of glass per day has been in operation at the Chelyabinsk-65 site in Russia since 1987. The phosphate glass composition employed at this facility appears to be less suitable for plutonium disposition than is the borosilicate glass planned for U.S. vitrification facilities. To our knowledge, the time and money that would be needed to modify the Russian facility to make borosilicate instead of phosphate glass and to integrate WPu with its process stream have not been estimated, but these requirements seem unlikely to differ greatly from those we have estimated for the modifications that would be needed to U.S. vitrification facilities. Russian authorities, however, have so far strongly resisted approaches to WPu disposition that would “throw away” the plutonium without generating any electricity from it, irrespective of arguments that electricity generation with WPu is costlier than with LEU.

While the estimates of timing provided here for variants of the current-reactor/MOX option and the vitrification-with-wastes option differ in detail, the uncertainties in both cases—relating more to resolution of institutional issues in the reactor case and to resolution of technical issues in the vitrification case—are bigger than the differences in the reference-case point estimates. Thus it would not be meaningful to say more than that the two sets of options are comparable.

Under the most optimistic assumptions that we consider defensible, any of the advanced-reactor options would be at least a full decade slower to get started than our leading-candidate options would be, given a timely decision to proceed. The delay could easily be longer even for the best developed of the advanced options (HTGRs, LMRs), and it would probably be two decades or more for the least well developed of them (molten salt reactors, accelerator-based convertors). We believe that the direct and indirect security risks of delays of this magnitude should be considered unacceptable, given that the current-reactor/spent-fuel option and the vitrification-with-wastes option provide the means to avoid these risks and given that the advanced-reactor options do not, according to our assessment, offer sufficient advantages in other aspects of security, economics, or environmental, safety, and health characteristics to offset their timing liability.

OTHER ASPECTS OF SECURITY

With respect to those aspects of security that depend on the details of handling, processing, and transporting various plutonium forms, vitrification entails

fewer and somewhat simpler steps than the current-reactor options, and hence would be somewhat easier to safeguard.

With respect to security of the final plutonium forms, the current-reactor options obviously meet the spent fuel standard, and the panel judges that the vitrification option meets this standard also. The plutonium in the spent fuel assembly would be of lower isotopic quality for weapon purposes than the still weapons-grade plutonium in the glass log, but since nuclear weapons could be made even with the spent fuel plutonium this difference is not decisive. Under typical assumptions, the radiological barrier presented by glass logs would be about three times smaller than that presented by a fuel assembly (but still very high), and the mass of a glass log—containing, coincidentally, about the same amount of plutonium as a fuel assembly—would be about three times greater.⁶ The difficulty of separating the plutonium from the accompanying materials would be roughly comparable in the two cases.

To summarize the situation with respect to security of the current-reactor/MOX and vitrification options: (1) the two options are comparable in timing; (2) vitrification has a modest advantage in safeguardability of the handling, processing, and transport steps; and (3) the current-reactor/spent-fuel option has a modest advantage in the barriers associated with the final plutonium form because of the difference in plutonium isotopics. The panel concludes that the two approaches are comparable in security overall, that either would be adequate, and that no other option known to us is superior.

Use of advanced reactors could reduce the quantity of residual plutonium from the disposition process, even on a once-through basis in the case of the HTGR, and with the assistance of reprocessing and plutonium recycle in other cases. Reprocessing and recycle could also accomplish such reductions in conjunction with current or evolutionary LWRs. In some of these cases, most notably the HTGR, the residual plutonium's isotopic quality for weapon purposes could also be reduced to below that characteristic of typical spent LWR fuel. As noted earlier, however, these changes would not bring much reduction in overall security risk unless commercial spent fuel stocks were similarly transformed. Although society might eventually decide to do this and might choose advanced reactor types for the purpose, transforming today's very dangerous stocks of

⁶ For example, a large glass log of the type expected to be produced at Savannah River's DWPF would contain 1,700 kg of glass in a 450-kg steel jacket; at 1.3-percent WPu in glass, it would hold 22 kg of plutonium; and, at the expected defense high-level-wastes content of 20 percent by weight it would produce a gamma-ray dose rate of 2,600 rem (roentgen-equivalent-man) per hour at the surface of the container 30 years after the log was produced. By comparison, a MOX fuel assembly from a Westinghouse PWR would have a mass of about 660 kg, would contain about 18 kg of plutonium after irradiation to 40 megawatt-days per kilogram of heavy metal (assuming initial WPu content of 4.0 percent of heavy metal), and would produce a gamma-ray dose rate of 7,900 rem per hour at the surface of the assembly 30 years after discharge from the reactor.

excess WPu to meet the spent fuel standard does not require advanced reactors and should not wait for them.

ECONOMIC COMPARISONS

The monetary costs of alternative approaches to the disposition of WPu are of secondary importance compared to the security aspects. The panel has nonetheless devoted considerable effort to developing a reasonably comprehensive and internally consistent set of estimates of the costs of alternative approaches, including accounting for electricity revenues where appropriate, in order to try to sort out diverse claims in the literature about the costs or profitability of WPu disposition operations, to assist in ranking options that are not readily distinguishable on security grounds, and to facilitate planning for the needed investments.

The panel's estimates indicate that the most likely costs for the borosilicate-glass/vitrification option and for the less costly among the current-reactor/spent-fuel options both fall in the range of \$0.5–\$2 billion (1992 dollars), expressed as the discounted present value, as of the start of plutonium disposition operations at the reactor or melter, of the stream of incremental costs associated with plutonium disposition, less electricity revenues where it is appropriate to count them. (The real discount rate assumed was 7 percent per year.)

The range of central estimates for all of the current-reactor options extends from \$0.5 billion to about \$5 billion in these terms. The lowest central estimate, at about \$0.5 billion, is for the option using currently operating U.S. LWRs that need no modification to use MOX safely, with the fuel fabricated at the FMEF at the Hanford site. Four of the options studied have central-estimate costs around \$1 billion: use of MOX fuel from FMEF in currently operating CANDU reactors in Canada; use of MOX from FMEF in a single, currently mothballed, partly completed PWR that would be completed for this purpose; use of MOX from an entirely new fuel fabrication plant in currently operating U.S. LWRs that need no modification to use MOX safely; and vitrification with defense high-level wastes at the Savannah River site. The estimates in the range of \$5 billion are for options involving the construction of both new reactors and new MOX fabrication facilities.

Although the central estimates in all cases considered correspond to net costs, the panel's judgmental 70-percent confidence intervals include a possibility of profits from WPu disposition in some of the cases in which reactors that would not otherwise operate are completed or built from scratch for the purpose of WPu disposition and use MOX fuel from FMEF. A profit from plutonium disposition would only occur, however, if the fabrication of MOX fuel were significantly cheaper than now appears likely (compared to fabrication of equivalent LEU fuel), and if, at the same time, the electricity produced in the reactor could be sold at prices somewhat higher than now seem likely. (For

every case in which light-water reactors for plutonium disposition could charge high enough electricity prices to make a profit, an even larger profit would arise if the same reactors used LEU fuel rather than MOX, without addressing the problem of WPu disposition.)

The range of \$0.5–\$5 billion—covering the best estimates of net present value, at reactor or melter startup, of most of the options considered—corresponds to \$10,000 to \$100,000 per kilogram of WPu, or \$40,000 to \$600,000 for a nominal “bomb’s worth” of 4–6 kilograms. Even the higher figure is probably less than what this weapon material once cost to produce, as well as much less than would be spent in the attempt to recover such material if it went astray. It is incomparably less than would be spent to try to deter or otherwise prevent its use in the form of a bomb in the hands of a potential adversary. Thus, funding should not be allowed to become a barrier to carrying out plutonium disposition.

ENVIRONMENT, SAFETY, AND HEALTH (ES&H)

The greatest dangers to public welfare associated with the existence and disposition of WPu are unquestionably those connected with national and international security—that is, the dangers associated with the potential uses of this material in nuclear weapons, as well as the dangers that could be posed for global arms reduction and nonproliferation prospects by failure to manage the WPu in a manner widely understood to preclude its reuse in weapons. The pre-eminence of these security dangers, however, should not obscure the need for careful attention to the ES&H risks posed by the WPu under the different possible options for its disposition.

The panel regards it as very important that the governments involved express in the strongest terms their commitment to respect reasonable ES&H constraints in their WPu disposition programs, and that they demonstrate this commitment by promulgating promptly an appropriate set of ES&H criteria for the WPu disposition process and by putting in place whatever mechanisms and resources are required to give confidence that those criteria will be met. The panel believes that options for plutonium disposition should:

- comply with existing regulations, of the country in which disposition takes place, governing radioactivity and radiation from civilian nuclear-energy activities;
- comply with existing international agreements and standards on the disposition of radioactive materials in the environment; and
- not add significantly to the ES&H burdens that would result, in the absence of programs for disposition of WPu, from appropriate management of civilian nuclear-energy generation and of the environmental legacy of past nuclear weapons production.

The panel believes that both the current-reactor/spent-fuel and vitrification-with-wastes approaches, suitably designed, can meet these criteria. Most of the ES&H impacts of WPu disposition using either of these options can be expected to represent modest additions, at most, to the routine exposures to radiation and risks of accident associated with other civilian and military nuclear-energy activities underway in the United States. There is no apparent reason that the activities involved in WPu disposition using either of these approaches should not be able to comply with all applicable U.S. ES&H regulations and standards.

While there are differences in detail in the ES&H challenges and risks posed by the two options—e.g., a somewhat more complicated set of plutonium-handling operations for the reactor options than for the vitrification option, and a greater relative increase in plutonium content of the final waste form for the vitrification option than for the reactor options—these differences do not consistently favor one class of options or the other, and none is large enough in relative or absolute terms to justify choosing one class of options over the other.

ES&H issues that will need further attention in the next phase of study of these options include: developing and testing the systems to ensure adequate safety against criticality accidents in the melter for the vitrification option; confirming the conditions under which full-MOX cores can be used without adverse impacts on safety in reactors of currently operating commercial types; and determining the conditions that will provide for adequate assurance against long-term criticality in geologic repositories containing either spent fuel or glass logs from plutonium disposition operations. These issues differ considerably in the nature and complexity of the work that will be needed to settle them to the satisfaction of the technical and regulatory communities, but it is the panel's judgment that suitable approaches exist for all of them.

Both the current-reactor/spent-fuel option and the vitrification-with-wastes option would add WPu to a set of nuclear activities that would be going on in any case, and both would leave the residual WPu in a waste form—spent fuel in one case and HLW-bearing borosilicate glass logs in the other—that will exist in large quantities and will need to be safely managed whether used for WPu disposition or not. The panel emphasizes, in this connection, that a U.S. geologic repository is not likely to be ready to receive wastes of any kind before 2015. Vitrified waste logs, with or without plutonium, and plutonium-containing spent fuel from nuclear reactors, whether WPu has been incorporated in some of it or not, will need to be stored in engineered facilities until a geologic repository is ready to receive them.

RECOMMENDATIONS

Parallel Project-Oriented Programs

The current-reactor/spent-fuel and vitrification-with-wastes options are the two leading contenders for plutonium disposition to the spent fuel standard. Because it is crucial that at least one of these options succeed, because time is of the essence, and because the costs of pursuing both in parallel are modest in relation to the security stakes, the panel recommends that project-oriented activities be initiated on both options, in parallel, at once.

Although both of these options are technically feasible and have been recommended by the panel in part because they can be deployed comparatively rapidly, some significant uncertainties accompany both of them. The areas of uncertainty in the current-reactor/spent-fuel option are primarily in licensing and public acceptance. Those in the vitrification-with-wastes option relate mainly to criticality and to the technical issues of mixing plutonium and high-level wastes. Further paper studies will not significantly reduce these uncertainties; the best way to reduce them is through the experience gained in moving toward implementation.

In connection with the current-reactor/spent-fuel option, work should be started to seek out specific reactors and MOX fabrication options that would minimize multiple plutonium transportation steps so as to reduce this aspect of security risk, to identify locations that are most amenable to public acceptance, and to ascertain the willingness of the plant owners to participate and the conditions they would impose. Detailed engineering studies should be completed and licensing applications submitted to the Nuclear Regulatory Commission. From that base, the sound cost estimates, schedules, and financial plans that are essential to considering full project authorization could be prepared.

In connection with the vitrification-with-wastes option, laboratory work and realistic testing should be started to address the technical uncertainties. Research and development plans, program schedules, and key milestones should be defined. As the results of the research and development are obtained, detailed cost estimates and ES&H analyses can be developed and submitted to the U.S. Department of Energy and the Defense Nuclear Facilities Safety Board in pursuit of project authorization.

The pursuit of the two options in parallel, as project-oriented tasks with near-term milestones and aggressive schedules, should be aimed at bringing both processes online by the end of the century or as shortly thereafter as possible. It would be a mistake to spend tens of millions of dollars and additional years of paper studies to try to demonstrate, in the absence of actual work toward deployment, which of the two options should be selected over the other. It will likely be less expensive in the long run, and clearly superior from the security and ES&H perspectives, to proceed with both now. If either option falters

due to technical, licensing, or other difficulties, the pursuit of the other option can continue without the loss of time that would have been associated with choosing early and choosing wrong. Indeed, it may prove to be desirable to implement both options for different parts of the stockpile: as one example, it may turn out that reactor approaches are preferable for the relatively pure plutonium in the components of dismantled weapons, while vitrification may prove to be the better alternative for the tons of plutonium that exist in scrap, solutions, and other forms.

We recommend, therefore, that the U.S. Department of Energy's Programmatic Environmental Impact Statement process (scheduled to be completed in 1996) should be oriented toward a decision to pursue both the current-reactor/spent-fuel option and the vitrification-with-wastes option, not toward a decision to eliminate one or the other. Subsequent preparation of Environmental Impact Statements (EISs) for both options, and the participation of the public in these processes, should proceed in parallel too. The inclusion, in the EIS and public participation processes, of the results of the specific project-oriented activities mentioned above will be essential to the success of those processes. Full project authorization for one of the options would not be granted until the EIS is completed and approved.

Working with Russia

The fundamental objective of the WPu disposition program will not be achieved unless the Russians carry out a disposition program in parallel, on a similar time scale, and adhering to disposition standards equivalent to those of the United States. The project-oriented activities recommended above would lend themselves to forming joint projects with the Russians to assure such a parallel approach. Joint projects will also serve to develop a technical consensus on the disposition process and standards, which, as pointed out elsewhere in this report, does not exist today. The panel recommends that the United States immediately initiate joint project-oriented activities with Russia covering both the MOX and the vitrification options. The panel also strongly concurs with the parent committee's recommendation that the United States and Russia should continue discussions with the aim of agreeing that whatever disposition options are chosen, an agreed, stringent standard of accounting, monitoring, and security will be maintained throughout the process—coming as close as practicable to meeting the standard of security and accounting applied to intact nuclear weapons.

The Longer Term

Follow-on studies should continue on the longer-range questions of whether and how the residual security risks of WPu and other plutonium should eventually be reduced beyond the spent fuel standard. It is essential, however, that

such longer-range studies not be allowed to draw resources or attention from the pursuit of the two options closest to hand for moving rapidly toward achieving the spent fuel standard for the weapons plutonium that poses a “clear and present danger” today.

REFERENCES

NAS 1994: National Academy of Sciences, Committee on International Security and Arms Control. *Management and Disposition of Excess Weapons Plutonium*. Washington, D.C.: National Academy Press, 1994.

INTRODUCTION

The first and second Strategic Arms Reductions Treaties (START I and START II) call for deep reductions in the strategic nuclear forces that have been deployed on the territories of the United States, Russia, and three of the other states of the former Soviet Union. The United States and Russia have decided to undertake even deeper reductions in tactical nuclear weapons under reciprocal unilateral initiatives. If these agreements and initiatives are fully implemented, the United States and Russia will have large numbers of nuclear warheads that are not required either for deployment or for reserves and stockpile support. We call these “excess” warheads. Some 10,000–20,000 warheads in the United States and at least a similar number in the former Soviet Union are likely to fall into this category, depending on the ultimate scope of reductions and decisions concerning the size of nondeployed reserves. These excess nuclear weapons on the two sides could contain well over 100 metric tons of plutonium, and perhaps 1,000 metric tons of highly enriched uranium (HEU), much of which may also be declared excess to military needs.¹

Given this situation, the United States has an interest in:

1. minimizing the risk that either weapons or fissile materials could be obtained by unauthorized parties;

¹ Throughout this report we use metric tons (MT) as the measure of the amounts of plutonium and HEU; all references to tons are to metric tons. One metric ton is 2,205 pounds, roughly 10 percent larger than an English ton.

2. minimizing the risk that weapons or fissile materials could be reintroduced into the arsenals from which they came, halting or reversing the arms reduction process; and
3. strengthening the national and international arms control mechanisms and incentives designed to assure continued arms reductions and prevent the spread of nuclear weapons.

In pursuit of these objectives, the U.S. and Russian governments have agreed that the United States will buy 500 tons of excess Russian HEU, which will be “blended down” to low-enriched uranium (LEU)² so that it can be used for nuclear reactors but not for weapons. The United States will later resell the material to fulfill demand for nuclear fuel on the domestic and world market.

Plutonium, though it can be used as a fuel in either current or future reactor designs, does not lend itself to the commercial approach being taken with HEU for two reasons: (1) plutonium cannot be isotopically diluted so that it cannot be used in weapons; and (2) it costs more to use as fuel in current light-water reactors than LEU, even if the plutonium is available free while the uranium must be mined, processed, and enriched. Plutonium’s toxicity and the need to safeguard it from diversion and theft³ require special handling procedures that greatly increase the costs of its use.

Accordingly, in 1992 the U.S. government asked the Committee on International Security and Arms Control (CISAC) of the National Academy of Sciences (NAS) to identify and evaluate the approaches that could be used for the disposition of plutonium from excess nuclear weapons. The Panel on Reactor-Related Options for the Disposition of Excess Weapons Plutonium was formed by the NAS in November 1992 to support CISAC’s study. The panel consists of three members of CISAC (Richard Garwin, John Holdren, and Michael May) and four additional members selected for their relevant expertise on issues related to reactors and reactor wastes (John Ahearne, Robert Budnitz, Thomas Pigford, and John Taylor) (see list of panel members on p. iii).

² Natural uranium includes only 0.7 percent of the fissile isotope uranium-235 (U-235), with almost all of the remaining 99.3 percent being U-238. (Isotopes are different forms of the same chemical element having differing numbers of neutrons—235 neutrons and protons together in the case of U-235, and 238 in the case of U-238.) To be usable in a weapon, the concentration of U-235 must be greatly increased from its level in natural uranium, a process known as enrichment. Typical weapons-grade uranium contains more than 90 percent U-235. Because the various isotopes of an element are chemically indistinguishable, enrichment requires physical separation techniques. These are costly and time-consuming, and the relevant technology is not widely available—a set of circumstances that has constituted one of the primary technical barriers to proliferation of nuclear weapons. Some types of nuclear reactors, by contrast, can operate with natural uranium, though most of the world’s power reactors use LEU containing 3–5 percent U-235. HEU can easily be blended with natural or depleted uranium to produce LEU.

³ We follow the Committee on International Security and Arms Control report (NAS 1994, p. 6) in using the term “diversion” for cases in which the state that owns the material returns it to weapon use and the term “theft” for cases in which the material is illicitly obtained by other parties.

Analyses prepared by the panel served as input to the deliberations of CISAC in its broader charge, which included consideration of disposition options not related to nuclear reactors, as well as issues of preliminary storage and management of the weapons plutonium (WPu). CISAC's report was released in early 1994 (NAS 1994). Completion of the current report of the Reactor Panel, which in some respects is more detailed than the analyses originally presented to the parent committee but does not alter in any significant way the conclusions of those analyses, required additional time. The responsibility for the content of the panel's report, which has been subjected separately to the Academy's review process, rests solely with the members of the panel; similarly, the non-CISAC members of the panel bear no responsibility for the conclusions that CISAC draws, in its own report, from this and other inputs.

We took the task of our panel to be the identification and illumination of all those options for the destruction or alteration of WPu that entail either emplacing and irradiating it in nuclear reactors or immobilizing it in waste forms similar to those contemplated for disposal of fission-product wastes originating in nuclear reactors. (CISAC decided to include the latter options within our purview because of the familiarity of some of our panel members with radioactive waste vitrification technologies and the related radioactive waste management issues.)

The reactor-related options for disposition of excess WPu can be divided into several categories. Some options (described as "minimized accessibility" options in the CISAC report) would destroy only a fraction of the plutonium, but would increase the difficulty of recovering it and using it in nuclear explosives. Reactors, for example, could accomplish this objective by using the plutonium in a once-through cycle, leaving a significant amount of plutonium (different in its isotopic characteristics from the original plutonium) embedded in spent fuel that would also contain intensely radioactive fission products. We call this category of options the "spent fuel" approach. Alternatively, the plutonium could be irradiated in reactors more briefly (accomplishing the job faster, at the cost of providing a smaller barrier to weapons use of the material); this we refer to as "reactor spiking." Similar objectives could be accomplished by immobilizing the plutonium in a solid waste form—such as a glass, a ceramic, or a cement—which might or might not include radioactive fission products from prior reprocessing of nuclear fuel; we call this the "immobilization" approach. Under this approach, the isotopic composition of the plutonium would remain weapons-grade, a factor whose significance we examine in later sections.

Finally, it is also possible in principle, through repeated separation and reuse of the plutonium in the spent fuel, to consume the plutonium nearly completely, so that the resulting wastes would no longer require significant safeguards. Compared to the other approaches, this "elimination" approach could not be begun as quickly and would require a much longer time to complete, as

well as entailing substantially higher costs and technical, political, and institutional uncertainties.

Within each of these approaches, there is a variety of possible technical options that might be used (involving different reactor types, waste treatment approaches, and the like). In this report, we identify these options, and for each one we offer a description of its general characteristics, its state of technological development (including the steps remaining to be accomplished before it could be made operational), and the characteristics of the option as a means of WPu disposition (including the rate at which the plutonium could be processed, the characteristics of the final form in which the plutonium would be left, and the various security risks involved in reaching that final form). We then assess and compare the options on the basis of a variety of criteria: timing and other aspects of security; cost; and environment, safety, and health. Our recommendations rely heavily on these comparisons.

ROAD MAP OF THE REPORT

This report is organized as follows. In the remainder of the introduction, we outline some of the assumptions related to security issues and goals of disposition, as well as the broader context of the issue, that we have taken from the parent study. Chapter 2 provides background information on a variety of topics: principles of nuclear physics and engineering relevant to the plutonium disposition task; the characteristics of weapons-grade and reactor-grade plutonium; a general scheme for classifying the various approaches to plutonium disposition within our purview; the amount of plutonium likely to be excess to military needs in the United States and Russia, in the context of the overall world stocks of plutonium and HEU; and the character, number, and scale of nuclear facilities around the world that may be relevant to the plutonium disposition mission. Chapter 3 introduces and elaborates the criteria we think relevant to comparative assessment of reactor-related options for plutonium disposition. Chapter 4 examines the options involving irradiating the WPu in nuclear reactors (including accelerator-driven subcritical reactors). Chapter 5 examines the immobilization options. Chapter 6 provides analysis and conclusions concerning the relative merits of the various options when judged against the criteria presented in Chapter 3. Finally, Chapter 7 summarizes our conclusions and recommendations.

UNCERTAINTIES

It is inevitable that sizable uncertainties would burden any study of the topic we were assigned, even in the absence of the time constraints that affected our particular effort. Some of the options we consider involve reactors, fuel types, or reprocessing schemes that have never been tested at all, or that have

been tested only at scales much smaller or for times much shorter than those relevant to the plutonium disposition mission. In some cases, important questions relating to performance, safety, or economics remain to be resolved. In the past, nuclear industry projections about time and cost needed to develop new options, and about the costs of using these options once they are developed, usually have proven to be underestimates. The time lags before particular options are deployable, moreover, will depend on corporate, regulatory, political, and public decision processes whose outcomes cannot be predicted with confidence. The bilateral, and possibly multilateral, agreements that may be involved in carrying out plutonium disposition are also a cause of uncertainty, particularly with respect to timing.

All we can do in dealing with such uncertainties is to characterize their origins, magnitudes, and implications for our findings as forthrightly and clearly as possible. When the uncertainties are such as to call into question the very availability of an option, or its availability in a time frame of interest for our purposes, we call particular attention to them. We do so also where there is reason to believe that further focused research effort could reduce important uncertainties relatively quickly.

GOALS, TIMING, AND RELATED FACTORS

As already noted, the primary goal of long-term disposition of excess WPu should be to reduce the risks to national and international security associated with this material's existence. These risks arise in a complex international context that includes global efforts to reduce nuclear arms and prevent their spread, ongoing unrest in the former Soviet Union, and expanding civilian use of plutonium as a nuclear fuel in several countries. These contexts are described in CISAC's 1994 report, and we will not repeat that description here. Rather, our task is to investigate the extent to which reactor-related options could be used to reduce these security risks while meeting the other criteria described in Chapter 3. We address these contexts only as they relate directly to the security risks posed by the various disposition options.

The Importance of Timing

One important question with respect to long-term disposition of excess WPu needs to be addressed at the outset: Why bother? Why not simply continue to store this material indefinitely?

There is no doubt that interim storage lasting for a decade or decades will be necessary because of the time required to implement any of the plausible long-term disposition options. Extending such storage indefinitely would not be an acceptable approach, however, for two reasons. First, maintaining this material in readily weapon-usable form for an indefinite time, with no chosen plan

for its disposition, would send negative political signals for nonproliferation and arms reduction, suggesting that the current arms reductions could be rapidly reversed at any time. Second, indefinite storage of such readily usable nuclear-weapon material would extend substantial risks of diversion and theft into the indefinite future. For these reasons, we believe that it is urgent to begin now to work toward consensus on appropriate disposition options and to implement these options expeditiously once a decision is reached. Indeed, we believe that timing—the time when particular disposition options could realistically be expected to begin operations (considering technical, political, economic, and institutional factors), and how much time would be needed to complete those operations—is one of the key security criteria by which disposition options should be judged, as it will determine how rapidly the liabilities of prolonged storage can be diminished.

In addition to these security issues bearing on timing, some social concerns are also relevant. The communities where plutonium is now stored, or where storage facilities may be constructed, are demanding assurances that storage will have an end-point—that these communities will not become, in effect, permanent plutonium dumps. Such assurances cannot be credibly provided until a plausible road map for ultimate disposition of the plutonium is agreed. Similarly, those with potential commercial or other interests in plutonium disposition, such as reactor vendors or the personnel at government sites that might be involved, deserve timely answers as to whether their capabilities are likely to be employed in this endeavor.

In most cases, the monetary costs of storage are not a major factor contributing to the urgency of disposition. In the United States, the surplus WPu is currently being stored in the form of intact “pits”⁴—avoiding the costs of processing this material into some other form—in simple but highly secure bunkers (called “igloos”) at the dismantlement site. Since all the security required for handling nuclear weapons would be needed at this site in any case, the net additional cost of storing plutonium there is modest. While the United States is considering construction of a more advanced and costly storage facility, decisions about the further disposition of nuclear-explosive materials resulting from disarmament might not have a decisive impact on decisions concerning the need for such a facility, as this site is designed to store not only weapon components resulting from dismantlement but also plutonium in a variety of residue and scrap forms currently stored throughout the U.S. weapons complex. Russia is planning to build a new storage site with U.S. assistance, but once this facility is built (assuming that it will be), the annual cost of continued storage will again be modest, having little impact on decision-making concerning long-term disposition. This situation stands in contrast to that which obtains in the commer-

⁴ “Pit” is the term for the fission-explosive core of an implosion-type nuclear weapon. See also the discussion in “Some Relevant Aspects of Weapon Science” in Chapter 2.

cial plutonium market, where reprocessors charge roughly \$2–\$3/gram-year for storage of civilian separated plutonium—a charge that would amount to \$200–\$300 million per year if applied to 100 tons of WPu.

Cost and Other Objectives

Having noted that timing and other aspects of security are crucial to sensible choices about plutonium disposition, we also point out at the outset of our analysis some factors that should *not* be allowed to drive decision-making regarding plutonium disposition.

For example, although cost inevitably will be a consideration in choosing among long-term disposition options, this factor is of distinctly secondary importance. The net discounted costs of the most promising options we identify amount to between hundreds of millions and a few billion dollars spread over a period of more than a decade—quite small on the scale of ongoing military expenditures designed to reduce major security risks. Differences in the costs of options within this range deserve far less weight in decision-making than differences in security.

Similarly, exploiting the energy value of this plutonium should not be a central criterion for choosing among disposition options. While the stockpile of excess WPu looms large in security terms, amounting to the equivalent of many thousands of nuclear weapons, it is quite small in energy terms—less than a tenth of the plutonium in the world, and the equivalent of only a few months' fuel for the world's power reactors. As described in detail later in the report, moreover, it would be costlier to fabricate free plutonium from weapons into reactor fuel for today's most widely used and economical reactor types than to purchase low-enriched uranium fuel for these reactors, and under plausible assumptions about uranium availability and associated processing costs over the next few decades, the economic disadvantage of plutonium fuel is unlikely to disappear any time soon. Whatever economic value this plutonium might someday represent is small by comparison to the security stakes. Thus plutonium disposition is fundamentally a problem of security, not energy. The cost of management and disposition of WPu must be seen as an investment in security, just as the cost of its production was once viewed.

For much the same reasons, the need for disposition of this small fraction of the world's plutonium stock should not drive decisions regarding the future of civilian nuclear power. That future depends on economic, political, and technical factors outside the scope of this study. We do not take a position on what type of nuclear-energy future would be most appropriate for the United States, the countries of the former Soviet Union, or the world. The stock of excess WPu is not essential to the future of any civilian nuclear development programs. Of course, our consideration of reactor options for the disposition of WPu must take account of the capabilities now available and likely to become available in

the next decade or two in the civilian nuclear-energy-generation sector, and our findings may include insights about the relative diversion resistance of reactor and fuel-cycle combinations that are candidates for civilian use. But national decisions about the future of nuclear energy will (and should) be shaped by many factors in addition to plutonium disposition and plutonium safeguards concerns. We view our task as illuminating the plutonium disposition possibilities that different reactor options would offer, *not* as recommending national nuclear-energy strategies on the basis of these possibilities.

Finally, while tritium production issues were not part of the panel's charge, and we have not examined tritium production in detail, we see no essential reason why plutonium disposition and tritium production need be linked. The arguments on this point are outlined in CISAC's report (NAS 1994, pp. 152–153).

U.S. AND RUSSIAN PLUTONIUM DISPOSITION: DIFFERENCES AND LINKAGES

One critical part of the context for disposition of WPu is the ongoing political, economic, and social turmoil in the former Soviet Union. These crises have raised doubts about whether more than one nuclear-armed state will arise from the former Soviet Union, whether recent arms reduction agreements and pledges will be successfully implemented, and whether adequate accounting and security for nuclear weapons and fissile materials is being maintained. Recent seizures of smuggled weapon-usable materials highlight the urgency of these concerns. In the case of plutonium disposition, these crises raise questions concerning whether the economic resources and political attention needed to ensure secure management and disposition of fissile materials will be available in the near term.

The optimal approaches to long-term plutonium disposition may be different in the United States and Russia.⁵ The risks involved in storing, handling, processing, and transporting plutonium are much higher in Russia under present circumstances, and the two countries' economies and plutonium fuel policies are very different. The most important officials and agencies in the Russian government strongly prefer disposition options that use surplus WPu to generate electricity in reactors. To convince Russia to pursue, in the near term, options that dispose of the plutonium without deriving electricity from it would be difficult (though perhaps not impossible, particularly with sufficient financial incentives).

While U.S. and Russian disposition approaches may differ, rough parallelism in the timing and scale of long-term disposition would be desirable, so that

⁵ If current arms agreements are successfully implemented, all the nuclear weapons of the former Soviet Union will be transported to Russia for dismantlement, leaving Russia alone with the burden of dealing with the excess plutonium from arms reductions.

both nations' available plutonium stocks would remain comparable. After long-term disposition, neither nation's excess plutonium should be greatly more accessible for use in weapons than the other's.

U.S. consideration of the risks associated with the various options for management and disposition of its own plutonium must be informed by an awareness of the potential linkages between U.S. choices and the choices that may be made in the former Soviet Union. U.S. policy could affect the management and disposition of excess WPu in the former Soviet Union in a variety of ways—including financial assistance, negotiated agreements to pursue particular approaches, outright purchase of former Soviet WPu, or merely setting an example. What is done with excess WPu in the United States and the former Soviet Union, moreover, could affect the fate of the substantially larger (and still growing) quantities of separated and unseparated plutonium discharged from civilian nuclear-power reactors worldwide. This study, therefore, will consider disposition of both U.S. and Russian excess WPu, although necessarily with more detailed attention to the U.S. case.

REFERENCES

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BACKGROUND

Consideration of the options for disposition of weapons plutonium (WPU) by irradiating it in nuclear reactors or mixing it with radioactive waste streams from reactors, which is the task of this report, requires reference to the physics and technology of nuclear fission as applied to both nuclear-power production and nuclear weapons. Limitations of time and space preclude presentation of a thorough review to these topics here. In the first part of this chapter, we cover in very abbreviated form just the aspects of fission physics and technology most germane to technical arguments made elsewhere in this report.¹ The remainder of the chapter elaborates on the ways in which reactors or waste streams from reactors could be used in the disposition of WPU, and surveys the numbers, capacities, and distribution of the extant nuclear facilities potentially relevant to such an enterprise.

¹ For accessible but much more thorough introductions to nuclear fission in the nuclear power context, see the report of the American Physical Society study group on the nuclear fuel cycle (APS 1978) or any good introductory text on nuclear reactors (e.g., Nero 1979). On nuclear weapons physics and technology, see Cochran et al. (1987), Serber (1992), and Mark (1993).

PHYSICS AND TECHNOLOGY OF NUCLEAR FISSION

Fission and Chain Reactions

Fission arises because nuclei of atomic number 92 and higher are so large and rich in protons that they are unstable if set into vibration of large amplitude, from which one of the modes of decay is division into two smaller nuclei and two to four free neutrons. Vibration of an amplitude sufficient to cause this fission can be induced in a heavy nucleus by the absorption of a suitably energetic neutron. In the case of heavy nuclei with an odd number of neutrons to start with—such as uranium-233 (U-233), U-235, plutonium-239 (Pu-239), and Pu-241—the absorption of a neutron with the very low energy associated with thermal motion at room temperature is sufficient to induce fission. For heavy nuclei with an even number of neutrons to start with, fission can only be induced if the absorbed neutron carries an energy of a million electron volts (MeV) or more.

Depending on the number of neutrons released per fission, the energy distribution arrived at by the neutrons, the densities of heavy nuclei in the vicinity and their probabilities of fissioning as a function of the energy of an incident neutron, and the probabilities of nonproductive absorption of neutrons or their escape from the vicinity (which depend on the geometry and composition of the materials at hand), it may happen that, for each and every nuclei that fissions, exactly one of the resulting neutrons induces yet another fission. This situation corresponds to a chain reaction that is just “critical,” in which the fission rate and thus the rate of nuclear-energy release do not change with time. (This would be the case, for example, in a nuclear reactor operating at constant power level.) If the circumstances are such, on the other hand, that the neutrons released by each fission succeed in inducing *more* than one additional fission, the chain reaction is “supercritical,” and the fission rate and rate of nuclear-energy release grow with time; this growth can be gradual, as in a nuclear reactor during the startup phase, when its power is being increased from zero up to the reactor’s rated output, or it can be extremely rapid, as in a nuclear bomb.² Similarly, an unintended chain reaction (as sometimes occurs when a sufficient quantity of plutonium or enriched uranium is brought together in a geometry favorable for a chain reaction) is known as a “criticality accident.”

Nuclear-reaction probabilities are expressed in terms of “cross-sections,” with dimensions of area, such that the rates of fission or of nonfission capture associated with a flux of neutrons of N neutrons per square centimeter per sec-

² The doubling time of the energy release rate in a nuclear reactor in startup typically would be measured in seconds, minutes, or even hours; the doubling time in a nuclear bomb is a small fraction of a microsecond.

ond in a material containing n of a given type of transuranic nucleus per cubic centimeter are

$$\begin{aligned} \text{fission rate} &= N (\text{neutrons/cm}^2\text{-sec}) \times n (\text{nuclei/cm}^3) \times \sigma_f (\text{cm}^2), \\ \text{nonfission capture rate} &= N (\text{neutrons/cm}^2\text{-sec}) \times n (\text{nuclei/cm}^3) \times \sigma_c (\text{cm}^2), \end{aligned}$$

where σ_f and σ_c are the reaction cross-sections corresponding to fission and nonfission capture, respectively, for the indicated type of heavy nucleus and the specified incident neutron energy. The cross-sections depend quite strongly on this energy, so that, for the usual case in which the neutron energies are distributed over a range of values, an integration of flux times cross-section over this range of neutron energies is required in order to obtain the reaction rates.³

The heavy isotopes that are most important in both the civilian and military applications of nuclear energy to date are U-235 and -238 and Pu-239, -240, and -241. Uranium-235 and -238 are the two main naturally occurring isotopes of uranium, constituting, respectively, 0.715 and 99.285 percent of natural uranium by weight. Plutonium is virtually nonexistent in nature, but is manufactured in nuclear reactors by means of the nonfission capture of neutrons in U-238.⁴ If the plutonium is removed from the reactor at short intervals, it consists almost entirely of Pu-239; if it remains in the reactor for longer periods, larger quantities of the higher plutonium isotopes are formed by successive absorption of further neutrons. Key characteristics of these isotopes as regards their interactions with neutrons—hence their performance in nuclear reactors and nuclear bombs—are summarized in Table 2-1. (Also shown in the table is another isotope of uranium, U-233, which can be manufactured in nuclear reactors containing thorium by a process, analogous to the production of Pu-239 from U-238, beginning with absorption of a neutron in thorium-232. U-233 has not been important in nuclear energy or nuclear weaponry to date, but has properties that would permit its use in either in the future.)

In consequence of these properties and the energy distributions of the neutrons emitted in fission, it can be shown that U-233, U-235, Pu-239, and Pu-241 all are capable of sustaining chain reactions in either a fast-neutron or a thermal-neutron environment. Thus they can serve as fuels in either “fast” or “thermal” reactors, as well as in nuclear bombs.⁵ Isotopes that can sustain a chain reaction based on thermal neutrons are called “fissile,” and all such isotopes can also sustain a chain reaction based on fast neutrons.

³ This integration is often performed in a very approximate way by representing the neutron energy distribution in terms of a modest number of groups of neutrons with energies in different ranges, whereupon the neutron flux in each group is multiplied by an average cross-section for the group's energy range and the results are summed.

⁴ The absorption of a neutron in U-238 gives U-239, which undergoes beta decay with a half-life of 23.5 minutes to yield neptunium-239, which in turn undergoes beta decay with a half-life of 2.3 days to yield Pu-239.

⁵ Here the terms “fast” and “thermal” refer to the average neutron energies in the reactor core; “thermal” means the neutrons are in thermal equilibrium with the surroundings.

TABLE 2-1 Nuclear Properties of Key Uranium and Plutonium Isotopes

Isotope	Thermal Incident Neutrons ^a			Fast-Spectrum Neutrons LMFBR ^b			Generic PWR Spectrum ^c			ABC (graphite-salt) ^d						
	σ_f (barns)	σ_c (barns)	ν	η	σ_f (barns)	σ_c (barns)	ν	η	σ_f (barns)	σ_c (barns)	ν	η				
U-233	529	46	2.49	2.30	2.63	0.26	2.53	2.30	62.3	7.6	2.50	2.23	203	21.4	2.51	2.28
U-235	583	98	2.43	2.08	1.82	0.53	2.49	1.93	46.1	10.3	2.42	1.98	195	36.9	2.45	2.06
U-238	negl	2.7	negl	negl	0.05 ^e	0.26	2.47	0.40	0.1	0.9	2.80	0.28	0.02	5.6	2.43	0.01
Pu-239	748	269	2.88	2.12	1.79	0.45	2.94	2.35	110	61.3	2.88	1.85	545	311	2.90	1.85
Pu-240	negl	286	negl	negl	0.39	0.52	2.88	1.23	0.58	127	3.14	0.01	0.15	225	2.82	0.002
Pu-241	1,011	358	2.94	2.17	2.39	0.42	2.99	2.54	119	39.1	2.93	2.21	566	211	2.97	2.21

ABBREVIATIONS:

LMFBR = liquid-metal fast breeder reactor.

PWR = pressurized-water reactor.

ABC = accelerator-based conversion reactor.

σ_f = fission cross-section (barns); 1 barn = 10^{-24} cm².

σ_c = nonfission capture cross-section (barns).

ν = neutrons produced per fission.

η = neutrons produced per neutron absorbed in fuel.

negl = negligible.

^a 0.025 electron volts or 2,200 m/sec, compared with 14,000 km/sec at 1 MeV.

^b MCNP calculations at Los Alamos National Laboratory (Venneri 1994).

^c In commercial light-water reactors, the ratio of water to fuel is not sufficient to fully thermalize the neutrons, so that the very high "thermal" cross-sections shown are not achieved. Furthermore, the Pu-239 resonance prevents its cross-sections from diminishing as much with increasing energy as those for U-235. The neutron spectrum in a light-water reactor also changes with irradiation. The data here are for the spectrum in a generic PWR, with fuel initially enriched to 3.5 percent U-235, after 22 megawatt-days per kilogram of heavy metal (MWd/kgHM) exposure (Schnitzler 1994).

^d Graphite-moderated molten-salt accelerator-driven system of 500 megawatt-thermal (MWt) nominal power, with flux of 2.1×10^{14} neutrons per square centimeter per second (n/cm² sec), after five years (Venneri 1994).

^e At an incident neutron energy of 2 MeV, however, U-238 has a fission cross-section of about 0.5 barn and a ν of 2.46. Thus the fission of U-238 by neutrons in the high-energy "tail" of the neutron-energy distribution contributes some energy in fission reactors and in fission explosives, even though U-238 by itself cannot sustain a chain reaction (see text).

Plutonium-240 (and the other even-numbered isotopes of plutonium) can sustain a chain reaction only in a fast-neutron environment; these isotopes are not fissile, but they are usable fuels for fast-neutron reactors and for bombs. Uranium-238 cannot sustain a chain reaction even in a fast-neutron environment, because, on the average, fewer than one of the neutrons produced by the fission of a U-238 nucleus retains enough of its energy for long enough to induce another such nucleus to fission.

Nuclear bombs must rely on fast-neutron chain reactions, which may seem paradoxical at first because the fission-reaction probability (cross-section) is so much larger for slow neutrons than for fast ones. But, in a growing chain reaction that depends on the high reaction probability of thermal neutrons, time is required for the fast neutrons produced in fission to become thermalized through nonfission/noncapture collisions with nuclei; this introduces a sufficient lag in the growth of the chain reaction to prevent its reaching explosive proportions before thermal expansion reduces the fuel to subcritical density. Nuclear bombs and fast reactors compensate for the low fission cross-section for fast neutrons by means of a high density of the chain-reacting nuclei and a low density of neutron-absorbing materials.

Thermal reactors, by contrast, make use of neutron moderators—materials whose combination of low atomic number and low propensity to absorb neutrons permits rapid slowing of the neutrons with low losses to absorption—in order to establish a thermal-neutron spectrum in the reactor core and thus take advantage of the high fission cross-sections at thermal-neutron energies. This allows maintaining a chain reaction in fuel with much lower concentrations of U-233, U-235, or Pu-239 (and lower power density, hence greater flexibility with respect to cooling arrangements) than are required in a fast reactor. The best of all moderator materials are very pure graphite (impurities absorb neutrons) and heavy water (deuterium oxide). The advantage of heavy water compared to ordinary water is that deuterium—the one-neutron isotope of hydrogen—is much less likely to absorb a neutron than is the no-neutron isotope of hydrogen that most ordinary water molecules contain.

Reactors of suitable design using either graphite or heavy water as a moderator can sustain a chain reaction in natural uranium, despite its mere 0.7 percent of U-235. The use of ordinary water⁶ as the moderator, which is the most common choice in the world's power reactors, entails enrichment of the uranium fuel to a U-235 concentration typically between 2 and 5 percent. Typical fast-reactor designs would require U-235 concentrations of 20 percent or so in order to perform satisfactorily with U-235/U-238 fuel. Similarly, using

⁶ Ordinary water is called "light water" in nuclear jargon, to contrast it with heavy water which, as noted, contains one-neutron deuterium in place of the common no-neutron isotope of hydrogen. The two main types of reactor designs that are moderated with light water—boiling-water reactors (BWRs) and pressurized-water reactors (PWRs)—together constitute the class called light-water reactors (LWRs).

U-235/U-238 mixtures containing less than 20 percent U-235 for practical nuclear bombs would be extremely difficult, and at the 2–5 percent U-235 concentrations typical of today’s commercial-reactor fuel it is impossible. (From the standpoint of bomb design, the higher the U-235 concentration the better, and high-enriched uranium [HEU] produced for weapon purposes typically contains over 90 percent U-235.)

Because different isotopes of uranium are essentially identical chemically, the enrichment process for increasing the concentration of U-235 above its value in natural uranium depends on physical rather than chemical means of separation, most of which exploit the 1.3-percent difference in mass between U-235 and U-238 atoms. Such approaches typically involve first converting the natural uranium to uranium-hexafluoride gas, followed by separation of the isotopes using the difference in their diffusion rates through a “cascade” of thousands of porous barriers (gaseous diffusion plants) or using the effects of their differing inertial masses in other devices, such as very high-speed centrifuges.⁷

In general, enrichment technologies that can achieve the 2–5 percent U-235 concentration required for light-water reactor fuel can also be used, at additional cost, to produce the much higher U-235 concentrations required for nuclear weapons. But nearly all enrichment techniques demand sophisticated technology in large and expensive facilities, and lack of widespread access to this enrichment technology up to the present has been considered one of the primary technical barriers to the spread of nuclear weapons capability. This barrier, as with the similar one applying to reprocessing technology (see below), was of course never absolute, and it has eroded somewhat over time with the diffusion of relevant knowledge and, to some extent, hardware.

Reactor Cooling and Electricity Generation

Most of the energy of the fission reactions is deposited in the nuclear fuel within a short distance of the site of the fission’s occurrence. The rate of such energy deposition in a typical reactor is such that the fuel would overheat (losing its structural integrity) and then melt within a very short time unless provision is made for removal of the thermal energy by a coolant. From the time of the conception of the neutron chain reaction, it was realized that, by restricting the rate of extraction of heat, the coolant could be maintained at a steady elevated temperature and used to run a heat engine of some kind for the generation of mechanical work—as in an ordinary steam engine for propelling submarines or ships.

Most commercial reactors drive steam turbines coupled to generators for the production of electrical power. Attainment of satisfactory efficiencies—30

⁷ A more recently developed technique exploits differences in the excited atomic states of U-235 and U-238 to permit selective excitation by tuned lasers, followed by selective ionization and separation by an electric field.

percent or more—in the conversion of thermal energy to electricity in this way requires steam temperatures above 550° F (290° C), which implies, for effective heat transfer, a somewhat higher temperature at the surface of the fuel. The fluid chosen as coolant must have satisfactory heat-transfer properties at these elevated temperatures, must be chemically compatible—at these temperatures and the corresponding pressures—with the fuel or cladding with which it is in contact, and must have neutronic properties (i.e., propensities to slow down and to absorb neutrons) consistent with establishing in the reactor core an intensity and average energy of neutron flux that permits criticality in nuclear fuel of the chosen type.

The light-water reactors that dominate world nuclear electricity generation today employ ordinary water as both neutron moderator and coolant. Graphite-moderated reactors may be cooled with water or with gas, which was carbon dioxide in early gas-cooled reactors and generally is helium in more recent designs. In fast reactors, neither the coolant nor other materials in the reactor core may contain high densities of low-atomic-number elements, since the elastic collisions of neutrons with such elements rapidly slow the neutrons to energies where U-238 no longer fissions but does strongly absorb neutrons, with consequent reduction of reactivity. For this reason, water is an unsatisfactory coolant for fast reactors. Most contemporary fast-reactor designs employ liquid-metal coolants, such as sodium or lead.

The requirement for cooling the nuclear-reactor core is not restricted to the times when the chain reaction is underway, but extends afterwards because of the phenomenon of “afterheat.” This refers to the energy released by the radioactive decay of fission products, which process continues—albeit at a rate that declines with time—after the chain reaction has been shut down. In a reactor that has been operating at its rated power level for some time and then is quickly shut down, the afterheat power amounts initially to about 7 percent of the operating level. Thus a reactor that operates at a rated thermal power of 3,600 megawatts-thermal (MWt)—corresponding, at 33 percent thermal-to-electric conversion efficiency, to about 1,200 megawatts-electric (MWe)—would have an initial afterheat power of about 250 MWt.

This would be more than enough power, in most reactor types, to quickly overheat and then melt the fuel if cooling were not maintained following shut-down of the chain reaction. Since overheating and, to an even greater extent, melting, poses the threat both of significant damage to the reactor and of release from the fuel of some of the highly radioactive fission products, reactors must be provided with an emergency core-cooling capability that can be relied on if the normal cooling system becomes disabled. In most contemporary power reactors, the emergency core-cooling system involves “active” elements such as pumps, valves, and sensors that would work to provide and circulate coolant if an accident such as a pipe break allowed the normal coolant to escape. A number of newer reactor types—including some modified light-water reactor (LWR)

designs, some liquid-metal reactors (LMRs), and the modular high-temperature gas-cooled reactor (MHTGR)—are designed to be able to avoid significant damage or radioactivity release, from any plausible loss-of-coolant accident, based on “passive” heat-removal mechanisms alone, with no or minimal reliance on the proper functioning of any pumps, valves, sensors, or switches.

Nuclear reactors intended not for electricity generation but only for the production of WPu have no need for the high temperatures associated with efficient thermal-to-electric energy conversion. Such “production” reactors can therefore be designed to operate at lower power densities and temperatures, which alleviates a number of design problems, greatly eases the problem of cooling the reactor core in both normal operation and emergencies, and, thus reduces costs as well as increasing safety. The same advantages would be available in a reactor with the sole function of burning plutonium, without electricity generation, as a means of disposition of this material. (Whether these advantages are worth the extra costs and delays of building such a reactor is discussed in Chapter 6.)

Energetics and Fuel Consumption

Of the energy released by the fission of a uranium or plutonium nucleus, about 80 percent is in the kinetic energy of the two main fission fragments, some 3 percent is in the form of “prompt” gamma emissions from the excited fission-product nuclei, about 2.5 percent is carried by the neutrons resulting from fission, another 2.5 percent is gamma emissions resulting from the capture of these neutrons in surrounding materials, and about 12 percent materializes subsequently as the energy of gamma, beta, and neutrino emissions from the radioactive fission products. In characterizing the energy made available in a nuclear-fission chain reaction, it is customary to include, in the case of a nuclear reactor, all of the energy forms just mentioned except the neutrinos (since these escape the reactor carrying their energy with them).

The available energy, by this definition, is about 200 MeV per uranium nucleus and about 210 MeV for a plutonium nucleus. Converting to everyday units gives 82 gigajoules (GJ), or 0.95 megawatt-days (MWd) of thermal energy per gram (g) of U-235 that is fissioned; the figure for thermal fission of Pu-239 is 0.98 MWd/g. In calculating the yields of nuclear weapons in terms of the equivalent quantity of high explosives, neither the neutrinos nor the delayed forms of energy release are counted, which gives about 175 MeV per nucleus or 72 GJ/g of U-235. With the customary value of 1,000 kilocalories per kilogram (kcal/kg) or 4,184 kilojoules per kilogram (kJ/kg) for the explosive energy of TNT, this makes the complete fission of 1 kg of U-235 in a bomb equivalent to 17 kilotons of TNT.⁸

⁸ Since fission bombs are not 100-percent efficient—that is, not all of the fissionable material they contain undergoes fission—actual yields are less than 17 kilotons per contained kilogram of fissionable material.

The foregoing numbers lead to the widely used approximate rule of thumb that all reactors, irrespective of details of type and fuel, fission about one gram per day of heavy nuclei (and produce about one gram per day of fission products) per megawatt of average thermal output. This ratio—one megawatt-day of output per gram fissioned—means that a large modern power reactor of nominal 1,200-MWe capacity, thermal-to-electric generating efficiency of 33 percent, and annual average capacity factor of 75 percent—providing $1,200 \text{ MWe} \times 365 \text{ days} \times 0.75 / 0.33 \text{ MWe/MWt} = 1$ million MWd of thermal energy per year—will fission 1 million grams or 1 metric ton (MT) of heavy nuclei per year. A reactor of this size that derived all of its energy from plutonium, then, would fission a ton of plutonium per year, and this relation provides a helpful metric for the WPu quantities addressed in this report: 100 tons of WPu represents the amount of fissionable material consumed by 100 large power reactors in a year. In mid-1993, world nuclear-power capacity was equivalent to more than 270 such reactors (see the latter part of this chapter).

The preceding figure does not mean, however, that it is feasible to destroy 100 tons of surplus WPu in the space of a year using a third or so of the world's power reactors. The technical reasons this is not possible (leaving aside logistic and institutional obstacles to such an approach) are that (1) most reactors unavoidably produce plutonium at the same time as, although in most reactor types at a slower rate than, they burn it; and (2) it is not possible, in general, to burn up all of the plutonium (or other fissionable material) that is in a reactor at any one time.

The production of plutonium occurs, as described above, in consequence of the absorption of fission neutrons in the principal isotope of natural uranium, U-238. This process is called “fertile-to-fissile conversion,” with U-238 correspondingly termed a “fertile” material. The customary index of the rate of production is the “conversion ratio,” CR, defined as

$$\text{CR} = (\text{fissile atoms produced}) / (\text{fissile atoms destroyed}),$$

which can be calculated as a ratio of instantaneous rates or, more practically, in terms of total production and destruction over a period of time.⁹ The LWRs that dominate world nuclear-energy generation today have conversion ratios around 0.6, while the heavy-water reactors in commercial use in Canada and a few other countries have conversion ratios in the range of 0.7 to 0.8. Any reactor with a conversion ratio greater than unity is called a “breeder reactor,” and the conversion ratio is then called the “breeding ratio”; reactors with a conversion ratio less than unity are called “burners” or “convertors.” Of the new fissile material produced in a reactor by the conversion of U-238 to Pu-239 (or, in the

⁹ The definition's generality allows it to account for production of fissile Pu-241 by neutron absorption in Pu-240 as well as production of Pu-239 by neutron absorption in U-238, and it is applicable as well to the “thorium cycle,” in which U-233 is produced by neutron absorption in thorium-232.

thorium cycle, by conversion of thorium-232 (Th-232) to U-233), part gets fissioned while still in the reactor—contributing to energy output and reducing the quantity of fissile material that otherwise would need to be provided in fresh fuel—and part is discharged from the reactor in the spent fuel.

The only way to achieve conversion ratios near zero, as would be essential if the aim were to burn up surplus WPU without generating new fissile plutonium or U-233 in the process, would be to use fuel containing no U-238 or Th-232. Such “nonfertile” fuels are possible and have served as the basis for some high-temperature gas-cooled reactor (HTGR) designs. But, as we discuss later, their development for use in existing light-water, heavy-water, or fast reactors would require considerable effort and a corresponding investment of time and money. While nonfertile fuels for HTGRs are closer to availability, more development and higher costs would be involved in using these reactors than in using existing reactor types with fertile fuels. Even with nonfertile fuels in hand, it would not be easy to burn an initial stock of WPU down to zero, for reasons to which we now turn.

An important part of the difficulty of “burning up” fissile material completely is that, for fertile and nonfertile fuels alike, the fuel tends to lose either its structural integrity or its capacity to sustain a chain reaction long before its fissile content is exhausted. It tends to lose its structural integrity because of the combined effects of cyclic thermal stresses, corrosion, and the structural damage caused by fission neutrons and the fission-product tracks, as well as the problems posed by the pressure from gaseous fission products; when these take too high a toll, the result is excessive leakage of fission products into the reactor coolant, generating problems in maintenance and compliance with environmental standards.¹⁰ Or, before the fuel starts to lose structural integrity, its reactivity may fall below the level required, because of the combination of diminishing density of fuel nuclei as these are burned up and growing density of fission products, some of which are strong neutron poisons (absorbers). The amount of fission energy derivable from fuel before this happens can be increased—within the limits of fuel structural integrity—by increasing the initial concentration of fissile nuclei. This measure may require the addition to the fuel of “burnable poisons” to offset the high reactivity that would otherwise be associated with the high initial fuel density.¹¹ It may be attractive for economic reasons (to reduce the amount of fuel that must be fabricated for a given energy output)—or to

¹⁰ Of course it is always possible to manufacture the fuel to be tougher, but the extra structural material entailed in doing so tends to degrade both the neutronic and the heat-transfer properties of the fuel, at the same time that the extra material and extra care in manufacturing increase the costs.

¹¹ Burnable poisons absorb neutrons, and thus hold down reactivity, early in the fuel’s operating life when the density of fuel nuclei is high and that of fission-product neutron poisons is low; as time goes on, the absorbing capacity of the burnable poisons is used up, ideally at a rate that just compensates for the buildup of fission products and burndown of fuel nuclei.

increase the rate at which WPu can be loaded into reactors of a given size—but it does not reduce the amount of plutonium still remaining in the fuel when it is finally spent.

The quantity of energy generated by a given batch of fuel before it is considered spent is called the “discharge burnup” (or, alternatively, the “discharge exposure” or “discharge irradiation”). This is customarily measured in megawatt-days of thermal energy per metric ton of initial heavy metal (MTHM or just MTHM), where “initial heavy metal” refers to the quantity of uranium, plutonium, and (sometimes) thorium and heavier elements in the fuel at the time it is first loaded into the reactor. Typical average discharge burnups for commercial LWRs are in the range of 25,000 to 40,000 MWd/MTHM, with the most recent fuel designs achieving the higher figures.¹² Canadian heavy-water moderated reactors (called CANDU, for Canadian deuterium-uranium) using natural uranium fuel achieve discharge burnups of about 7,000 MWd/MTHM; LMRs with fuel enrichments of 20–30 percent U-235 or plutonium achieve figures in the 100,000-MWd/MTHM range; and HTGRs are being designed to use fuel with enrichments above 90 percent to achieve discharge burnups of 500,000 MWd/MTHM and higher.

If every gram in an initial metric ton of heavy metal in fuel were actually fissioned, at 1 MWd/g the total burnup would be 1,000,000 MWd/MTHM. Correspondingly, each 10,000 MWd/MTHM of burnup represents the fission of about 1 percent of the heavy metal atoms initially present. Thus, for example, low-enriched uranium (LEU) fuel that achieves a discharge burnup of 33,000 MWd/MTHM starting with a U-235 content of 3.3 percent has fissioned altogether about 33 kg or 3.3 percent of the heavy atoms initially present in each ton of fuel, much of the fission occurring in the initial 33 kg stock of U-235, but a significant contribution coming from the fission of plutonium produced during the reactor’s operation by the absorption of neutrons in U-238. Thus, for example, a 1,200-MWe LWR fueled with LEU at 3.3 percent U-235, and running at a capacity factor of 75 percent with a discharge burnup of 33,000 MWd/MTHM and thermal efficiency of 33 percent, would load annually about 30 MTHM (containing 1,000 kg of U-235), and would discharge, in spent fuel (after three years), about 250 kg/yr of U-235 and 300 kg/yr of plutonium, two-thirds of it the fissile Pu-239 and Pu-241 isotopes. A breeder reactor with the same electrical output would discharge 1,200–2,000 kg of plutonium per year, depending on fuel characteristics and operating mode.

If the fuel is not LEU but plutonium-based to start with, and the burnup and the conversion ratio are about the same, the plutonium content of the spent fuel necessarily will be higher. Even in a plutonium fuel that contains no fertile material from which new plutonium is produced in operation, the spent fuel will

¹² Increasing burnups attained by commercial LWRs to 50,000 MWd/MTHM is undoubtedly feasible and may be economic.

always contain a substantial amount of unburned plutonium (which is the same as saying that discharge burnup will fall substantially short of 1,000,000 MWd/MTHM), because if the plutonium residual gets too small the reactivity of the fuel will fall to the point that the chain reaction cannot be maintained.

Fuel Preparation, Refueling, and Reprocessing

The composition and configuration of nuclear fuel varies considerably across different types of reactors, not only in respect to the fractional content of fissile isotopes, as mentioned above, but also in terms of chemical composition (where the main possibilities include heavy-metal oxides and carbides, metallic fuels, and molten salts) and in the size, shape, and cladding of the individual fuel elements. In LWRs, the fuel consists of uranium-dioxide or mixed uranium and plutonium oxide (MOX) pellets stacked in long, slender tubes of a zirconium alloy (typically 4 meters in length and 1 centimeter in diameter); these are fitted into rectangular arrays of 60–260 rods called fuel “assemblies,” in which the rods are spaced so as to allow coolant to flow between them.

Preparation

The preparation of fuel begins with the acquisition of the heavy metal, which in the case of uranium can be by mining and refining of ore or by reprocessing of previously irradiated fuel (discussed in more detail below), and in the case of plutonium can only be by reprocessing since there are no plutonium ores. (In the case of acquiring plutonium from surplus nuclear weapons, the origin of the plutonium was also reprocessing of irradiated reactor fuel, albeit at some considerably earlier time.) Fuel preparation may (or may not) also include enrichment of U-235 content above that of natural uranium, for which purpose a precursor “conversion” step from uranium oxide in ore to uranium-hexafluoride gas (UF_6) is required. Other preparation steps might include reconversion of the UF_6 gas to oxide (if uranium enrichment was undertaken) and conversion of plutonium metal to oxide (if plutonium from nuclear weapons is to be used in MOX fuel).

The further steps termed “fuel fabrication” would include forming oxide pellets, preparing the cladding tubes and loading the pellets into them, and joining the fuel rods into fuel assemblies. Preparation of the metallic fuel used by some fast-neutron reactors would require a similar amount of processing, lacking only conversion to oxides. Preparation of the particulate carbide fuels used in the HTGR would entail completely different technology, as would the preparation of fuel for molten-salt and particle-bed reactors.

Refueling

In most reactor types, refueling is performed in batches with the reactor shut down. Under a typical fuel management scheme, a quarter or a third of the fuel in the core would be removed during an annual refueling shutdown, and a corresponding quantity of fresh fuel would be added. If a third of a core is changed out each year, the residence time of a batch of fuel until it accumulates its full discharge burnup will be three years. A few reactor types, such as the Canadian heavy-water reactors (CANDUs) and the Russian RBMK reactors (the type that suffered the accident at Chernobyl), have the capacity to be refueled continuously, fuel element by fuel element, while the reactor is running; and some reactor types that have been contemplated but not built at commercial scale would use fluid fuels or particle-bed fuels to attain continuous refueling capability. In the context of options involving reactor “spiking” of plutonium or efforts to eliminate plutonium nearly completely through repeated recycling, these different fueling characteristics can assume considerable importance in governing the trajectories of in-reactor and out-of-reactor plutonium inventories versus time.

Reprocessing

At the same time that the first reactors were built (during World War II) for production of WPu, technology was developed for chemical reprocessing of the irradiated fuel to extract the plutonium it contained. This technology involved chopping up the irradiated fuel, dissolving it in acid, and performing a series of chemical operations on the resulting radioactive brew in order to separate the heavy metal from the fission products and, within the heavy metal fraction, the uranium from the plutonium. These separations, while not exceptionally difficult in chemical terms, are also not simple, and they are greatly complicated by the shielding and remote-handling requirements imposed by the intense radioactivity of the fuel, as well as by the need to minimize emissions of radioactivity to the environment.

The facilities built during World War II to perform such “fuel reprocessing,” as it is called, were large, complex, and costly. This remained true in the postwar period, and although a number of different reprocessing technologies have now been tried or envisioned, all are (or would be) technically challenging and costly to build and to operate for commercial use. These challenges and costs have been considered one of the significant barriers to the acquisition of a plutonium-based nuclear-weapon capability by additional countries, but that perception needs to be tempered with recognition of two points. First, the technology has by now become rather widespread, with reprocessing plants known

to exist in 14 countries.¹³ Second, much of the cost and technical sophistication involved is associated with achieving high standards of purity in the separations while meeting stringent environmental, safety, and worker-exposure standards; organizations willing to be less fastidious about these matters could perform reprocessing with considerably less elaborate and less costly facilities (see, for example, OTA 1977; Comptroller General 1978; NAS 1994, pp. 150–151). In addition, spent fuel which has cooled for many years would be significantly easier for a potential proliferator to reprocess, because of the lower radioactivity (which lowers doses to workers and radiation degradation of solvents, and ultimately would allow reprocessing with only limited contact maintenance), and the greater simplicity of the chemistry after some of the more difficult to separate fission products (such as zirconium and niobium) have decayed to insignificant levels.

Up until now, nuclear fuel reprocessing capabilities in the United States have seen more use in nuclear-weapon programs than in civilian power generation. A combination of slower than expected growth of civilian nuclear-energy generation after the mid-1970s and continuing discoveries of cheap new reserves of uranium led to deferral of the transition to reprocessing, recycle, and breeder reactors.¹⁴ Neither reprocessing and recycle for convertor reactors (which reduce the uranium-ore and enrichment requirements of such reactors by about a third) nor breeder reactors (which greatly expand the supply of nuclear fuel but require reprocessing and recycle in order to do so) can compete economically with the “once-through” use of LEU in convertor reactors if uranium-ore and uranium enrichment are as inexpensive as they are today.

It is often argued, in favor of fuel reprocessing and plutonium recycle, that this technology will reduce significantly the problems associated with radioactive waste management—at least by reducing the volume of high-level wastes and wastes from uranium-ore processing, and possibly also by permitting the separation of different waste components to enable transmutation or other special treatment of the most long-lived isotopes. If reduced costs of waste management were the result, reprocessing could become economically competitive with once-through fuel cycles sooner than would otherwise be the case.¹⁵

¹³ Most of these are, to be sure, under International Atomic Energy Agency safeguards, as discussed elsewhere in this report.

¹⁴ It was also the case that the U.S. government, during the Carter administration, had a policy of discouraging plutonium recycle and the deployment of breeders because of proliferation concerns; but it is clear that the other factors were the primary ones, because the reversal of the Carter policy in the Reagan and Bush years had no effect.

¹⁵ The assumption of large savings in waste management costs because of reprocessing, for example, is a major factor in the much-quoted conclusion of a 1994 study by the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD 1994) that reprocessing is already economically competitive or close to competitive with once-through use of LEU.

It is not clear at this stage of the technology, however, that reprocessing and recycle would in fact reduce the magnitude of the nuclear-energy waste management problem or the costs of managing it. The pros and cons of reprocessing (and subsequent processing, e.g., transmutation) for waste management have been under intensive investigation in an ongoing National Research Council study (in which panel member Thomas Pigford is a participant).¹⁶ Without wishing to try to anticipate here all of that study's findings, we suggest that it is not prudent, on present evidence, to assume that fuel reprocessing would generate significant cost savings in waste management.

Plutonium recycle is being practiced in commercial LWRs and a few breeder reactors in Europe, and reprocessing plants have been built there to support those operations. Japan has an extensive research and development program underway to prepare for similar plutonium recycle activities.

If energy from nuclear fission is to play a significant role in the long-term energy future of the planet, exploiting the large energy-efficiency advantage of breeding and reprocessing (as much as 50–100 times more energy extractable per kilogram of uranium than with convertor reactors) eventually will become economically justifiable (unless uranium can someday be recovered from seawater at low enough cost to make its use cheaper than a breeding cycle). How far in the future this might occur depends on many factors—the size of future growth of electricity demand, the availability of attractive nonfission power sources, the size of uranium resources versus their cost of extraction, the relative construction costs of breeder and nonbreeder reactors—that are beyond the scope of our inquiry here. We will only say, therefore, that based on current evidence and arguments as we understand them, it would not be prudent to assume—in thinking about options for WPu disposition—that civilian nuclear fuel reprocessing will spread substantially beyond the few countries that now practice it in the time period extending 30–50 years from the present, with which we are mainly concerned.

We *do* take up the question, later in this report, as to whether the deployment of fuel-reprocessing capacity explicitly dedicated to the mission of WPu disposition would be desirable. As will be seen, the role of reprocessing and recycle in the plutonium disposition context would be to enable the destruction by fission, in multiple passes through reactors, of a significantly higher fraction of the WPu than could be fissioned in once-through reactor disposition. We call attention here just to one technical characteristic of reprocessing operations, which turns out to be very important in determining the pace at which WPu inventories could be consumed in reprocessing-based approaches. This characteristic is the duration of the cooling-off period between discharge of the spent fuel from the reactor and the beginning of reprocessing operations on it. In the batch

¹⁶ The forthcoming National Research Council study is described in a 1992 interim report (National Research Council 1992); see also Ramspott et al. (1992).

reprocessing approaches used so far for reprocessing high-burnup civilian power fuels, the cooling-off period has generally been one to two years or more. Shortening this time substantially, as would seem desirable if the purpose of the activity is rapid destruction of an inventory of surplus WPu, poses technical problems because of the more intense radiation field of the hotter fuel. So-called “online” reprocessing schemes, which have been proposed for the continuous (as opposed to batch) reprocessing in fluid fuel or particle-bed reactor systems for burning plutonium and other actinides, face particularly difficult problems in this respect.

Some Differences Between Plutonium- and Uranium-Based Reactor Fuels

An aspect of the physics of fission that is very important to the way the reaction is controlled in power reactors is the existence of “delayed” neutrons, which are produced not in the fission reaction itself but in the subsequent radioactive decay of some of the shorter-lived fission products. These delayed neutrons amount to about 0.7 percent of all of the neutrons associated with the thermal fission of U-235 and a fraction about three times smaller of the neutrons associated with the thermal fission of Pu-239. (The delayed-neutron fraction from fission of Pu-240 is intermediate between that for Pu-239 and that for U-235.) A reactor in a condition such that criticality depends on the delayed neutrons—that is, in which the “prompt” neutrons are not enough in themselves to produce an average of one new fission for each fission that takes place—will have an effective time between neutron generations very much longer than the fraction of a second between a neutron’s production and its absorption to cause another fission.¹⁷

This means, in turn, that the characteristic time on which changes in the reactor’s power level take place is much longer than it otherwise would be, making possible the dynamic control of the power level by means of the mechanical movement of control rods into and out of the core. Because the delayed-neutron fraction is smaller in plutonium fission than in fission of U-235, the performance demanded of control systems is somewhat greater in a plutonium-fueled reactor than in a uranium-fueled one; and because the delayed-neutron fraction in Pu-239 is smaller than in Pu-240, the additional demands on control systems are largest for those plutonium fuels in which the Pu-239 content is highest.

Another factor that contributes to additional control requirements for plutonium-based fuels is that plutonium’s higher thermal-neutron absorption cross-section, compared to that of U-235, leads to higher average energy of the neutron spectrum in plutonium fuels. (Higher absorption of the slowest neutrons

¹⁷Typically, minutes rather than microseconds. In a nuclear bomb, criticality needs to be achieved based on prompt neutrons alone, but this “prompt critical” condition must be assiduously avoided in power reactors.

means the average energy of the neutrons that are not absorbed will be higher.) This reduces the absorbing power of the control rods, since the absorption cross-sections of the materials in the control rods are lower at higher neutron energies. The more energetic neutron spectrum (i.e., the higher flux of fast neutrons for a given rate of production of fission heat) in plutonium fuel will also tend to increase the rate of radiation damage to structural materials in and around the core, which can bear on the operating lifetime of these materials and, even, in some circumstances, on safety.

Yet another distinction between plutonium- and uranium-based fuels involves a sharp increase in the fission cross-section of Pu-239—a “resonance”—at neutron energies near 0.3 eV. This resonance poses the problem that heating of the core from an increase in reaction rate tends to increase the reaction rate still further by pushing some of the neutrons upwards in energy into the resonance region, thus promoting the undesirable form of feedback known as a positive temperature coefficient of reactivity. This tendency is most pronounced when there is little Pu-240 present. Depending on other characteristics of the fuel and the moderator that also influence the temperature coefficient of reactivity, it may be necessary to add to plutonium-fueled reactor cores some burnable neutron absorbers (preferably those with resonance absorption in the 0.3 eV region, such as erbium) to offset the extra reactivity during the period when the fuel is relatively fresh (before neutron absorption in Pu-239 has built up the Pu-240 content).

Plutonium-based fuel also exhibits levels of heating from radioactive decay somewhat higher than those exhibited by uranium-based fuels that have experienced an equal quantity of fission. In principle, this puts greater demands on post-shutdown (including emergency) core-cooling capabilities, as well as on cooling for spent fuel storage, but in practice the difference is small compared to the performance margins with which such systems are designed.

By virtue of the relatively high fission cross-section of Pu-239 in a fast-neutron spectrum and, especially, by virtue of this isotope's high ratio of neutrons produced to neutrons absorbed in such a spectrum (Table 2-1), plutonium-based fuel offers superior performance in fast-neutron breeder reactors compared to fuel enriched in U-235. Since plutonium fuel offers only comparable nuclear performance to uranium fuel in thermal reactors but superior performance in fast-breeder reactors, it might be argued that the latter represent the best use of plutonium for nuclear power. But because fast-breeder reactors seem unlikely to be economically competitive with thermal-convertor reactors for some decades to come, saving WPu until this happens would impose a severe time delay on its final disposition. It is relevant, in this connection, that if plutonium were unavailable for the purpose at the time in the future when fast-breeder reactors started to become competitive, the breeders could be started up with uranium fuel enriched to roughly 20 percent U-235; doing this would impose some performance penalty until the breeders built up their own plutonium supply, but

it would work (APS 1978). Alternatively, breeders could be started with plutonium separated at that time (given that abundant supplies of plutonium in spent fuel will continue to be available), rather than plutonium stored in separated form between now and then.

Some Relevant Aspects of Weapon Science

In the construction of the nominally 50,000 nuclear weapons in the world, U-235 and Pu-239 have been used as the principal fissionable materials to sustain the fast-neutron chain reaction essential to nuclear explosions.¹⁸ Highly enriched uranium (in weapons, typically 90 percent U-235 or more) can be used in either gun-type nuclear weapon designs like that used at Hiroshima or in the more efficient implosion design (Serber 1992). Because of the plutonium's higher neutron background from spontaneous fission, plutonium can be used only in implosion weapons. These background neutrons can cause fissions in a weapon's fissile material and cause the nuclear chain reaction to begin before the designer intended, a circumstance known as "preinitiation." In a worst case, in which the chain reaction begins the instant the material is sufficiently compressed to sustain such a chain reaction—the instant known as "first criticality"—most weapons will have a "fizzle yield" considerably less than the design yield that would be achieved if the chain reaction were initiated at the optimum moment.

This preinitiation problem is the single greatest difference between using reactor-grade and weapons-grade plutonium in nuclear weapons. Normal weapons-grade plutonium contains about 6 percent Pu-240, an isotope with a much higher rate of spontaneous fission (and hence higher neutron generation) than Pu-239. In the case of reactor-grade plutonium, with a typical Pu-240 content in the neighborhood of 25 percent, the rate of spontaneous neutron generation and hence (for a given design) the probability of achieving only a fizzle yield is several times greater.

But the fizzle yield is not zero. In the case of the Trinity implosion weapon, this fizzle yield would have amounted to about 1 kiloton (Mark 1993). As indicated in the CISAC plutonium study (NAS 1994), moreover, virtually *any* mixture of plutonium isotopes can be used to fabricate an implosion weapon that will dependably give the fizzle yield or more.¹⁹ For a fizzle yield of 1 kiloton, the destructive radius (for instance, the distance to which the blast would produce overpressures at or above the level of 5 pounds per square inch (psi) asso-

¹⁸ All nuclear weapons include a "primary" nuclear-explosive component in which the nuclear-energy release comes predominantly from the fission of plutonium or highly enriched uranium. In thermonuclear weapons, a "secondary" nuclear-explosive component that derives its energy from a combination of fusion and additional fission is ignited by energy from the "primary."

¹⁹ The only exception would be virtually pure Pu-238, in which the very high rate of heat generation by radioactive decay would pose practical problems for weapon design that would probably be insurmountable.

ciated with severe damage to industrial and residential buildings) would be more than one-third that of the Hiroshima bomb; a device with such a yield would be a fearsome terrorist weapon, capable of destroying a substantial part of a city at a single stroke. The 1994 CISAC report indicates, moreover, that with technical sophistication greater than is likely to be available to terrorists, nuclear weapons can be made from reactor-grade plutonium with reliable yields considerably higher than 1–2 kilotons.

Reactor-grade plutonium poses difficulties for bomb-makers beyond that of preinitiation. Chief among these are the problems of coping with extra heat from the radioactive decay of the shorter-lived isotopes of plutonium that are more abundant in reactor-grade than in weapons-grade plutonium, and the problems of gamma radiation from extra americium-241, which is the daughter product of 14-year half-life Pu-241 and hence more abundant in reactor-grade plutonium than in weapons-grade plutonium.

The magnitudes of the main differences between reactor-grade and weapons-grade plutonium are indicated in Table 2-2. While the collective impact of these differences on the task of the bomb-maker is surely not negligible, neither should it be overstated. In a recent article on this subject (Mark 1993, p. 122), a former leader of the Theoretical Division at the Los Alamos National Laboratory expressed the view that “The difficulties of developing an effective [weapon] design of the most straightforward type are not appreciably greater with reactor-grade plutonium than those that have to be met for the use of weapons-grade plutonium.”²⁰

As the CISAC committee noted in its 1994 report and as we emphasize in Chapter 3 of this report, it is necessary to distinguish, in assessing security threats, among the different kinds and degrees of vulnerability of plutonium in different forms to acquisition and misuse by terrorists, versus acquisition and misuse by small- to medium-sized countries, versus misuse by the nation from whose arsenals the plutonium came in the first place. Terrorists might care little about the differences between reactor-grade and weapons-grade plutonium; nations would be likely to care more, in the sense of preferring to make weapons from WPu if all else were equal, but they would also be better able to minimize the penalties of working with reactor-grade plutonium if circumstances made it desirable for them to do so.

²⁰ We emphasize that Mark’s statement does not say designing effective nuclear weapons is easy, which it is not; his point is rather that the *extra* difficulties imposed by the use of reactor-grade plutonium do not add a great deal to the difficulties that bomb designers must face in any case.

TABLE 2-2 Properties of Weapons-Relevant and Reactor-Relevant Mixtures of Plutonium and Uranium

	Weapons-Grade Pu	Reactor-Grade Pu		Weapons-Grade U (HEU)	Reactor-Grade U (LEU)	Natural U
Composition (g/g)						
Pu-238	0.0001	0.0130	U-234	0.0012	0.00025	0.000057
Pu-239	0.9380	0.6030	U-235	0.9400	0.03500	0.007193
Pu-240	0.0580	0.2430	U-238	0.0588	0.96475	0.992750
Pu-241	0.0013	0.0560				
Pu-242	0.0002	0.0500				
Am-241 ^a	0.0022	0.0350				
Specific activity (curies/g)	0.22	6.2		9.5×10^{-6}	1.9×10^{-6}	7.0×10^{-7}
Neutron background (n/g-sec)	52	340		1.4×10^{-3}	1.2×10^{-2}	1.3×10^{-2}
Heat generation (watts/kg)	2.5	14		2.6×10^{-4}	5.4×10^{-5}	1.9×10^{-5}
Surface gamma dose ^b (rem/hr)	0.94	15		1.5×10^{-3}	5.7×10^{-5}	1.2×10^{-5}
Dilution volume ^c						
in air (km ³ /g)	4,200	29,000		0.18	0.038	0.013
in water (m ³ /g)	4.2×10^6	2.8×10^7		32	6.5	2.3

^a Americium-241 is included here because its buildup from the decay of Pu-241 (half-life 14 years) significantly influences the heat output and gamma dose from both weapons-grade and reactor-grade plutonium. It is also a nuclear-explosive material, with a critical mass around 100 kg (Mark 1993). The same reference gives the concentration of Pu-241 + Am-241 as 0.0035 in WPu and 0.091 in reactor plutonium. The distributions shown here assume that the WPu is 20 years old and that the reactor plutonium is 10 years old. (Since production of reactor plutonium has been growing with time while that of WPu has been declining, the average WPu is older than the average reactor plutonium.)

^b The surface gamma dose is calculated for a sphere incorporating several kilograms of alpha-phase metal, using the approximation that D (grays/hr) = $0.5 \times D_a \times \mu_t / \mu_m$, where D_a is the rate of gamma-energy release in the solid material (J/kg-hr), μ_t is the mass energy-absorption coefficient for tissue at the 0.06 MeV energy of Am-241's gamma ray, and μ_m is the mass energy-absorption coefficient for plutonium or uranium metal (6.03 cm²/g and 5.78 cm²/g, respectively).

^c The dilution volume is the volume of air or water that would be required to dilute a gram of the indicated material to the concentration limits specified in the U.S. Code of Federal Regulations for continuous public exposure. It is a measure of radiological hazard potential.

This panel did not evaluate the likelihood that a national government would decide to use one or another type of fissile material, but focused rather on the obstacles to use and therefore the security risks associated with the different forms of fissile materials. CISAC reached the general conclusion that the differences in security risk between different grades of separated plutonium—that is, plutonium not mixed with radioactive fission products—are smaller than the

differences between separated and unseparated plutonium, or, stated another way, that the mixing of plutonium with fission products is a substantially bigger barrier to weapons use than is alteration of the plutonium's isotopic composition. Our own comparisons of the reactor-related options for WPu disposition proceed from that premise.

CLASSES OF DISPOSITION OPTIONS

As described in the 1994 CISAC report, the options for long-term disposition can be divided into three broad categories, which are identified as follows:

- “indefinite storage,” in which the material remains in weapon-usable form, with the only barriers to its reuse in weapons deriving from political commitments and security measures;
- “minimized accessibility,” in which physical, chemical, radiological, or isotopic barriers would be created to reduce the material's accessibility and usefulness for use in weapons; and
- “elimination,” in which the material would be removed essentially completely from human access (as, for example, by fissioning it so completely that less than a critical mass remained).

As noted in the introduction of the 1994 report, the committee rejected indefinite storage and concluded that minimizing the duration of intermediate storage of the excess WPu should be a key security criterion for judging disposition options, judgments with which this panel concurs. Thus the options within our purview are in the “minimized accessibility” and “elimination” categories. In essence, the options related to reactors and reactor wastes are designed either to create radiological, chemical, and isotopic barriers to the use of this material in weapons (by fabricating it into fuel and irradiating it in reactors, or by mixing it directly with radioactive wastes) or to fission it virtually completely (by repeated cycles of reactor irradiation and reprocessing).

More specifically, the options involving the use of plutonium in fission reactors can be divided into three general categories: the “spent fuel” option, the “reactor-spiking” option, and the “destruction” option. They are distinguished by their different end-points and different purposes, as follows.

- (a) The *spent fuel* option employs a once-through fuel cycle to process WPu into spent reactor fuel similar in its radioactivity, and in the isotopic composition of the contained plutonium, to the spent fuel that already exists (in much larger quantities) from civilian nuclear-power generation. Net destruction of plutonium (taking into account all plutonium isotopes) in this option ranges from slightly negative to a posi-

tive 80 percent.²¹ *The purpose of this approach is to create substantial physical, chemical, and radiological barriers to use of the WPu in nuclear explosives by the original owners of the material or by others. This would reduce the WPu management problem to a modest part of the reactor spent fuel management task that exists in any case.*²²

- (b) The *reactor-spiking* option uses briefer irradiation of WPu in nuclear reactors to process the plutonium more rapidly (but with correspondingly smaller change in its isotopic composition) into a constituent of a spent fuel form with enough radioactivity to provide a moderate degree of protection for a few years after discharge.²³ Net destruction of plutonium in this option is small. *The purpose of this approach is to achieve a degree of physical, chemical, and radiological protection for the entire stock of surplus WPu more quickly, or with fewer reactors, than in the spent fuel option.* The reactor-spiking option achieves this aim at the cost that the resulting degree of protection is not as high as

²¹ Negative net destruction—i.e., an increase in the total amount of plutonium present—occurs in a one-third MOX LWR core because new plutonium production from absorption of neutrons in U-238, which occurs in the entire core, exceeds the destruction of WPu by fission and transmutation, which occurs only in the one third of the core that is fueled with plutonium. In contrast, a 100-percent MOX LWR core operated to high levels of fuel burnup can yield a net plutonium consumption close to 50 percent. A high-temperature gas-cooled reactor, using nonfertile fuel at very high burnup, could achieve a once-through net consumption of plutonium of about 80 percent. (With respect to “negative net destruction,” it needs to be added that if the same reactor generated the same amount of energy without any input of WPu—i.e., using only LEU fuel—the amount of plutonium in the world would have increased even more: the new plutonium production from absorption of neutrons in U-238 throughout the core would not have been offset by consumption of *any* WPu. Clearly, the proper reference point for determining the net effect of reactor disposition of WPu on total plutonium inventories depends on whether the plutonium disposition occurs in a reactor that would have operated anyway, or in an additional reactor that displaces electricity from some non-nuclear source. See also the related discussion in connection with the “elimination” option, below.)

²² Given 100 tons of WPu and any practical rate of loading it into reactors, the amount of WPu in spent fuel would at no time be as much as 10 percent of the amount of civilian reactor plutonium in spent fuel: the total civilian plutonium in spent fuel was about 530 tons at the end of 1990, and continuing production in reactors (less separation from spent fuel by reprocessing) is increasing this figure at about 60–70 tons/yr; there is likely to be more than 1,100 tons of civilian plutonium in spent fuel by the year 2000 and no chance that as much as 100 tons of WPu will be in spent fuel by then.

²³ One standard that has been suggested for such protection is the level specified by the Nuclear Regulatory Commission, which describes self-protecting materials as producing “a total radiation dose in excess of 100 rems per hour at a distance of 3 feet from any accessible surface without intervening shielding” (OFR 1992, Parts 51–199, p. 284). This formulation can be questioned, however, on the grounds that (1) dose rates 3 feet from the end of a fuel assembly on its axis are very much lower than those 3 feet off its axis (because of self-shielding in the UO₂ and shielding by the assembly end-plates and other hardware), making the standard as stated impossible to meet at the low irradiations characteristic of the reactor-spiking approach; and (2) a 100-rem/h (roentgen-equivalent-man per hour) dose rate at 3 feet off-axis may not provide as much protection against theft as is desirable even in the short term.

in the spent fuel option. If the ultimate aim of disposition is to make the WPu at least as inaccessible for weapons purposes as plutonium in spent fuel from typical civilian power reactors, the reactor-spiking option cannot stand alone but must be followed by processing in either the “spent fuel” mode (as just described) or the “elimination mode” (described next).

- (c) The *elimination* option employs fission and transmutation in a nuclear reactor to convert nearly all of the plutonium processed into other elements. *The purpose of this approach is to eliminate the plutonium essentially completely from human access.*²⁴ Destruction fractions as high as 80 percent may be achievable without fuel reprocessing through the use of nonfertile fuels. Destruction fractions much above 80 percent are only achievable in practical systems through the use of fuel reprocessing and plutonium recycle. As described in the remainder of this report, the time required for such options, the technical uncertainties, and the costs involved are all very much larger than in the case of the spent fuel or spiking options.

A fourth approach, which we denote the “waste-spiking” option, achieves a result similar to those of the “spent fuel” or “reactor-spiking” options by processing the plutonium in waste management facilities rather than in reactors:

- (d) The *waste-spiking* option mixes the WPu with radioactive wastes from previous reactor operations—most probably with military reactor wastes—for storage and subsequent ultimate disposal by one of the schemes that will need to be selected for such wastes in any case. *The purpose of this approach is similar to that of the spent fuel option: to create substantial physical, chemical, and radiological barriers to further use of the WPu in nuclear explosives.* This way of doing so, unlike the spent fuel approach, does not change the isotopic characteristics of the plutonium.

Each of these categories has many variants—reactor types, operating schemes, waste types, and so on—and the different approaches and variants have different combinations of international security advantages and liabilities, as well as different combinations of advantages and liabilities with respect to technological readiness, economics, institutional requirements, and other factors. Characterizations of—and comparisons among—the different approaches and variants in these respects are presented in Chapters 3–7.

²⁴ Destruction is reckoned on a “net” basis, i.e., consumption minus production of all plutonium isotopes. An alternative criterion to the complete elimination of the WPu is to require that the total plutonium residue from the disposition campaign be just equal to the total plutonium residue that would have resulted from generating the same quantity of electricity from the same reactors *without* WPu disposition (see Garwin 1995).

PRESENT AND FUTURE FISSILE MATERIAL STOCKPILES

Excess Weapons Plutonium Stockpiles

How much excess WPu is there likely to be? As noted, current nuclear arms reduction agreements and pledges, if successfully implemented, would mean the retirement of tens of thousands of nuclear weapons. The U.S. Department of Energy has recently stated publicly that “up to approximately 50 metric tons of plutonium will (or may) become available by about 2005 . . . [for] civil (unclassified) purposes,” from both weapons dismantlement and other sources.²⁵ Similarly, the Russian government has indicated that it expects to have 50 tons of plutonium and 500 tons of HEU that are excess to its military needs. We will use 50 tons on each side as a nominal figure for the amount of plutonium requiring disposition. But it should be remembered that this number could grow depending on further disarmament initiatives and decisions concerning how much material should be retained in military reserves.

The schedule on which this material may become available for disposition depends on the planned schedule of arms reductions and the rate of dismantlement activities. Some WPu is already available, in principle. The United States and Russia have each indicated that they have already dismantled thousands of nuclear weapons over the last several years; the plutonium components of these weapons, containing many tons of plutonium, are currently in storage. The currently agreed arms reductions are to be carried out on an uneven schedule stretching to the year 2003; dismantlement of the retired weapons is ongoing, but may lag behind the retirements. In any case, it appears that none of the disposition options could be implemented quickly enough for the dismantlement schedule to be a major limiting factor in determining when disposition could be carried out.

Total World Plutonium and HEU Stockpiles

The plutonium and HEU resulting from arms reductions are only part of the world's stocks of these materials, which include:

1. Military plutonium and HEU in operational nuclear weapons and their logistics pipeline.
2. Military plutonium and HEU held in reserve for military purposes, in assembled weapons or in other forms.
3. Military plutonium and HEU withdrawn from dismantled weapons and considered excess.

²⁵ Willett (1993, p. 2). The uncertainty implied by the parenthetical “(or may)” reflects continuing debate within the U.S. government over how much of these materials should be kept as military reserves.

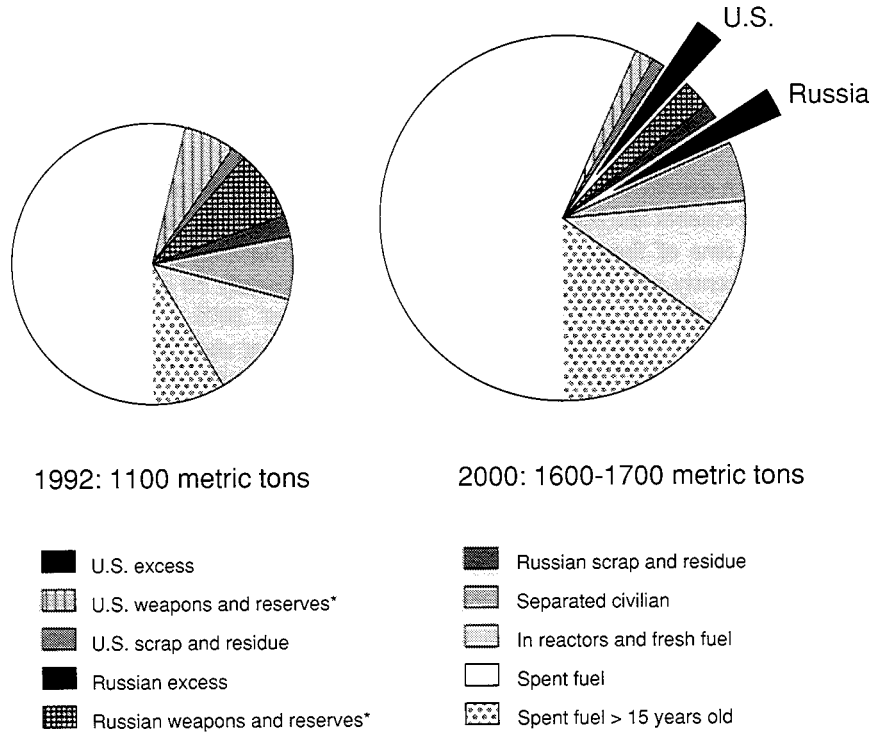


FIGURE 2-1 World plutonium stockpiles

* Includes weapons and weapon components from weapons already dismantled.

SOURCE: Committee estimates, based on IAEA (1993), Albright et al. (1993) and USDOE (1993).

4. Separated plutonium and HEU in storage in preparation for use in military or civilian reactors.
5. Plutonium and HEU currently in reactors.
6. Irradiated plutonium and HEU in spent fuel from reactors.
7. Military and civilian plutonium and HEU outside the categories above, including excess stocks, scrap, residues, and the like.

The problem of management and disposition of excess WPU (category 3) is the focus of this report, but policy for it must take into account the large stocks of plutonium and HEU in the other categories, since with varying degrees of difficulty they can all be used in nuclear weapons (see Figure 2-1.)

Official figures on total stockpiles of plutonium and HEU are generally not available. Quantities in military stockpiles are considered military secrets, and

quantities in civil stockpiles are often considered commercial secrets. Unofficial public estimates suggest that, at this writing, the total world stockpile of plutonium amounts to roughly 1,100 tons.²⁶

Military Plutonium Stockpiles

Roughly one-quarter of the world's plutonium stock is held in military stockpiles. The U.S. Department of Energy recently declassified information concerning the U.S. stock, indicating that the United States had produced 102 tons of military-related plutonium (of which 13 tons is fuel-grade, rather than weapons-grade).²⁷ Of this stock, 33.5 tons is in scrap, residues, and other forms at several sites in the nuclear weapons complex, leaving 68.5 tons that is either in intact weapons or in components from dismantled weapons now stored at Pantex near Amarillo, Texas. Russia is believed to have a somewhat larger total stock of military plutonium, but has not released comparable figures. Stocks held by the other declared nuclear-weapon powers (Britain, France, and China), and the threshold states (Israel, India, Pakistan, and North Korea) are small by comparison.

While the primary focus of this report is the excess WPu resulting from arms reductions—which is initially in the form of weapons components from dismantlement—both the United States and Russia also have large quantities of military plutonium in scrap and residues from past operations of their nuclear weapons complexes, most of which is also likely to be considered excess. While the amount of plutonium in these forms is smaller than the amount in pits that will result from arms reductions, the volume of the materials is much greater, the material is in many different forms, and for some of these the environment, safety, and health risks are substantial. Even characterizing the constituents of these materials accurately is difficult. Some of them can be processed readily to plutonium metal or oxide that could then be fed into many of the disposition options described in subsequent sections. Some reactor options (typically the more advanced ones that would take longer to bring on line) are more capable than others of handling variations in the form of the initial fuel feed, though there are materials that none of the reactor options could plausibly handle. Processing some of these materials into more tractable forms, moreover, would entail additional environment, safety, and health hazards. The vitrification option described in this report may be a particularly promising approach for stabilizing and ultimately disposing of the plutonium in these less tractable forms.

²⁶ These figures are adapted from Albright et al. (1993).

²⁷ This figure for the amount produced in the United States does not necessarily correspond to the amount in the current stock, as there have been transfers of plutonium from abroad.

Civilian Plutonium Stockpiles

The remaining three-quarters of the world's plutonium is held in civilian stocks. Recent International Atomic Energy Agency (IAEA) estimates indicate that as of late 1992, some 86 tons of this plutonium was in separated form, awaiting use in civilian reactors; in this form, if diverted, it would be directly usable in nuclear weapons (see below).²⁸

The remainder of the world's civilian plutonium stock, over 600 tons, is in spent reactor fuel or in reactors in some 26 countries. Not all spent fuel is equally inaccessible, however. When the fuel is in the reactor it would be difficult to divert, and when it first leaves the reactor its very high levels of radioactivity offer significant protection against diversion. Over the decades after leaving the reactor, however, the fuel's radioactivity will decay substantially, allowing it to be handled more readily and making it potentially more susceptible to diversion. Roughly 140 tons of the plutonium in the world's spent fuel was discharged prior to 1980, and is thus well over a decade old. During the first decade after discharge, the radioactivity of the fuel declines rapidly as the shorter-lived radionuclides decay; for about a century thereafter, the radioactivity of the spent fuel is dominated by the contributions of 30-year half-life cesium-137 and 29-year half-life strontium-90, and hence declines by about 50 percent every 30 years.

Projected Plutonium Stocks in the Year 2000

Plutonium production for the U.S. military stockpile has ceased, and the United States has proposed a global convention banning further production of plutonium and HEU for weapons. Military production in Russia has declined drastically, and continues at a low level only because the three reactors still operating provide needed heat and power to the surrounding areas. U.S.-Russian discussions of how best to terminate this production are underway. Thus, world military stockpiles in the year 2000 will be quite similar to the stockpiles today, except that many tens of tons of plutonium will have been made available if the dismantlement of nuclear weapons is carried out on schedule.

Future civil stockpiles involve somewhat greater uncertainties. Plutonium production in reactors can be projected with some confidence, but the future of both reprocessing of spent fuel and the use of the resulting plutonium in reactors

²⁸ See IAEA (1993). More than half of this accumulated plutonium belonged to Great Britain and Russia; while other reprocessing countries have decided to use plutonium in light-water reactors to reduce the buildup of excess stocks, neither of these countries has yet taken this route. Because of these civilian plutonium programs, an infrastructure of existing and planned civilian facilities exists to store many tons of plutonium, fabricate it into reactor fuel, and use it in reactors. These facilities, however, are already burdened with managing civilian plutonium; using them to handle excess military plutonium would require substantially expanding them or displacing the civilian plutonium in some way—an option discussed further in subsequent sections.

are currently the subjects of considerable controversy. The cumulative amount of plutonium discharged from the world's civilian power reactors will nearly double, from over 700 tons today to nearly 1,400 tons by the year 2000. As the rate of plutonium reprocessing continues to outpace the rate at which plutonium fuels are used in reactors, the civilian stock of separated plutonium will grow. Recent IAEA estimates suggest that the stock of separated civilian plutonium in storage may increase to between 110 and 170 tons by the latter part of this decade or early in the next century, depending on the scale of reprocessing and plutonium use over the intervening period.²⁹ In other words, it is very likely that in the early years of the next century the amount of separated plutonium in civil stocks will in fact be equal to or larger than the stocks of military separated plutonium freed from weapons as a result of arms reductions.

HEU Stockpiles

In addition to plutonium, there are large stocks of another directly weapon-usable material, highly-enriched uranium.³⁰ The total world stockpile probably amounts to roughly 1,500 tons. All but about 20 tons of this is in military stocks, either incorporated in weapons, held in reserve, or intended for use as naval reactor fuel. Over 95 percent of these military stocks are held by the United States and Russia. The United States and Russia have ceased production of HEU, and only very limited, if any, production continues elsewhere. The only likely changes in the world stockpile by the year 2000, therefore, will be decreases resulting from the use of existing stocks in naval and research reactors and the "blending down" of HEU to low-enriched reactor fuel.

WORLD NUCLEAR-ENERGY SYSTEMS RELEVANT TO PLUTONIUM DISPOSITION

Nuclear-Power Plant Types and Numbers

Table 2-3 summarizes the types and numbers of nuclear-power reactors worldwide as of the end of 1993.³¹ As indicated in the table, nearly 80 percent of the 419 power reactors operating at that time were light-water reactors

²⁹ IAEA (1993). Albright et al. (1993) provide roughly similar estimates.

³⁰ The security risks posed by this material before it is diluted to low-enriched uranium are even greater, in some respects, than those of plutonium: as noted earlier, HEU can be used in relatively simple, gun-type bomb designs, while plutonium cannot; and HEU is easier to handle and to conceal than plutonium. The emphasis in the 1994 CISAC study on the security risks of plutonium as opposed to those of HEU is based on the assumption that the easily accomplished prescription for "denaturing" excess HEU by dilution with natural or depleted uranium will in fact be promptly carried out in both the United States and Russia.

³¹ Adapted from *Nuclear News* (1994).

(LWRs), 9 percent were gas-cooled reactors (GCRs), and 8 percent were heavy-water reactors (HWRs). Of the 87 power reactors in various stages of partial completion, three-quarters were LWRs. Only 4 liquid-metal reactors (LMRs) were in operation at the end of 1993, with another 5 partly completed. For approximately 30 of the reactors of all types listed as partly completed, no construction is currently underway and plans for completion are indefinite.

In the United States, there were 109 operating commercial nuclear-power plants at the end of 1993, with a capacity of 99,400 MWe. Of these, 72 were pressurized-water reactors (PWRs) and 37 were boiling-water reactors. There are also 2 experimental LMRs in the United States: the EBR-II (20 MWe) and the Fast Flux Test Facility (FFTF) (130 MWe equivalent). Operation of both FFTF and EBR-II has recently been discontinued. Of the 7 U.S. LWRs listed as partly completed, none is actively under construction at this time. Five U.S. LWRs and one GCR that were shut down within the past five years have been decommissioned or are awaiting decommissioning.³² Forty-two nuclear-power plants are operating in the former Soviet Union, with a total capacity of 34,000 MWe. Of these, 25 are PWRs, 15 are light-water-cooled, graphite-moderated reactors (LGRs in Table 2-3, widely known as RBMKs), and 2 are LMRs. As indicated in Table 2-3, 25 of these reactors (13 PWRs, 11 RBMKs, and 1 LMR) are in Russia, where, under current plans, all of the dismantlement of former Soviet nuclear weapons will take place, and therefore all of the excess plutonium from weapons will arise. Twenty-one additional nuclear plants are under construction in the former Soviet Union, but at this time construction work has been suspended on all but five.

Civilian Plutonium Separation and Use

Nearly all of the reactors just described use low-enriched or natural uranium as their fuel. Nevertheless, as noted earlier, several countries are reprocessing plutonium from spent fuel for use as fresh fuel in nuclear reactors. In most cases, the original plan was to use this plutonium as fuel for liquid-metal "breeder" reactors. But delays in the commercialization of breeder reactors have led a number of countries to pursue major programs to use plutonium as mixed-oxide (MOX) fuel in LWRs, to limit the buildup of excess stocks of separated plutonium.³³ As just noted, despite these recycling programs, some 86 tons of separated plutonium has built up, and reprocessing continues to outpace the use of the resulting plutonium.

³² These are the Rancho Seco (California), San Onofre (California), Yankee (Massachusetts), and Trojan (Oregon) pressurized-water reactors, the Shoreham (New York) boiling-water reactor, and the Fort St. Vrain (Colorado) gas-cooled reactor.

³³ For a discussion of the major civilian plutonium programs in the world, see NAS (1994, Appendix B).

TABLE 2-3 Nuclear-Power Reactors of the World

Country	Number of Reactors					Total	Capacity (GWe)
	LWR	LGR	HWR	LMR	GCR		
United States	109/7	--	--	--	--	109/7	99.4/8.5
France	53/5	--	--	1/1	1/0	55/6	56.5/8.3
Japan	44/7	--	1/0	0/1	1/0	46/8	35.9/7.8
Germany	21/0	--	--	--	--	21/0	22.6/-
Russia	13/11	11/1	--	1/3	--	25/15	19.8/13.6
Canada	--	--	22/0	--	--	22/0	15.4/-
Ukraine	12/6	2/0	--	--	--	14/6	12.1/5.7
Great Britain	0/1	--	--	1/0	34/0	35/1	11.7/1.2
Sweden	12/0	--	--	--	--	12/0	10.0/-
Korea	8/4	--	1/3	--	--	9/7	7.2/5.9
Spain	9/6	--	--	--	--	9/6	7.1/5.7
Belgium	7/0	--	--	--	--	7/0	5.5/-
All Others ^a	42/18	2/-	10/13	1/-	--	55/31	28.8/17.5
Totals	330/65	15/1	34/16	4/5	36/0	419/87	332/74

NOTES: Figures before the slash refer to operating reactors, figures after the slash to partially completed ones. Countries are arranged in descending order of net capacity in operating reactors.

ABBREVIATIONS:

LWR = light-water reactor.

LGR = light-water-cooled, graphite-moderated reactor.

HWR = heavy-water reactor.

LMR = liquid-metal reactor.

GCR = gas-cooled reactor.

GWe = gigawatt-electric.

^a Of the countries not listed separately, none has as much as 5 GWe of operating nuclear capacity.

SOURCE: *Nuclear News* 1993.

The present world civilian reprocessing capacity (counting the recently opened British Thermal Oxide Reprocessing Plant [THORP]) amounts to roughly 5,600 MTHM/yr (IAEA 1992). The major centers of civilian reprocessing are in France, Britain, and Russia, though Japan has a small facility and is building a large one, and several other countries have small-scale capabilities. All of these plants use aqueous homogeneous processing (PUREX; plutonium and uranium recovery by extraction). If planned facilities in Japan and Russia are completed, roughly an additional 2,000 MTHM/yr will be added by the year 2005. During the 1990s, these facilities are expected to produce some 20 tons of separated plutonium each year.

MOX fabrication capacity worldwide currently amounts to roughly 75 MTHM/yr (Berkhout et al. 1993). Substantial additional capacity is under construction. Since most of this fuel is being used in LWRs at loadings of the order of 5 percent, rather than in LMR fuel with typical loadings of roughly 20 percent, the existing MOX fabrication capacity is substantially less than required to handle the 20 tons of civilian separated plutonium likely to be produced by reprocessing each year over the next decade. Hence it now appears inevitable that the substantial current excess stocks of civilian plutonium will continue to increase. The existing and planned MOX fabrication facilities would not be able to fabricate WPu into fuel unless their capacity were substantially expanded or civilian plutonium were displaced. Neither the United States nor Russia currently have operating MOX fabrication capacities on any significant scale, though both have incomplete facilities that could be completed in order to facilitate the disposition of excess WPu.

In short, the existing world reactor capacity is more than sufficient to handle the projected quantities of excess WPu (as well as the civilian plutonium being separated), but the available plutonium fuel fabrication capability is insufficient (taking into account continuing civilian reprocessing). Thus if disposition options involving the use of WPu as reactor fuel are chosen, provision of additional plutonium fuel fabrication capability may be required.

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CRITERIA FOR COMPARING DISPOSITION OPTIONS

The primary motivation of the U.S. government in its search for the most suitable means of disposition of surplus weapons plutonium (WPu)—and thus the primary motivation driving the current study—is to minimize the security risks posed by the existence of this material. Our parent committee concluded in its 1994 report that the tens of thousands of plutonium pits that will emerge over the next decade from the dismantlement of surplus nuclear weapons in the United States and Russia must be regarded as a “clear and present danger to national and international security,” and we agree.

Accordingly, our discussion of relevant criteria for comparing the options for WPu disposition begins with the security risks: their nature, the disposition-option characteristics that influence them, and the formulation of figures of merit to quantify or otherwise illuminate those influences. The issues of timing and capacity—how quickly an option can be put into operation and how rapidly it can process WPu thereafter—will be seen to be tightly intertwined with other aspects of security, and we treat these matters together here. We then turn to criteria related to economics; environment, safety, and health; and other considerations.

TABLE 3-1 Threats Associated with Surplus Weapons Plutonium

Types of Threats

1. *Diversion* of the WPu by the original possessor nation for reincorporation into nuclear weapons (“breakout”), which may be
 - a. overt, or
 - b. covert.
2. *Theft* of the WPu by or for other countries or subnational groups, with or without the complicity of insiders in the custodial organization, by means that are
 - a. forcible, or
 - b. overt but not forcible (as could occur under loss of national authority),
or
 - c. covert.
3. *Harmful influences* of the management of WPu on
 - a. the strength and stability of institutions for nuclear weapons management and monitoring in the United States and the former Soviet Union;
 - b. incentives and disincentives for further nuclear arms reductions in the United States, the former Soviet Union, and other nuclear-weapon states;
 - c. incentives and disincentives for acquisition of nuclear weapons by other countries; and
 - d. management of reactor plutonium in ways that increase its accessibility to prospective bomb-makers.

Time Frames in Which the Threats May be Operative

- the *near term*, roughly the next 10 years, within which the quantities of WPu accumulated from dismantlement activities are increasing and most disposition options would be in their developmental or initial operational stages;
 - the *middle term*, roughly from 10–50 years hence, within which most disposition options would be in full operation and at the end of which the bulk of the surplus WPu would have been processed; and
 - the *long term*, beyond 50 years hence, wherein the surplus WPu would be in whatever final form and location had resulted from the disposition option selected.
-

CRITERIA RELATED TO SECURITY AND TIMING

The Context for Security Concerns and Criteria

In view of the linkages, noted earlier, between U.S. and Russian choices for disposition of excess WPu, we have been attentive in this report to circumstances in both countries. We have also been attentive to the implications, for choices about management of surplus WPu, of the existence of civilian stocks of both separated and unseparated plutonium, which pose security risks in widely varying degrees and which altogether contain considerably more plutonium than the military stocks.

It is necessary to ask, more specifically, what standards of physical protection and monitoring are appropriate for *all* of the various forms of plutonium that occur in both the nuclear-weapons and the nuclear-energy sectors. Related questions include:

- (a) Is it worthwhile to invest significant resources—or to tolerate significant additional delays, risks, and uncertainties—to transform the small stock of surplus WPu into a form that is substantially more difficult to recover for use in weapons than the larger and growing stock of plutonium in spent fuel?
- (b) Should the existing levels of security and monitoring for separated and unseparated civilian plutonium be upgraded, regardless of what is decided about the disposition of WPu?
- (c) Could the options that might become available for eliminating surplus WPu, or otherwise making it less accessible for use in weapons than is plutonium in civilian spent fuel, be expanded (with tolerable cost, uncertainties, and timing) to do the same with the global stock of plutonium in spent fuel if that were deemed desirable?
- (d) How might choices about the disposition of WPu influence decisions about the management of reactor plutonium in ways that affect—for better or for worse—the danger that the latter might be used in weapons?

Questions (a) and (b) relate to the strategy for managing the security risks from plutonium of all kinds, and we return to them shortly. Issues (c) and (d) relate to the properties and implications of particular candidate options for the disposition of WPu, and we address them in the subsequent sections devoted to those options.

Specific Security Concerns and Threat Characteristics

It is useful, for purposes of developing criteria relating to security, to subdivide the security threats associated with surplus WPu using the framework presented in Table 3-1, which distinguishes among threats of diversion (by the

original possessor nation), theft (by other countries or subnational groups), and harmful influences (with respect to nuclear weapons management, arms control, and nonproliferation). Options for the disposition of WPu should then be judged with respect to:

- i. how the options would affect the difficulty, duration, cost, and detectability of attempts—within the categories of diversion and theft listed under items (1) and (2) in Table 3-1—to acquire WPu and carry out the processing and fabrication steps necessary to fashion it into functional nuclear bombs, and
- ii. on how the options might influence the weapon management, arms control, and nonproliferation institutions, incentives, and outcomes indicated under item (3) in Table 3-1.

Ideally, these evaluations should take into account the interaction of the time dimensions of different disposition options with the possible changes over time in the relative importance of different threats.

Candidate disposition options typically consist of several steps, beginning with intact nuclear weapons, proceeding through some number of intermediate processing, storage, and transport steps, and ending with either the physical destruction of the plutonium (by fission or transmutation) or its disposal in a form and location where it is intended to remain until its disappearance via radioactive decay.¹ Evaluation of the security benefits and liabilities of any such option, with respect to the threats described above, requires assessing the security risks, with respect to each type of threat, of each step the option entails. Such an assessment must take into account the barriers to acquisition and weapons use of the material associated with the form of the material at each step, the additional barriers to acquisition associated with the way the step is implemented, the quantities of material at risk at each step and the time interval during which it is at risk, and the interaction of these risk factors with the characteristics of the threat. These factors are summarized in Table 3-2.

A Matrix Scheme for Characterizing Options

The foregoing considerations suggest a matrix approach to characterization of the security implications of different options for the disposition of WPu, in which the rows of the matrix are the steps in an option and the columns portray,

¹ It should be noted that the dominant plutonium isotope, 24,000-year half-life plutonium-239, decays into 700,000,000-year half-life uranium-235, which is also a nuclear-explosive material.

TABLE 3-2 Factors Governing Security Risks of Disposition Steps

-
1. *Relevant quantitative characteristics: duration, integrated inventory, dilution*
 - a. start dates and end dates for the step in question;
 - b. integrated inventory, i.e., average inventory in step times duration in years (kilograms-years); and
 - c. dilution of plutonium in accompanying material (kilogram of material per kilogram of plutonium).
 2. *Barriers to the acquisition and use of the material to make nuclear explosives, specifically*
 - a. *barriers intrinsic to the form of the material, including*
 - i. *isotopic barriers*, meaning the relative difficulty of making nuclear explosives with material of this isotopic composition, or the difficulty of suitably altering its isotopic composition;
 - ii. *chemical barriers*, meaning the extent and difficulty of chemical processing required to separate the weapons usable substance(s) from accompanying dilutants and contaminants;
 - iii. *radiologic barriers* associated with the radiation fields and internal dose potentials of the weapon-usable substance(s) and accompanying materials; and
 - iv. *barriers of mass and bulk*, relating to the difficulty of moving the material in the course of theft or diversion, and the difficulty of concealing such activity.
 - b. *barriers dependent on the details of the option's implementation, including*
 - i. *locational barriers*, such as site isolation, difficult terrain, burial depth, difficulty of excavation and tunneling;
 - ii. *containment barriers*, such as massive containers, vaults, buildings, fences, detectors, alarms; and
 - iii. *institutional barriers*, such as proximity, capability, and reliability of guard forces, and intensity and reliability of monitoring.
 3. *The characteristics of the threat, including*
 - a. its type in the categorization scheme of Table 3-1;
 - b. complicity of custodial organization or individuals within it;
 - c. capabilities of attacking forces (numbers, weapons, training, organization, determination) in the case of forcible theft; and
 - d. knowledge, skills, money, and technology available to the prospective bomb-makers.
-

TABLE 3-3 Format for Characterizing Security Risks of a Disposition Option

Step	Time and Quantity		Qualitative Evaluations (scale 0 to 4) of				Qualitative evaluation of vulnerability (low, medium, high) based on threat-barrier interactions with respect to threats of						
	Start Date	End Date (kg-yr)	Integrated Inventory (kg-yr)	Dilution (kg of material per kg of Pu)	Intrinsic Barriers ^a	Implementation-Dependent Barriers	Overt Diversion	Covert Diversion	Forcible Theft	Covert Theft			
					Isotopic	Chemical	Logic	Mass/Bulk	Location	Containment	Institutional		
A													
B													
C													
D													
E													
F													
G													
H													
I													
J													

^a Intrinsic barriers refer to form of plutonium at end of step.

Example: once-through mixed-oxide (MOX)/spent fuel option in light-water reactor (LWR), followed by eventual emplacement of spent fuel in geologic repository.

- Step A: Storage as pits.
- B: Conversion to oxide.
- C: Transport as oxide.
- D: Fabrication to MOX.
- E: Transport as MOX.
- F: Storage as MOX.
- G: Burnup in LWR.
- H: Spent fuel storage.
- I: Spent fuel transport.
- J: Repository disposal.

for each step, the quantitative and qualitative risk factors and the vulnerability to different types of threat in the light of those factors. The format for such a matrix is illustrated in Table 3-3 for a disposition option centered around converting the WPu into spent light-water reactor fuel. In Chapter 6 we fill in a few such matrices for the plutonium management options of greatest interest. In the following paragraphs we elaborate on the justifications for the entries in this scheme.

Start Dates

The date at which operations in a particular step would be expected to commence determines the beginning of the period of potential vulnerability for the step and is relevant for consideration of the interaction of (potentially) time-varying threats with the “opportunities” presented by the disposition option. The start date of the first disposition step beyond the storage of pits is a particularly informative indicator of security risk, since it reflects the duration of a phase of plutonium management that is problematic both from the standpoint of the attractiveness, for weapons purposes, of the stored material and from the standpoint that delay in moving beyond pit storage may call into question the commitment of the possessor states to actually demilitarizing this material, with potentially harmful influences on the prospects for further nuclear arms reductions and for nonproliferation. This first start date beyond pit storage depends on the state of scientific and technical readiness of the disposition option and the research and development time needed to remedy any defects in these respects; on the time needed to construct all of the relevant facilities; on the time needed to accomplish any necessary licensing steps; and on the time needed to gain acceptance for the option by the relevant publics and decision-makers (which might include, for example, local and state as well as federal officials, electric utility managements, or foreign governments).

End Dates

Clearly, the overall security risk associated with a disposition option depends, among other factors, on the lengths of time that the WPu spends in its most weapon-usable forms and most vulnerable locations and processes within the option—hence the relevance of durations of the steps within an option, as determined by the combination of start date and end date. These durations depend on the quantity of plutonium to be processed altogether and on the rate at which it can be processed. We base our quantitative estimates in this study on a nominal quantity of 50 tons of WPu, roughly the amount expected to become surplus in the United States by the year 2005. Processing rates are based on scales of operation we judge plausible in light of the capacities of existing relevant facilities (if any) and the trade-offs among cost, timing, and other aspects

of security. (This question of scale is addressed in detail in subsequent chapters dealing with individual options and in Chapter 6, dealing with comparisons.)

Integrated Inventory

Somewhat more informative than duration alone as an indicator of the security-risk “exposure” associated with a given step in plutonium disposition is the *integrated inventory* for that step, defined as the integral under the curve of quantity of WPU in the step versus time. As explained in Appendix A at the end of this chapter, this indicator can be calculated for processing and transport steps as well as for storage steps, given sufficient information about how these operations will be conducted. For the purposes of the preliminary comparisons undertaken in Chapter 6, which necessarily are based on quite sketchy and tentative characterizations of the options, we calculate integrated inventories only for two phases of the disposition process—the phase in which the WPU exists in the form of pits and the phase between conversion of the pits to oxides and the loading of this material into a reactor or melter (marking a great decrease in vulnerability). These two integrated inventory figures are, in our view, reasonably informative indicators² of the timing aspect of security, and they are calculable with a minimum of assumptions about the details of the disposition options.

Dilution

An additional quantitative figure provided for each step is the degree of dilution of the plutonium in whatever matrix contains it, measured in kilogram of total material per kilogram of plutonium. This dilution figure can be used to determine how much material must be moved and processed to acquire a weapon’s worth of plutonium.

Intrinsic Barriers

Our qualitative evaluations of barriers will employ a scale in which 0 means “negligible,” 1 means “small,” 2 means “medium,” 3 means “large,” and 4 means “very large.” This is not intended to be a linear scale but rather to denote qualitatively significant differences in the barriers to weapons use of the material. In cases where the differences do happen to be more or less quantifiable (as

² We note that integrated inventory, like other indices, is an imperfect measure of security hazard, better in relation to some categories of threats than in relation to others. For example, the risk of forcible theft of a few bombs’ worth of material from a particular facility probably will not depend very much on whether there is 1 ton of plutonium there or 50 tons; but the risks of covert diversion or theft—and of overt diversion—do increase, in many circumstances, with quantity as well as with duration.

with, e.g., the radiologic barrier), the difference between adjacent levels is an order of magnitude (factor of 10) or more. The meaning of the levels in relation to the different barriers is elaborated further in the following subparagraphs.

Isotopic Composition. Relevant aspects include fractions of plutonium-240 (Pu-240) and Pu-242, which affect critical mass and difficulty of design; fraction of Pu-238, which complicates design through its high heat generation; and fraction of Pu-241, which governs buildup of heat-generating and radiologically hazardous americium-241. We take high-enriched uranium (HEU) as the 0 (“negligible”) reference point for isotopic barriers, and characterize weapons-grade plutonium (approximately 90 percent Pu-239) as 1, typical reactor-grade plutonium (approximately 60 percent Pu-239) as 2, and very-high-burnup plutonium (40 percent Pu-239 or less) as 3. Uranium with uranium-235 (U-235) or U-233 content less than 20 percent would qualify as 4, but no plutonium composition relevant to this study would so qualify.

Chemical Form. Relevant aspects include whether the plutonium is in metallic form (the most convenient for immediate use in a weapon, but not necessarily the most convenient for further processing or for storage if such steps will be part of a weapons effort) or oxide, carbide, nitrate, etc., and admixture of impurities (such as other metals, oxides, or carbides, or fission products, or other neutron absorbers, which, variously, affect chemical processing requirements and radiological hazard to bomb-makers). We take pure plutonium metal to be the 0 reference point for chemical barriers, and characterize pure plutonium oxides as 1, mixed uranium and plutonium oxides (MOX)—including MOX mixed with additional dilutents or neutron absorbers other than fission products—as 2, and plutonium embedded in spent fuel or vitrified radioactive waste as 4. (Here the jump from level 2 to 4 is not because the chemistry *per se* is very much harder, but because the penetrating radiation from the fission products necessitates that the chemistry be done using degrees of shielding and remote handling that greatly increase the difficulty.)

Radiologic Hazard. This barrier depends both on the gamma-emitting properties of the material, which govern the radiation field and shielding requirements associated with approaching and handling the material, and the beta- and alpha-emitting properties governing the hazards that occur if as a result of processing or dispersal the material is inhaled or ingested. We take natural, low-enriched, or depleted uranium to be the 0 reference point for radiologic hazard, and characterize HEU as 1, WPu in metal or plutonium oxide as 2, reactor plutonium in metal or plutonium oxide as 3, and plutonium in spent fuel or mixed with high-level radioactive wastes as 4.³ The twenty-fold or more dilution from

³ It should be noted that the gap between level 3 and 4 is very large in this case, amounting to a qualitative difference. While both WPu and reactor plutonium can be handled in small glove-box facilities, the penetrating gamma radiation from most commercial spent fuel or from large vitrified high-level waste logs being produced or planned for production in several countries is so intense as

plutonium metal or oxide to MOX reduces the radiologic hazard one level: WPu in MOX is characterized as 1, and reactor plutonium in MOX as 2.

Mass/Bulk. This barrier relates to whether the form of the material permits ready partitioning in a way that facilitates concealment on the person of a thief or divertor (as would be the case with small quantities of HEU or plutonium metal or oxide, taken as level 0), or is readily portable by one person although somewhat more difficult to conceal (as in the case of a pit, level 1), or can be moved by one person but only with some difficulty and little chance of concealment (fuel assemblies weighing tens of kilograms, level 2), or requires a forklift (approximately 100 kg or more, level 3) or a crane (approximately 1,000 kg or more, level 4).

Implementation-Dependent Barriers

For these barriers, too, it is convenient to employ a scale of qualitative distinctions ranging from 0 for negligible barriers to 4 for very large ones.

Location/Exposure. We associate level 0 with transport, 1 with processing at multiple sites, 2 with processing at a single site or storage at multiple sites, and 3 with storage at a single site, and we add 1 if the sites are remote or otherwise difficult of access.

Containment. We associate level 0 with storage containers that can be opened by hand or ordinary tools and are not equipped with seals; and with material in unsealed processing equipment that can be similarly opened. Level 1 is for containers and processing equipment with seals that render any tampering detectable after the fact. Level 2 is for containers and processing equipment requiring more substantial tools (such as industrial cutting equipment) to open, and also equipped with seals. The characteristics of the next level of containment then add 0, 1, or 2 to the rating; 0 is for containment that could be breached quickly and easily by a single individual (such as an ordinary industrial building behind a chain-link fence); 1 denotes a significant extra degree of difficulty, making it unlikely that an individual or small group could enter the facility and reach the material before guard forces could respond (as might be achieved by alarmed fences, intrusion-detection devices, special locks and reinforcement on doors and other penetrations, etc.); and level 2 denotes a significant further level of difficulty (such as imposed by a highly engineered vault or vault-like building, deep burial, etc.) requiring such quantities of people, equipment, and time to overcome that such an intrusion could not be accomplished covertly, even with the assistance of insiders.⁴

to require remotely operated facilities to handle materials in these forms. Such facilities represent a substantial increase in the sophistication required for successful processing.

⁴ It should be noted that the mere presence of fences, alarms, and vaults does not ensure an effective containment system; repeated vulnerability analysis and testing is required to determine whether there are weak points in the system that may have been overlooked.

Institutional Barriers. Here 0 would denote an absence of guards or other protective services beyond reliance on local police forces on the usual basis. Level 1 denotes typical industrial security in terms of private guard forces and monitoring of personnel entering and leaving. Level 2 denotes armed guard and monitoring capabilities that would be typical of a nuclear weapons laboratory such as Livermore, including the ability to counter intrusions that involve an insider at the facility. Level 3 denotes a capability (such as would characterize a nuclear weapons storage site) to successfully defend against organized attack by well-armed intruder groups—possibly including the participation of more than one insider—and corresponding inspection and monitoring capabilities. Level 4 means the same physical capabilities as 3, but with multinational or international participation that reduces the possibility of access to the material by the possessor country or by subnational groups under conditions of civil disorder.

The Threat and Vulnerability Interaction

Our characterization of vulnerability in the matrix format of Table 3-3 is presented in relation to four classes of threat: overt and covert diversion (categories 1.a and 1.b in Table 3-1), and forcible theft and covert threat (categories 2.a and 2.c). Harmful influences (all of category 3) do not lend themselves to characterization in the disaggregated step-by-step format of the matrix and so must be treated separately. Vulnerability at each step to each of the four threat categories considered are characterized simply as “low,” “medium,” or “high,” based on our judgment about the effectiveness of the relevant barriers against the indicated threats. Any more discriminating characterization than this probably would not be warranted in light of the uncertainties associated with threats and barriers alike. Of course, the *overall* security risk associated with any combination of disposition option and threats will be disproportionately influenced by the characteristics of the most vulnerable step or steps in that option, because the threats are all associated with human intervenors who can be expected to seek out the points of greatest vulnerability. Accordingly, our security-risk comparisons among disposition options—presented in Chapter 6—stress the identification and characterization, for each option, of the most vulnerable step or steps.

The foregoing matrix scheme for characterizing security hazards bears some relation to official classification schemes for nuclear materials subject to safeguards, such as those of the U.S. Department of Energy (USDOE 1993c), the U.S. Nuclear Regulatory Commission (NRC) (OFR 1992c), and the International Atomic Energy Agency (IAEA 1987). The DOE scheme, based on a combination of material quantities and “attractiveness levels” related to our “intrinsic barriers,” is shown in Table 3-4. The NRC classification is summarized in Table 3-5. The IAEA scheme is similar, defining “significant quantities” of different categories of material and characterizing “conversion times”

TABLE 3-4 Attractiveness Levels and Safeguards Categories From DOE Order 5633.3A

Type of Material	Attractiveness Level	Safeguards Category (I = greatest concern) Versus Kilograms of Contained:							
		Pu or U-233				U-235			
		I	II	III	IV	I	II	III	IV
Weapons ^a	A	Any quantity is Category I				Any quantity is Category I			
Pure products ^b	B	>2	0.4-2	0.2-0.4	<0.2	>5	1-5	0.4-1	<0.4
High-grade materials ^c	C	>6	2-6	0.4-2	<0.4	>20	6-20	2-6	<2
Low-grade materials ^d	D	NA	>16	3-16	<3	NA	>50	8-50	<8
All other materials ^e	E	Any reportable quantity ^f is Category IV				Any reportable quantity ^f is Category IV			

NOTE: NA indicates not applicable.

^a Assembled weapons and test devices.

^b Pits, major components, buttons, ingots, recastable metal, directly convertible materials.

^c Carbides, oxides, solutions of >25 gram per liter (g/l), nitrates, etc., fuel elements and assemblies, alloys and mixtures, UF₄ or UF₆ at ≥ 50 percent enrichment.

^d Solutions of 1-25 g/l, process residues requiring extensive reprocessing, moderately irradiated material, Pu-238 (except in waste), UF₄ or UF₆ at 20-50 percent enrichment.

^e Highly irradiated forms, solutions of <1 g/l, uranium in any form and quantity containing greater than 20 percent U-235.

^f Is 1 g of Pu-239 to Pu-242 and enriched uranium, 0.1 g of Pu-238.

SOURCE: USDOE 1993c.

TABLE 3-5 Nuclear Regulatory Commission Definitions Characterizing Special Nuclear Material (bracketed numbers refer to paragraphs in the source document [OFR 1992c])

Strategic special nuclear material (SSNM) means plutonium, U-233, or U-235 contained in uranium enriched to 20 percent or more in U-235 [73.2]

Formula quantity means SSNM in any combination that adds up to 5,000 g or more by the formula, grams U-235 + 2.5 (grams plutonium + grams U-233) [73.2].

Category IA material means SSNM directly usable in a nuclear-explosive device, except if

- the dimensions are large enough to preclude hiding the item on an individual (defined as ≥ 2 m in one dimension, or > 1 m in each of two dimensions, or > 25 cm in each of three dimensions), or
- the total weight of 5 formula kg of SSNM plus its matrix cannot be carried inconspicuously by one person (defined as ≥ 50 kg), or
- the quantity of SSNM in each container requires protracted diversions in order to accumulate 5 formula kg (defined as < 0.05 formula kg per container) [74.4].

Special nuclear material of moderate strategic significance means less than a formula quantity of SSNM but

- > 1,000 g U-235 contained in uranium enriched to 20 percent or more in U-235, or
- > 500 g of U-233 or plutonium, or
- > 1,000 g in combination calculated as grams contained U-235 + 2.5 (g U-233 + g Pu), or
- $\geq 10,000$ g U-235 contained in uranium enriched to ≥ 10 percent but < 20 percent U-235 [73.2].

Special nuclear material of low strategic significance means less than an amount of special nuclear material of moderate strategic significance but

- > 15 g U-235 contained in uranium enriched to 20 percent or more in U-235, or
- > 15 g of U-233 or plutonium, or
- > 15 g in combination calculated as grams contained U-235 + g U-233 + g Pu, or
- > 1,000 g U-235 contained in uranium enriched to ≥ 10 percent but < 20 percent U-235, or
- $\geq 10,000$ g U-235 contained in uranium enriched above natural but < 10 percent U-235 [73.2].

Special nuclear material that is not readily separable from other radioactive material and that has a total external radiation dose rate in excess of 100 rem/hr at a distance of 3 feet from any accessible surface without intervening shielding is exempt from certain safeguards requirements [73.6].

estimated to be needed for a divertor to transform the material into a “finished Pu metal component;” the IAEA also presents a relation for estimating “expected accountancy capability,” meaning the minimum loss of nuclear material that will be detectable with a given probability and given risk of false

alarm, as a function of the material stock or throughput and the measurement accuracy expected for IAEA safeguards applied to the activity in question (e.g., uranium enrichment, MOX fabrication). We have drawn on elements of all of these approaches in developing our own characterization scheme. Our scheme is more disaggregated and more context-dependent than these other approaches, as we believe is necessary in order to illuminate—as fully as present information and judgment allow—the security characteristics of alternative options for WPU disposition.

Criteria for Choice

Characterization of the security risks of the various disposition options in the matrix format just described will provide insight into the loci of greatest risk within each option, as well as a basis for judgmental comparison of overall risk between options. (The comparison involves judgment in reaching a conclusion from comparison of multiattribute and partly qualitative characterizations; we can find no defensible way to compute a single quantitative index of overall risk for each option, which would require agreeing on numerical values and relative weights for each relevant characteristic.)

A useful assessment should not only provide a basis for relative comparisons among options, however, but also should give some guidance with respect to some absolute standard of adequacy. It is important to ask, in other words, “How good is good enough?” and, having formulated such standards, to ascertain which options meet them. The 1994 study by CISAC put forward two standards for use in this connection, the “stored weapons standard” and the “spent fuel standard.”

The “stored weapons standard” holds that “to the extent possible, the high standards of security and accounting applied to storage of intact nuclear weapons should be maintained for these [nuclear explosive] materials throughout [the] processes [of dismantlement, storage, and disposition]” (NAS 1994, p. 31). We take this standard to be appropriate for the management of WPU up to the point in the disposition process where the spent fuel standard has been attained.⁵ The argument for the stored weapons standard is that the pathway leading from any separated plutonium form (a pit, an ingot, plutonium oxide, or even plutonium and uranium mixed oxide) is sufficiently direct and easily traversed by at least some potential proliferators that applying *less* than the stored weapons standard to the protection of such material could lead to the highly undesirable result that dismantlement of surplus nuclear weapons and disposition of their nuclear-explosive materials could produce an *increase* in proliferation risk.

⁵ We do not think that civilian spent fuel, or WPU embedded in spent fuel or any form that meets the spent fuel standard, needs to be guarded with the same degree of meticulousness accorded to intact nuclear weapons.

The “spent fuel standard” means that the plutonium has become “roughly as inaccessible for weapons use as the much larger and growing stock of plutonium in civilian spent fuel” (NAS 1994, p. 34). We note that this does *not* imply a specific combination of radiation barrier, isotopic mixture, and degree of dilution of plutonium. Rather, it describes a condition in which WPu has become roughly as difficult to acquire, process, and use in nuclear weapons as it would be to use plutonium in commercial spent fuel for this purpose. The rationale for the spent fuel standard is, first, that the bulk, composition, and ionizing-radiation field of spent fuel pose very appreciable barriers to the theft or diversion of this material and extraction of the contained plutonium for use in nuclear weapons and, second, that the existence in the world of many hundreds of tons of civilian plutonium in spent fuel means that there would be little security gain from special efforts to completely eliminate the WPu, or to render it much less accessible even than the plutonium in spent fuel, unless society were prepared to take the same approach with the global stock of civilian plutonium.⁶ Addressing this global issue, on the other hand, would entail time periods (and costs) far greater than needed or appropriate to address the “clear and present danger” from the existing WPu.

Additional questions do suggest themselves, as indicated earlier in this section: To what standard of protection for WPu and civilian plutonium *combined* should society aspire in the longer term? How well does aging spent fuel need to be protected? Should separation of plutonium from fission products be avoided altogether? Should the inventory of plutonium be minimized, insofar as possible, by fissioning it in ways that do not produce more? Or should commerce in plutonium for energy production be allowed to become commonplace, accompanied by a vigorous effort to develop and implement safeguards adequate to the challenge this would represent? We do not think these questions can be answered except in the context of an evolving understanding of the role society expects nuclear fission to play in its energy future, and of the technological and institutional options through which fission might do so with reduced risks to both safety and security, compared to the fission technologies of today. These are crucial issues, but they go far beyond the mandate of this panel.

⁶ Concerning the spent fuel standard, we are aware that the accessibility of plutonium in commercial spent fuel is quite variable and increases with time as the fission-product radioactivity that provides the principle barrier to processing of the material for weapons use decays. An appropriate interpretation of what sort of spent fuel constitutes the standard follows from consideration of the situation that will exist at the time in the future when most of the surplus WPu at issue here is being processed for final disposition, say, between 2000 and 2030. There is likely to exist, in that period, upwards of 1,000 tons of civilian plutonium in spent fuel, ranging in age from freshly discharged to several decades old. If the inaccessibility of WPu is made comparable to that of civilian plutonium in the middle of this age distribution—that is, civilian plutonium in spent fuel 20–30 years old—the existence of the WPu in this form would not markedly increase the security risks already associated with the civilian spent fuel.

ISSUES AND CRITERIA IN ECONOMIC EVALUATION OF ALTERNATIVES

The monetary costs (or benefits) of alternative approaches to the disposition of WPu are of secondary importance compared to the security aspects: the security risks associated with this material are so great that it is difficult to imagine choosing the riskier of two candidate approaches because it saves money, all the more so because the total sums involved are unlikely to be nearly as large as those that the United States and the former Soviet Union routinely invested in the past in attempts to buy security against nuclear-weapon dangers. It is nonetheless worthwhile to illuminate the economic dimension of alternative disposition approaches as clearly as possible, both to assist in ranking schemes that are not readily distinguishable on security grounds and to facilitate planning for the investments that will be required for the option that is chosen.

Principles and Pitfalls in Cost Comparisons

The monetary costs of any disposition option arise in connection with pre-operational, operational, and decommissioning phases: preoperational activities include research, development, demonstration, licensing, and construction of new facilities or modifications to existing ones; operational activities consist of the actual processing and handling of the surplus WPu; and decommissioning consists of dismantling the facilities, disposing of their components, and managing the sites (restoring them to other uses, or quarantining them) after the operational phase is over. For options that entail the generation of electricity in the course of plutonium disposition, revenues as well as costs will be associated with the operational phase.

Comparison of the economics of alternative options requires that the costs and revenues be estimated on a consistent basis. Such calculations can be quite complex, and the range of conventions and assumptions routinely used in carrying them out is wide. Relevant factors include:

- (a) the treatment of inflation;
- (b) whether the activities are carried out by government or civilian entities, or a combination, and corresponding assumptions about
 - the real cost of money and rates of return appropriate for the entities operating the option, and
 - property taxes and insurance costs associated with the facilities and operations involved;
- (c) conventions and assumptions relating to the components of the capital investments associated with the activities, including
 - the costs of land, materials, labor, and purchased components in the region where the option will be implemented;

- the conventions by which indirect as well as direct costs are included in the calculation;
 - the size of the “contingency” factor allowing for anticipated but a priori unspecifiable sources of growth in construction costs beyond the baseline estimate; and
 - the inclusion or exclusion of interest on investments made before the operational phase commences (often termed “interest during construction,” although in principle the category is broader);
- (d) the degree of comprehensiveness in inclusion of all of the facilities and operations needed to perform the plutonium disposition mission;
- (e) the means by which subsidiary benefits of plutonium disposition operations (such as the generation of electricity) are taken into account in the economic calculations;
- (f) the treatment of “sunk” costs in relevant facilities and operations, that is, costs incurred prior to the current consideration of the possible use of such facilities and operations for disposition of WPu; and
- (g) the operational lifetime of the facilities (or, in some cases, the period over which the investment in them is to be written off).

Variations and inconsistencies in the treatment of these factors make it practically impossible to derive informative conclusions about costs of alternatives from direct comparison of final cost estimates obtained in different studies of the individual disposition options; rather, it is necessary to construct a consistently based set of estimates starting from the building blocks (such as estimates of direct construction costs, or of labor and materials requirements) that such studies provide.⁷ In the paragraphs that follow, we explain our own assumptions about the factors (a)-(g), identify some generic uncertainties, and call attention to differences between our treatment and those of others.

Inflation and the Real Cost of Money

The concepts of inflation and real cost of money are reviewed in the box on p. 76. We choose 1992 U.S. dollars as our reference currency, and, following the Office of Management and Budget guidance for cost-benefit analysis of federal programs (OMB 1992), we choose the gross domestic product (GDP) implicit price deflator (annual average) as the index of inflation. We assume the post-1992 annual rate of inflation, based on this index, will be 3.0 percent per

⁷ Studies that offer estimates of costs without providing sufficient detail about the derivation of these to permit such a procedure are not useful for purposes of making systematic comparisons.

THE CONCEPTS OF INFLATION AND REAL COST OF MONEY

Because of inflation, the purchasing power of a dollar in 1993 is less than that of a dollar in 1992, that of a dollar in 1992 much less than that of a dollar in 1972, and so on. As a result, the rather common practice of calculating the total costs of an activity over a span of time by adding up its yearly costs in then-current dollars—as in, e.g., calculating U.S. defense spending for the 1980s as the sum of current-dollar defense expenditures for 1980–1989—amounts to an aggregation of apples and oranges (more precisely, of bigger apples with smaller ones).

To avoid the distortion associated with this practice, expenditures (and income) from different years should be converted to the currency of a single reference year before being summed. Performing this conversion requires, in addition to the choice of a reference year, choice of an index of inflation from among such candidates as the consumer price index, the producer price index, and the implicit price deflator for gross domestic product (GDP); it may also require estimating future rates of inflation. (If inflation rates are high, it is appropriate to specify not just a reference year but a reference date for the currency, e.g., January 1, 1992 dollars.)

Although the inflation rate and the time-value of money are sometimes conflated in perception and analysis, they are distinct concepts. If the effects of inflation are removed by working with constant dollars of a specified reference year, it remains true that having a 1992 dollar today is preferable to having a 1992 dollar next year, and preferable by far to having a 1992 dollar 10 years hence. This

year.⁸ The corresponding relation of 1992 dollars to dollars of other years encountered in recent studies is shown in Table 3-6.

For federal government projects, values for the real cost of money in the range of $r = 4$ –5 percent per year often have been used in recent years, corresponding to nominal rates of return on long-term government bonds in the range of 7 to 8 percent per year and inflation rates, experienced and projected, in the

⁸ The average annual rate of inflation from 1972–1992, based on the GDP implicit price deflator, was 3.7 percent per year, the corresponding average for 1987–1992 was 3.5 percent per year, and that for 1990–1992 was 3.2 percent per year (US Dept of Commerce 1992); the January 1993 estimate of the Council of Economic Advisors for the period 1992–1995 was 2.6 percent per year (CEA 1993).

time-value of money is associated, if the money is yours, with the real (inflation-corrected) rate of return that the money could bring if invested; and, if the money must be borrowed from a bank, or raised by the sale of stocks or bonds, its time-value is associated with the real rate of return you must pay its owners in order to have the use of it. It is conventional to use the term "cost of money" for both of these closely related cases.

The real cost of money is equal to the nominal (current-dollar) cost of money minus the rate of inflation and is ordinarily expressed as an annual percentage rate. One must specify a figure for the real cost of money in order to compute the levelized annual capital charges associated with an initial investment that is to be retired over a specified number of years, or to compute the "present value" of a stream of future costs or revenues (see Appendix B at the end of this chapter).

The real cost of money depends on the entities (individuals, institutions) involved and the financial instruments (e.g., bank credit cards, home mortgages, electric utility financing, venture capital, government borrowing)—including whether the returns involved are subject to income tax—and it varies with time because of macroeconomic conditions that affect the relative scarcity or abundance of capital. It is customary, for purposes of financial evaluation of projects, to choose a single annual percentage rate that is thought to be appropriate to the entities, context, and time frame involved. Since the choice of this figure is not only dependent on a variety of factors and judgments but also crucial to the outcome of project evaluations (often determining whether a given project will appear to be a money-maker or money-loser), the question of what figure to choose can be controversial.

range of 2.5 to 3.5 percent per year. A figure of 4 percent per year was chosen by the Technical Review Committee for DOE's Plutonium Disposition Study (USDOE 1993a), and, as noted, is certainly defensible as a value of the real cost of money to the government based on prevailing government-bond yields and inflation rates.

The guidelines published by OMB for benefit-cost analysis of federal programs, however, call for the use of a real cost of money of 7 percent per year for projects that have effects in the private sector, i.e., outside the government (OMB 1992). OMB (p. 9) identifies this figure as "approximately the marginal pre-tax rate of return on investment in the private sector in recent years." The rationale for using this figure for the analysis of federal projects is that use of a

TABLE 3-6 Inflation: Relation of 1992 Dollars to Dollars of Other Years

1982	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994
0.695	0.783	0.803	0.829	0.862	0.900	0.939	0.977	1.000	1.030	1.061

SOURCE: The figures for 1982–1992 are derived from data for current-dollar and constant-dollar calendar-year (as opposed to fiscal-year) gross domestic product, as reported by the Council of Economic Advisors (CEA 1993); the index of inflation embodied in these figures is thus the implicit price deflator for GDP. The figures for 1993 and 1994 are based on our assumed inflation rate of 3.0 percent per year.

lower one can lead in practice to what is seen as unfair government competition with the private sector or to unwanted taxpayer subsidy of projects too risky or unrewarding to be undertaken privately.

In any case, the OMB states (p. 9) that the 7-percent per year real rate is to be used for all “public investments and regulatory programs that provide benefits and costs to the general public,” reserving lower figures for the analysis of cases in which a program’s only effect is to save the government money. It is clear, therefore, that the 7-percent figure would be the appropriate one, in the view of the OMB, to use for the analysis of projects for the disposition of WPu—this is after all a public benefit, whether or not electricity is generated in the process—and we use this figure in most of our analyses. (It should be noted that the real cost of money used in the evaluations of private firms depends on project risk, tax liability, and shares of financing from debt and equity, as well as other factors, and the resulting figures are often considerably higher than 7 percent per year.)

Some consequences of DOE’s decision to use a figure of only 4 percent per year in its Plutonium Disposition Study are discussed below.

Property Taxes and Insurance

Corporate and individual income taxes enter, where appropriate, into the determination of the real cost of money, while, by convention, property taxes and insurance are accounted for by the addition, to the fixed charge rate (FCR), of a simple annual percentage of the initial capital investment (see “The Concepts of Inflation and Real Cost of Money”). Government entities do not in general pay property taxes; private entities generally do. Government activities are usually “self-insured” (although liability is sometimes limited by law); private entities usually must pay for insurance. There is controversy about the existence, magnitude, and appropriateness of the “subsidies” for government activity associated with government’s freedom from property taxes and insurance costs, and about whether evaluations of the costs of government projects should be

adjusted to cover these factors. In this study, we perform economic evaluations of facility costs both with and without an adjustment, to the fixed charge rate, of a nominal 2-percent per year of initial capital investment for the sum of property taxes and insurance. Thus we have:

$$\text{FCR} = \frac{\text{CRF (capital recovery factor) without allowance for property taxes and insurance}}{\text{CRF} + 0.02/\text{yr with allowance for property taxes and insurance,}}$$

with CRF as given in Appendix B at the end of this chapter. The composition of the capital investment, I , to which the fixed charge rate is applied, is elaborated in "Composition of the Capital Investment" on p. 80.

Costs of Land, Materials, Labor, and Purchased Components

These "direct" costs of construction can vary significantly from region to region within the United States, and they can vary drastically between countries. Distortions in cost comparisons of different plutonium disposition options can arise from failure to account for such differences, as well as from inconsistencies in what has been included at all (e.g., whether land costs have been included).

In DOE's 1992–1993 Plutonium Disposition Study (PDS), no allowance was made for land costs, on the supposition that needed facilities would be constructed on existing DOE sites. Estimates of the costs of labor, materials, and purchased components are supposed to have been developed (in January 1, 1992 dollars) using standardized assumptions and models for a central-U.S. site as documented in the Nuclear Energy Cost Data Base maintained by the Oak Ridge National Laboratory (USDOE 1988). We have assumed, for purposes of our own economic evaluation, that these PDS direct-cost estimates were correctly and consistently obtained. In the several cases in which we consider plutonium disposition options not evaluated in the DOE study, we have assumed that the estimates of direct costs made available to us by others were derived by procedures roughly consistent with those used in the DOE work.

These assumptions are difficult to substantiate in the absence of a detailed vetting or reconstruction of the economic estimates in the various other studies, which would have required time and effort well beyond what was available to us for this study. They must be considered a source of significant uncertainty in our evaluation of the comparative economics of different options. This uncertainty is estimated on a case-by-case basis in Chapter 6; generally we take it to be in the range of ± 30 percent.

Relevant costs of labor, materials, and purchased components for plutonium disposition options would be much more difficult to estimate, of course, for the former Soviet Union. A recent study (Burns and Roe 1992) of fossil-fuel power-plant construction prospects in the former Soviet Union estimated labor costs at

COMPOSITION OF THE CAPITAL INVESTMENT

The initial capital investment, I , which the capital recovery factor (CRF) multiplies to give the levelized annual capital charges (LACC), can be subdivided in any number of ways. In the analysis of large construction projects, the following categorization is widely used:

$$I = OC + IDC,$$

where OC is the "overnight" cost, meaning what it would cost to build the facility if this could be done "overnight," and IDC is "interest during construction." The overnight cost is given by

$$OC = DCC + ICC + OPC + CGY,$$

with

DCC = direct construction costs,

comprising the costs of construction labor, land, materials, and purchased components;

ICC = indirect construction costs,

comprising construction facilities, equipment, and support services, safety and environmental engineering, inspection and other quality-assurance activities, project administration, and the like;

25 percent of those for the United States and material costs at 50 percent of the U.S. figures. We have seen another estimate that the total costs of construction and operation of facilities relevant to plutonium disposition should be assumed to be half as large in the former Soviet Union as in the United States. We consider the many difficulties of meaningful comparisons between U.S. and former Soviet Union costs to be so great, however, as to make them not worth attempting at the present time, and we have confined our detailed economic estimates to contexts outside the former Soviet Union.

Indirect Construction Costs

In detailed cost-estimation work, indirect construction costs are broken down into detailed subcategories that are estimated individually. In comparisons at the concept-evaluation level, however, it is customary to estimate the indirect costs either as a flat percentage of total direct costs, typically between 25 and 40

OPC = any other preoperational costs chosen for inclusion in a particular study,

which could include research and development costs, costs of safety analysis and licensing work conducted by the government instead of by the contractor, and so on; and

CGY = contingency,

which is an allowance for sources of cost increases above the baseline capital-cost estimates, of kinds that inevitably arise but cannot be predicted in detail (and thus cannot be incorporated in the specific line-items in the direct- and indirect-cost budget estimates).

Interest during construction, IDC, is calculated by escalating the constant-dollar value of investments made n years before the start of operation by a factor of $(1 + r)^n$ (with r the real cost of money), applying a factor of $(1 + r)^{n-1}$ to the part of the investment made $n + 1$ years before the start of operation, and so on. When this quantity is being estimated in advance, it is customary to assume that the trajectory of cumulative investment versus time during the construction period will be an S-curve, which is to say that the distribution of annual investments will be symmetric around the midpoint of the construction period with the largest ones in the middle years.

Issues and problems that arise in the estimation of these capital-cost components in the case of plutonium disposition options are elaborated in the main text.

percent, or to divide the direct costs into two or three categories according to the expected amounts of "indirect" activities expected to be associated with them, applying a different indirect-cost percentage to each.

In the case of the contractor studies of advanced reactor options for DOE's PDS, the Department's Technical Review Committee (TRC) found that the treatment of indirect costs among the studies was not uniform (USDOE 1993a, p. SC6-5). It also appears, from our review of the contractor studies and the TRC report, that there were nonuniformities in the manner in which various preoperational costs were allocated between the "indirect-costs" category and a separate "preoperational-cost" category. It has been difficult to determine, from the often less-than-transparent contractor and TRC reports, how extensive and drastic these nonuniformities were, or to what extent the TRC was able to correct them in the process of arriving at its own adjusted cost estimates. It is, similarly, often difficult to tell what conventions concerning indirect costs were used in obtaining cost estimates found elsewhere in the literature for options not

considered in the PDS. These ambiguities represent a source of uncertainty of order ± 20 percent in our comparison of the DOE cost estimates with each other and with estimates for other options.

Contingency

The contingency is customarily calculated using a multiplier—typically 15–25 percent—on the sum of direct and indirect construction costs. In an attempt to gain more accuracy in cost estimation, the costs of a project sometimes are disaggregated into higher-uncertainty and lower-uncertainty components, to which different contingency multipliers are applied. (Given that the contingency factor is at best a rough attempt to capture sources of cost escalation that cannot be predicted in detail, the value of the more complicated approaches is questionable.)

DOE provided detailed guidance on the calculation of contingency factors to its contractors in the PDS, but the TRC found that the contractors' treatment of the contingency was "uneven" and varied from the guidance (USDOE 1993a, p. SC6-3). Again, it has not been easy for us to determine, in the time available for this work, either the magnitude of these variations or the extent to which the TRC was able to correct them in making its own adjusted cost estimates. It is similarly not always easy to tell whether cost estimates we have used for other options, from elsewhere in the literature, do or do not include a contingency factor. These ambiguities represent an additional source of uncertainty of order ± 20 percent in our comparison of the DOE cost estimates with each other and with estimates for other options.

Interest on Preoperational Investments

Since construction of major facilities inevitably extends over a span of years, the calculation of the effective capital investment as of the start of operation must account for interest on money spent for construction (and other preoperational investments) in the years prior to the start date. It is customary to refer to this correction as "interest during construction," but it should be applied as well to other preoperational costs (such as research and development and licensing work) if these are not included in the indirect construction costs where they would automatically be multiplied by the interest-during-construction factor.

The magnitude of the multiplicative correction factor depends on the duration of the preoperational period, the phasing of the investments during this period, and the real cost of money. Magnitudes are shown in Table 3-7 for construction periods of 3, 6, and 9 years, typical "S-curve" investment trajectories, and real costs of money of 4, 7, and 10 percent per year. (Some analysts tally up interest-during-construction costs in variable dollars using nominal rather than

TABLE 3-7 Constant-Dollar Multiplicative Correction Factors for Interest During Construction

Real Cost of Money (% per year)	Construction Period (in years)		
	3	6	9
4	1.08	1.15	1.22
7	1.15	1.27	1.41
10	1.22	1.41	1.64

NOTES: Multiplying the indicated factors by "overnight" costs (sum of direct plus indirect plus contingency) gives total investment at the start of operation, for use in calculating capital charges during operation.

Calculation proceeds as if each year's construction money is borrowed at the beginning of that year. Distributions of construction costs among years are: 0.25, 0.50, 0.25 for the 3-year case; 0.05, 0.15, 0.30, 0.30, 0.15, 0.05 for the 6-year case; and 0.03, 0.07, 0.125, 0.17, 0.21, 0.17, 0.125, 0.07, 0.03 for the 9-year case.

real costs of money, which leads to impressively large—but essentially meaningless—figures.)

Interest during construction does not appear to have been consistently included in the contractor studies for DOE's PDS, and the report of DOE's TRC for the PDS study presents some of its findings in the form of constant-dollar but undiscounted cost and revenue streams—a highly misleading format that does not account for the cost of money during construction or at any other time. This shortcoming is sharply criticized in the Peer Review Report commissioned by the DOE for the PDS (Brodsky et al. 1993, pp. 11–12). The TRC report compensates partly for this deficiency by also presenting figures for net discounted present values (NDPV) (as of 1994, in 1992 dollars, using a real discount rate of 4 percent per year) for each option. This procedure is a valid and relatively straightforward way to account for the real cost of money at all times during the project (including the equivalent of interest during construction)—see Appendix B to this chapter—although as noted the choice of the discount rate can be problematic. When we use the NDPV approach in this report, we present results for real discount rates of 7 percent per year.

Comprehensiveness

In comparative economic analysis of alternative options it is important not only to be appropriately comprehensive in including the costs of all of the relevant facilities and operations, but also to be *comparably* comprehensive in the costing of the different options: all of the important elements must be included for all of the options. A particularly common pitfall in this respect in the case of

economic analysis of nuclear-energy options is incomplete or inconsistent treatment of costs of radioactive waste management, including the dismantling and disposal (D&D) of the fuel-cycle facilities themselves at the end of their useful lives.

In U.S. practice, costs of managing low-level radioactive wastes generally are included in facility operating costs, based on a combination of experience and estimates concerning future requirements, while for ultimate disposal of high-level wastes—with which there is as yet no relevant experience—a highly approximate estimate of \$0.001 per kilowatt-hour is added to the fuel-cycle costs. (In some of the cost comparisons in DOE's PDS, this assessment for the cost of high-level waste disposal appears to have been omitted from MOX fuel-cycle costs while included for low-enriched uranium.)

Regarding D&D costs for nuclear reactors, the latest estimate of DOE's Nuclear Energy Cost Data Base group at the Oak Ridge National Laboratory (Delene and Hudson 1993) gives the formula

$$\text{D\&D (million 1992 dollars per unit, paid at shutdown)} = B + 0.020 (P - 1200)$$

where the base cost B is \$145 million for pressurized-water reactors, \$185 million for boiling-water reactors, and \$165 million for other reactor types, and where P is the unit thermal power in megawatts. Costs are constant at the 1,200-megawatt-thermal (MWt) value for unit sizes smaller than this and constant at the 3,400-MWt value for sizes larger than that. For other facilities, such as fuel fabrication plants, DOE's PDS recommended use of the "rule of thumb" that the at-shutdown cost of decommissioning is 10 percent of the construction investment as of startup. It is customary either to include in nonfuel operating costs the appropriate annual payment to an accumulating fund (equal to the at-shutdown figure times $r / [(1 + r)^n - 1]$, with r the real interest rate and n the operating lifetime in years) or capitalize the D&D cost as an increment to the initial investment I (equal to the at-shutdown figure divided by $(1 + r)^n$).

Another "comprehensiveness" issue that arises in comparative costing of plutonium disposition options is the need for consistent treatment of the costs of any required conversion of plutonium from the metallic form in which it is found in the pits from dismantled nuclear weapons to oxide or other forms required by particular disposition options. In the PDS, some vendors included the cost of conversion from metal to oxide in their overall cost estimates for MOX-burning reactor options, while others assumed that DOE would provide the plutonium to the fuel-cycle operators in oxide form, cost-free. Because not all options would require conversion to oxide, a fair economic comparison of all of the possibilities clearly requires that those options requiring this conversion should be assigned the costs of conversion.

Other problems of comprehensiveness identified by the TRC in the vendor studies for the DOE's PDS included (USDOE 1993a, pp. SC-2, SC-6):

- across-the-board neglect of site-support and infrastructure costs that would be incurred even at government sites, as well as of a variety of design-phase costs DOE believes should have been included;
- uneven coverage of the costs associated with waste streams;
- uneven representation of seismic considerations in cost estimation; and
- uneven coverage of spent fuel storage costs.

Again, we found it difficult to determine, from the available information, how large the distortions resulting from this array of omissions and inconsistencies might be, or to what extent the TRC was able to correct them in developing its modified cost estimates. We consider that these ambiguities produce uncertainties of the order of ± 25 percent in our comparison of DOE cost estimates with each other and with estimates for other options.

Net Versus Gross Costs and Revenues and the Treatment of Sunk Costs

Perhaps the most fundamental conceptual issues in the economic evaluation of plutonium disposition options are (1) distinguishing the net costs and revenues attributable to the plutonium disposition mission itself from the gross costs and revenues associated with larger contexts in which the disposition mission may be embedded, such as generation of electricity, and, relatedly, (2) determining which if any of the costs incurred in the past in connection with facilities and options related to the disposition mission should be considered as "sunk," that is, ignored in the preparation of cost estimates to be used in the characterization of the alternative options. For these purposes, it is useful to distinguish four types of situations according to whether (a) the facilities involved are pre-existing or new and (b) their use in the plutonium disposition mission will be single-purpose or multipurpose. Table 3-8 provides a matrix illustrating this categorization. In what follows, we treat in turn the marginal-versus-gross and sunk-cost considerations appropriate to each of the categories.

- (1) *Single-Purpose Use of New Facilities.* In this simplest case, new facilities are constructed and operated for no purpose other than the disposition of WPu, and no marketable products or other quantifiable economic benefits to society result besides the disposition of the WPu. In such situations, all of the capital charges and operating costs clearly are assignable to the plutonium disposition mission, there are no offsetting revenues or other economic benefits, and no costs need be ignored as sunk. Either the NDPV approach or the levelized annual cost (LAC) approach can be used in a straightforward way to characterize the costs in such a case.

TABLE 3-8 Cost-Estimation Categorization of Plutonium Disposition Facilities and Uses, With Examples

Use	Preexisting Facilities	New Facilities
Single-purpose	"Igloos" at Pantex weapon assembly/disassembly facility, used for WPu storage	Dedicated (nonelectricity-producing) plutonium burner reactor designed for this purpose
Multi-purpose	Existing commercial pressurized-water reactor using MOX fuel to process WPu and generate electricity	Advanced reactor constructed at government site for WPu processing and electricity generation

- (2) *Single-Purpose Use of Preexisting Facilities.* Because in this case the use is still single-purpose, all of the operating costs are assignable to the plutonium disposition mission as in the previous case. Capital charges should only be assigned to the mission, however, to the extent that (a) they are associated with renovation or modifications of the facility that were required to enable it to perform the mission or (b) the facility had residual value in another role from which it is being displaced by the plutonium disposition mission. The first circumstance can be accounted for straightforwardly in either the NDPV or LAC cost-estimation approaches; the second circumstance requires some care in the determination and representation of residual value, but can also be handled in either cost-estimation approach. Capital costs associated with the facility's earlier uses for other purposes should be ignored as sunk.
- (3) *Multipurpose Use of Preexisting Facilities.* This situation typically arises in connection with plutonium disposition options that generate electricity, but it also would apply to the incorporation of WPu into glasses being produced in facilities that would have to exist anyway for the purpose of disposing of high-level radioactive wastes. The costs that should be assigned to the plutonium disposition mission in such cases are only those attributable to the modifications—for purposes of plutonium processing—of the preexisting system, including any changes in operating cost and any replacement costs or avoided costs associated with decreased or increased outputs (e.g., steam, electricity, vitrified wastes). Either NDPV or LAC cost-accounting can be used, as long as care is taken to include only the *changes* attributable to the addition of the plutonium in whichever of the two prescriptions is chosen.

- (4) *Multipurpose Use of New Facilities.* If new facilities are constructed for the primary purpose of plutonium disposition but these also generate revenues (as from electricity or steam) or some other useful but unpriced product, the problem in costing the plutonium disposition is deciding how much of the project's costs to subtract in the form of credit for the other output(s). If the project in question generates electricity, for example, it still differs from the counterpart situation (3) in that the "byproduct" electricity now *replaces* electricity that society otherwise would be generating in some other way. In this case, the project should be credited with the "avoided cost" associated with not having to supply that electricity in the way that would be chosen in the project's absence. We take this figure to be $\$0.05 \pm \0.015 per kilowatt-hour (1992 dollars in 2015, the midpoint of a nominal 2000–2030 operating lifetime), as explained in Appendix C at the end of this chapter. The avoided-cost approach is obviously applicable, as well, to byproduct services other than electricity—including unpriced ones, for which one would credit the project with the cost of producing these services by the most plausible alternative means. The approach lends itself to use in either the NDPV or LAC costing models.

These prescriptions for the costs attributable to plutonium disposition in the four situations are summarized in compact form in Table 3-9. A given disposition option will often be made up, of course, of a variety of activities falling into more than one of the above categories, in which case a proper accounting of the costs may require particular care in matching the appropriate costing conventions to different parts of the project.

In DOE's PDS, in which all of the options considered involved the construction of new reactors for the combined purpose of plutonium disposition and electricity generation, the avoided costs were taken to range from about $\$0.03/\text{kWh}$ in the year 2000 to about $\$0.05/\text{kWh}$ in the year 2050 (1992 dollars) (USDOE 1993b, based on Hudson 1993). These figures are somewhat lower than ours mainly because their author (Hudson 1993) used a more complicated estimation scheme that took separate account of "energy credit" and "capacity credit": he assumed that 80 percent of the electricity displaced by the plutonium disposition option came from coal (with lower energy costs than for gas), while only 20 percent came from gas; and, while he assumed all the displaced *capacity* was gas-fired combined-cycle (consistent with our approach), he reduced the capacity credit by applying a low (45 percent) "firmness" factor based on an earlier study of government tritium-production reactors—in which it was supposed that electric utilities could not place high reliance on the capacity because of the unpredictability of the government's tritium-production schedule. Our approach is both simpler and more appropriate, we think, to the circumstances of plutonium disposition in a MOX-burning power reactor that is not producing

TABLE 3-9 Summary of Prescriptions for Costs Attributable to Plutonium Disposition in Different Situations

Use	Preexisting Facilities	New Facilities
Single-purpose	All operating costs plus incremental capital costs	All operating costs plus all capital costs
Multi-purpose	Incremental operating plus incremental capital costs	All operating plus capital costs less avoided costs

NOTE: "Incremental" refers to modifications needed to accommodate the plutonium disposition mission.

tritium and can be operated in the same manner as any other baseload power plant.

It should be noted, finally, that the PDS documents give the impression at several points that that study's economic treatment of byproduct revenues will be based on the supposition that the electricity-generation portion of dual-purpose nuclear-power plants is owned by a private entity, to which the government-owned plutonium disposition reactor will sell steam. In such a circumstance, it would be possible to calculate a reasonable figure for the steam credit to be subtracted from the government's plutonium disposition costs, by costing the electricity-generation equipment at private cost of money and subtracting the resulting capital charges and other electricity-generation operating costs from a plausible busbar selling price to get the amount of money the electricity-generating entity should be willing to pay for the steam. In fact, however, the economic calculations finally presented in the PDS documents were not done this way; instead, they simply credited against the total costs of the project—all estimates based on a real cost of money of 4 percent per year—the avoided-cost electricity revenues calculated by Hudson in the manner just described. The effects of the inappropriately low cost of money that was assumed and the inappropriately low "firmness" factor embedded in the avoided-cost calculation partly cancel, but the cost-of-money effect is bigger and produces, as a result, an over-optimistic impression about the financial aspects of the new reactor disposition options considered in the PDS.

Operational Lifetime of Facilities

Clearly, the operating lifetime assumed for a facility will have a strong effect on the annualized capital charges through the capital recovery factor, $r \times (1 + r)^n / [(1 + r)^n - 1]$, or, if the NDPV approach is used, on the discounted present values (DPVs) of the streams of operating costs and revenues. Although tax

policies sometimes encourage corporations to depreciate investments in facilities over periods shorter than the operating lifetime (whereupon the n in the formula for the CRF is the depreciation life rather than the operating life), cost comparisons of alternative approaches to obtaining a given service should set n equal to the expected operating lifetime. The operating lifetime of nuclear power reactors has typically been taken as 30 years for purposes of comparative economic assessment (including in most of the vendor studies for the PDS), although reactors may in fact operate for longer than this and NRC operating licenses are for 40 years.

Another economically significant aspect of operating lifetime is the coincidence, or lack of it, between the duration of plutonium disposition operations and that of electricity-generation operations in a dual-purpose facility. For example, some of DOE's studies of advanced-reactor approaches to plutonium disposition took credit for 20 years of additional electricity generation based on low-enriched uranium (LEU) fuel following a 20-year period of operation using WPU in MOX fuel.⁹ A reasonable, conceptual approach to the economics of such a situation is to treat it during the period of dual-purpose operation as the appropriate combination of cases (3) and (4), as described above, and then at the end of this period to credit the project with the residual value of the facilities based on their potential for electricity generation in the remainder of their operational lifetimes.¹⁰

If the economics of the entire operation is calculated as if conducted in the private sector, thus avoiding multiple cost-of-money distortions, it is easily seen that the costs (or benefits) attributable to plutonium disposition will—indeed must—be identical whether the plutonium-disposing entity continues to operate the facilities itself after the end of the plutonium disposition phase or, instead, sells the facility at the end of the plutonium disposition phase, at the facility's then DPV, to a purely electricity-generating entity. The DOE study's attribution of 40 years of electricity revenue to a 20-year plutonium disposition operation is, therefore, not wrong in principle; but it amplifies the exaggerated impression of profitability associated with calculating the project's costs based on a government cost of money not appropriate for the evaluation of an electricity-generating operation.

The impression, conveyed by the PDS, that the government can not only make a profit by building new power reactors for WPU disposition, but can in-

⁹ This entailed assuming a 40-year total operating lifetime for the reactors, larger than the 30 years stated in most of the vendor studies. It is possible that the higher figure would prove to be correct, however, so this discrepancy is not a major concern.

¹⁰ In the NDPV approach, for example, the value of the facilities at the end of the plutonium disposition phase is equal to the DPV, at that time, of the stream of avoided costs from electricity generation in the remainder of the facilities' operational lives, less the DPV, at that time, of the future operating and decommissioning costs. Working out the analogous procedure for the LAC approach is straightforward.

crease the profitability of the venture by continuing to operate the reactors on LEU after the WPu is gone, should already be seen to be suspect when it is noticed that, according to the DOE analysis, allocating the same amount of WPu to a larger number of reactors (and thus using a smaller fraction of the total operating lifetime of each for the plutonium disposition mission) greatly increases the profitability of the enterprise. That is, the more reactors are used in the disposition mission and the less plutonium each one processes, the more money the disposition operation appears to make. The reason for this bizarre result, of course, is that the basis of the apparent profitability of the enterprise is not the value of the plutonium for electricity generation but, rather, that the plutonium disposition mission has been assumed to justify the government's building electric power plants with money borrowed at publicly subsidized interest rates and then operating these plants in competition with power plants financed at much higher rates in the private sector.

ISSUES AND CRITERIA RELATING TO ENVIRONMENT, SAFETY, AND HEALTH

The greatest dangers to public welfare associated with the existence and disposition of WPu are unquestionably those connected with national and international security—that is, the dangers associated with the potential uses of this material in nuclear weapons, as well as the dangers that could be posed for global arms reduction and nonproliferation prospects by failure to manage the WPu in a manner widely understood to preclude its reuse in weapons. The pre-eminence of these security dangers, however, should not obscure the need for careful attention to the environment, safety, and health (ES&H) risks posed by the WPu under the different possible options for its disposition.

As is well known, plutonium itself poses radiological hazards, and the production, separation, and processing of the WPu used in the U.S. and former Soviet Union arsenals has been associated with a dismaying—and still unfolding—array of ES&H problems, the residues of which may cost hundreds of billions of dollars to clean up (National Research Council 1989, IPPNW/IEER 1992, EESI 1993). The circumstances under which these ES&H burdens were generated—above all, the isolation of weapon production facilities and bureaucracies, under Cold War secrecy, from the scrutiny of the public and the oversight of regulatory agencies—no longer exist. Nor is there any need, as some may have felt at times in the past, to push ES&H concerns aside in order to expedite programs crucial to national security. Certainly, reducing the security risks posed by surplus WPu is the paramount goal in choosing a disposition option for this material, but that goal can and must be accomplished subject to reasonable ES&H constraints.

Proposed ES&H Criteria for Disposition of Weapons Plutonium

We regard it as very important that the governments involved express in the strongest terms their commitment to respect such constraints, and that they demonstrate this commitment by promulgating promptly an appropriate set of ES&H criteria for the plutonium disposition process and by putting in place whatever supplemental institutional mechanisms and resources may be required to give confidence that those criteria will be met. We propose, as appropriate criteria, that any disposition option to operate in the United States

- (1) should comply with U.S. regulations governing allowable emissions of radioactivity to the environment, and allowable radiation doses to workers and the public, from civilian nuclear-energy activities;
- (2) should comply with international agreements and standards covering the disposition of radioactive materials in the environment; and
- (3) should not add significantly to the ES&H burdens that would be expected to arise, in the absence of WPu disposition, from appropriate management of the environmental legacy of past nuclear-weapon production and from appropriate management of the ES&H aspects of past and future civilian nuclear-energy generation.

Disposition options in Russia or in other countries should meet the same three criteria, with the single modification that in criterion (1) the regulations that apply would be those of the country involved rather than those of the United States.

We understand, in proposing this set of criteria, that some will argue that certain of the indicated national and international regulations are unduly restrictive (meaning that they impose economic or other burdens disproportionate to their ES&H benefits), even as others argue that some of these regulations are too lax. We are obviously not in a position, in this study, to reexamine the scientific basis and cost-benefit balance for all of the relevant regulations. But we think a formulation like the one proposed here could gain widespread acceptance as a practical basis for proceeding, and that it would be both unnecessary and unwise to allow the plutonium disposition mission to become hostage to reaching agreement on either tightening or loosening national standards on emissions and doses, or international ones on disposition of radioactivity in the environment.

Rationale for the Proposed Criteria

To contend, as we do, that the three ES&H criteria proposed above are appropriate is the same as contending that they are not only necessary but also sufficient. Let us consider necessity and sufficiency in turn.

We think the first two criteria are necessary because to argue that looser standards are needed to get the job done could generate such strong opposition

as to paralyze the processes of decision-making and implementation. The resulting delay could be highly injurious to the security goals driving the WPu disposition program. The third criterion is necessary because the standards on emissions, doses, and disposition of radioactivity in the environment incorporated in the first two criteria do not cover all of the ES&H characteristics of potential concern, and because to argue that the plutonium disposition mission requires accepting a significant increase in any of the much-debated impacts of what are, after all, activities at much larger scale both in civilian nuclear-energy supply and in management of the environmental legacy of nuclear-weapon production would, as with abandonment of either of the first two criteria, be likely to generate widespread objection and intolerable delay.

The argument for sufficiency depends on the implications of all three criteria taken together, and it has three parts.

- (a) The standards on emissions, doses, and disposition of radioactivity in the environment—of which key examples are described in “Some Relevant Standards Limiting Doses and Emissions,” p. 94—have been constructed to ensure that the health risks to the most exposed members of the public, from the radiological impacts of nuclear facilities in compliance with these standards, are much smaller than the risks of the same types experienced by individuals in the same population from other causes (see, e.g., IAEA 1982, 1986; OFR 1992a). The risk to the average, as opposed to most exposed, member of the public is necessarily much smaller. Appendix D at the end of this chapter provides some illustrations.
- (b) The radiological risks from plutonium disposition, in categories that would not be covered, or would not necessarily be adequately limited, by the standards mentioned in criteria (1) and (2) (such as the radiological risks from large accidents at nuclear reactors, criticality accidents in plutonium/waste mixing processes, or failures of confinement at nuclear waste repositories, the projected health damages to future generations from uranium-mill tailings, and risks to workers if they actually were exposed for a working lifetime to the currently permitted occupational dose rates¹¹), would be confined by criterion (3) to be a small addition to the risks of these kinds that do or will exist in any case from responsibly managed nuclear electricity generation and military nuclear waste disposal. These radiological risks of nuclear activities have been extensively studied and documented. (Among many major reviews in the last 20 years, we mention USEPA 1973; USNRC 1975, 1976, 1987, 1989; OECD 1976; APS 1978, 1985; NAS 1979; National Research Council 1980, 1992; and OTA 1984.) Although un-

¹¹ Concerning the worker doses, see Appendix D at the end of this chapter.

certainties remain, and although agreement is not unanimous that the radiological risks of nuclear electricity supply are acceptably small, the fact is that society *is* now bearing these risks in connection with the benefits of electricity supply or *will* bear similar ones in connection with the necessity of managing military nuclear wastes. It is difficult to argue, then, that a small addition to these preexisting and prospective risks should be considered too high a price to pay for the very large added benefits, in the security realm, of disposition of WPu.

- (c) The nonradiological impacts of nuclear-energy activities—in such categories as the alteration of land and vegetation for facility construction, consumptive use of water, emissions of chemical pollutants, the usual array of industrial hazards to workers (falls, maiming by machinery, electrical shock, and so on)—are not markedly different in kind or magnitude from those of other energy and nonenergy industrial activities of similar economic scale (except in the case of chemical emissions, which are generally smaller for the nuclear activities than for the others) (National Research Council 1980, Holdren 1982, Holdren et al. 1983); and these nonradiological impacts have not been a primary focus of public or regulatory concern about nuclear energy (OTA 1984, National Research Council 1989, Hohenemser et al. 1990). Accordingly, it seems clear that if the nonradiological ES&H impacts of WPu disposition meet our third criterion—i.e., they do not significantly add to the nonradiological impacts of existing and prospective nuclear-energy and military nuclear waste management activities—then they cannot logically be considered too high a price to pay for the very substantial security benefits of disposition.

After some decades of attention to environmental assessment and regulation of nuclear and nonnuclear technologies alike, sufficiency in ES&H performance has come to mean that the ES&H damages from an activity should be (1) small compared to damages of similar types from other causes, (2) small compared to the benefits of the activity, and (3) smaller than or comparable to the damages from other activities that generate comparable benefits (see, e.g., Budnitz and Holdren 1976, Holdren 1982, Hohenemser et al. 1983).¹² While uncertainties of many kinds preclude absolute declarations on such matters, points (a)-(c) strongly suggest that our three proposed ES&H criteria for plutonium-disposition options can be considered sufficient in these senses.

¹² An additional desideratum that is sometimes mentioned is that the damages should be at the level where reducing them further would cost more than the value of the reduction. The specificity and difficulty of the analysis required to establish that this condition is met, however, makes it impractical to apply it at the level of preliminary screening of options (to which the present study is confined).

SOME RELEVANT STANDARDS LIMITING DOSES AND EMISSIONS

The relevant U.S. regulations are under the jurisdiction of the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA) and are described in the U.S. Code of Federal Regulations, Titles 10 and 40 (OFR 1992a and 1992b).

The NRC standards limit the whole-body-equivalent dose to workers in the nuclear industry to 0.05 sieverts (Sv) (5 rem) per year. NRC limits for members of the public include: a 0.25 Sv (25 rem) whole-body once-in-lifetime emergency dose limit from a nuclear accident; a limit of 5 mSv/yr (500 mrem/yr) on the whole-body dose that could be received by an individual intruding inadvertently into a shallow burial site for low-level radioactive wastes between 100 and 1,000 years after emplacement of the wastes; a limit of 1 mSv/yr (100 mrem/yr) on the whole-body dose from all routine non-medical exposures combined; and a limit of 50 μ Sv/yr (5 mrem/yr) each on the whole-body dose from routine airborne effluents and routine liquid effluents from any single nuclear facility. The EPA standards limit whole-body doses to the public to 1 mSv (100 mrem) per year from all nuclear facilities and 0.1 mSv (10 mrem) per year from any one nuclear facility.

Limits on emissions of radioactivity include an NRC limit on total emissions, except tritium and dissolved gases, of 0.185 TBq (terabecquerel) (5 curies; Ci) per year from any nuclear reactor, and EPA limits on emissions from the entire nuclear fuel cycle, per electrical gigawatt-year of output, of 1,850 TBq (50,000 Ci) of krypton-85, 185 MBq (5 mCi) of iodine-129, and 18.5 Bq (0.5 mCi) of trans-

The Main ES&H Issues in Weapons Plutonium Disposition

According to the criteria proposed and justified above, the focus of an ES&H assessment of plutonium disposition options should be on identifying and analyzing those aspects where there is some possibility that a plutonium option either could have difficulty meeting relevant standards or could add significantly to the risks of similar kinds that do or will exist in any case from appropriately managed nuclear electricity generation and military nuclear waste disposal. Given what has already been said about the nonradiological hazards of nuclear-energy activities in relation to the radiological ones, and given also the relatively small physical quantities of WPu in relation to quantities of uranium

uranics.² The emissions limits have been designed to ensure that the dose limits are met.

The international regulations of greatest potential relevance to the plutonium disposition issue are those governing disposal of radioactive material in the oceans. Under the 1972 Convention on the Prevention of Marine Pollution by Dumping Wastes and Other Matter in the Ocean (known for short as the London Ocean Dumping Convention), the International Atomic Energy Agency (IAEA) was charged with developing regulations to restrict dumping of radioactivity into the oceans to levels that pose "no unacceptable degree of hazard to humans and their environment." The resulting IAEA guidelines are based on the proposition that additions of radionuclides to the oceans should not exceed rates that, if continued for 1,000 years, would lead eventually to doses exceeding 1 mSv (100 mrem) per year to the most exposed individuals (IAEA 1986). The models used by the IAEA to estimate these rates consider a variety of pathways by which humans could be exposed to radionuclides from seawater, including ingestion of fish, shellfish, seaweed, plankton, desalinated seawater, and sea salt; inhalation of evaporated seawater and airborne particulates originating from ocean sediments; and external irradiation from swimming and onshore sediments.

¹ The dose in sieverts (Sv) equals the specific energy absorption in grays (1 gray = 1 joule per kilogram of absorber) multiplied by the quality factor (QF = 1 for x-rays, gamma rays, and beta particles, 10 for neutrons, and 20 for alpha particles). The corresponding traditional unit is the rem (1 Sv = 100 rem).

² One becquerel (Bq) is the amount of radioactivity that yields one nuclear transformation per second. The traditional unit of radioactivity is the curie (Ci), which is the amount of radioactivity in a gram of radium-226, or 3.7×10^{10} Bq.

and nuclear wastes involved in nuclear-energy supply and nuclear waste management, it seems clear that the main concerns about the ES&H dimensions of plutonium disposition will arise not from the nonradiological aspects but from the radiological ones. Thus we focus on the radiological aspects here. Which specific radiological issues will be the critical ones in relation to our three ES&H criteria for plutonium disposition depends on whether reactor options or nuclear waste options are under consideration.

For the case of reactor options, the addition of WPu to the nuclear-energy system would introduce or affect the following activities with potential ramifications in terms of our ES&H criteria:

- conversion of WPu metal to PuO₂ (for light-water reactors and some other—but not all—reactor types);
- mixing the oxides and fabricating the MOX into fuel pellets, fuel rods, and fuel assemblies;
- the storage and transport steps associated with the preparation of the MOX fuel, its delivery to the reactor, and its storage there prior to use;
- reduction in the amount of uranium mined, milled, converted, enriched, and fabricated, by virtue of the substitution of MOX fuel for some of the uranium-only fuel that would otherwise have been used;
- any changes in the ES&H characteristics of reactor preparation, operation, and maintenance as a result of the use of WPu in its fuel; and
- any changes in the ES&H characteristics of waste management—including spent fuel storage and transport, further high-level waste processing (for other than once-through systems), emplacement and residence in a geologic repository, and management of low-level and transuranic wastes—that result from the use of WPu in the fuel.

The possibilities of significant impacts on nuclear-energy generation's ES&H characteristics from any and all of these items, then, will need to be carefully examined.

For the case of options in which the WPu is added to nuclear waste streams without prior introduction of the plutonium into reactors, the alterations to the baseline ES&H effects of these waste operations that would need to be considered include:

- the effects of conversion of the WPu metal to whatever form is required as input to the waste-processing operations, and of transporting the plutonium to the waste-processing facility and storing it there;
- the effects of the addition of the plutonium on the ES&H characteristics of the waste-processing operations, including particularly any effects of plutonium's potential to cause a criticality problem in waste processing, and of measures taken to offset this potential;
- the effects of plutonium addition on the ES&H characteristics of waste storage, transport, and emplacement and residence in a geologic repository, including the possible effects of plutonium's potential to cause a criticality problem in the repository at some future time, and of measures taken to offset this potential.

It is clear, from the relative problem-causing potential of the various ES&H issues in the preexisting nuclear-energy supply and nuclear waste management contexts, that particular attention needs to be given to the possible impacts of

WPu on reactor safety, the safety of the plutonium/waste mixing processes, and on the nuclear waste issues. Among the ES&H issues that are *not* particularly problematic in current practice, the one most likely to need special attention with the addition of WPu is the occupational risk from fuel preparation—where the far higher inhalation toxicity of plutonium per gram, as compared to that of uranium, calls for extraordinary precautions. These three issues—the ramifications of plutonium for reactor and plutonium/waste mixing process safety, nuclear waste issues, and occupational hazards of fuel preparation—receive the bulk of our attention to ES&H issues in the remainder of this report.

OTHER CONSIDERATIONS

In addition to the security, economic, and ES&H criteria just described, approaches to management and disposition of excess WPu must be acceptable to the public and the relevant institutions, and should, to the extent possible, avoid conflict with other policies and objectives.

Public Acceptability

Without public acceptance, successful implementation of any management and disposition approach is unlikely. Gaining public acceptance will require attention to ES&H protection, as described above, and encourage a decision-making process with genuine participation by local and national publics in decisions that affect them.

Institutional Acceptability

Similarly, acceptance by the various institutions that must give their approval will be a critical factor in the success of any management and disposition approach. Licensing in particular is likely to be a time-limiting factor in many cases, and clearly predictable difficulties in this regard could affect the choice of options. As with the public, early participation by the relevant institutions is essential.

Other Policies and Objectives

Management and disposition of excess WPu should, as with other activities, be guided by the agreements, laws, regulations, and policies of the state carrying it out. Where a particular approach would appear to contravene existing international agreements, for example, we have considered this a major obstacle.

Similarly, management and disposition of excess plutonium should ideally proceed in a manner supportive of the other policies of the state carrying it out. This includes in particular policies related to nonproliferation and nuclear fuel cycles (see discussion in Chapter 6, “General Considerations”). We do not be-

lieve, however, that promoting the future of civilian nuclear power—or the reverse—should be considered a significant criterion for choice among options for disposition of WPu. That future depends on broader economic, political, and technical factors outside the scope of this study. Finally, we do not believe that whether plutonium disposition options would also have the potential to produce tritium should be a major criterion for deciding among them.

**APPENDIX A:
INTEGRATED INVENTORY**

If $Q_i(t)$ denotes the inventory of WPu in the i th step at time t , the integrated inventory IT (units ton-years) is given by the integral from t equals minus infinity to plus infinity of $Q_i(t)dt$:

$$IT = \int_{t=-\infty}^{\infty} Q_i(t) dt .$$

In the case of a processing or storage step of finite duration, it is useful to think of this quantity as the average inventory in the step times the length of the period over which that step is operative in the plutonium disposition campaign. In the case of an initial inventory that is subsequently depleted over an indefinite period by, e.g., radioactive decay, the integrated inventory is the initial inventory times the average lifetime of a plutonium atom in that initial inventory before it decays.

More generally, an expression for $Q_i(t)$ itself can be obtained starting from the differential equation relating the rate of change of the inventory Q_i to the inflows to and outflows from that inventory,

$$dQ_i / dt = IF_i(t) - OF_i(t) ,$$

which integrates to

$$Q_i(t) = \int_{t'=-\infty}^t [IF_i(t') - OF_i(t')] dt'$$

or, in the case of an initial inventory at time $t' = 0$, subsequently altered by inflows and outflows,

$$Q_i(t) = Q_i(0) + \int_{t'=0}^t [IF_i(t') - OF_i(t')] dt' .$$

Let us consider some examples relevant to the sorts of options we consider in the report for WPu disposition.

Storage

Consider storage of pits as an example. Imagine that, starting at a time denoted $t = 0$, pits begin entering storage from the dismantling process at a rate of 10,000 kilograms per year (kg/yr), which continues for five years until the nominal 50,000 kg of surplus WPu is in this form. Suppose there is no further change in this inventory for another five years, whereafter pits are removed at a rate of 5,000 kg/yr to be converted into plutonium oxide. After 10 years at the

latter rate, no more pits will remain in storage. In this situation $Q(t)$, in units of kilograms, is given by

$$\begin{array}{ll}
 Q(t) & = 0 & t < 0 \\
 & = 10,000 \times t & 0 < t < 5 \text{ yr} \\
 & = 50,000 & 5 < t < 10 \text{ yr} \\
 & = 50,000 - 5,000 \times t & 10 < t < 20 \text{ yr} \\
 & = 0 & t > 20 \text{ yr.}
 \end{array}$$

The integrated inventory associated with this storage is then

$$\begin{aligned}
 IT &= \int_{t=0}^{20} Q(t) dt = \int_{t=0}^5 10,000t dt + \int_{t=5}^{10} 50,000 dt + \int_{t=10}^{20} (50,000 - 5,000t) dt \\
 &= 125,000 + 250,000 + 250,000 = 625,000 \text{ kg-yr} ,
 \end{aligned}$$

equivalent to an average inventory of 31,250 kg over the 20-year duration of the pit-storage phase of the campaign. The generalization to cases in which material is being added and removed at the same time is obvious.

Processing Steps

Storage of material taking place at a processing site would be treated in the manner just described. If, however, the characteristics of the process more than the characteristics of the site produce security risks that make it desirable to characterize separately the actual *in-process* integrated inventory, this can be done by calculating the average in-process inventory from

$$\text{avg inventory} = \text{avg throughput (kg/day)} \times \text{residence time (days)} ,$$

where "residence time" means the average time a kilogram of material spends in processing. Then the integrated inventory is given by

$$IT = \text{avg inventory (kg)} \times \text{duration of processing activity (yrs)} .$$

Transport Steps

Transport steps are treated in a manner analogous to that for processing steps. That is, if there are significant storage operations associated with transport on either end of the trip, these are treated in the manner described above for storage, while the integrated inventory for the actual trips is calculated, as for a process, by first determining the average "underway" inventory as

$$\text{underway inventory (kg)} = \text{annual transport (kg/yr)} \times \text{trip duration (days/365)}$$

and then calculating the integrated inventory as

$$IT = \textit{underway inventory (kg)} \times \textit{duration of transport activity (yr)} .$$

Ultimate Disposition

It would be fairly straightforward to calculate also an integrated inventory figure for the final phase of a plutonium disposition campaign, in which for example the plutonium is embedded in spent fuel or vitrified high-level waste that remains indefinitely in engineered storage or a geologic repository. In that case, the integrated inventory would be the initial inventory in this form multiplied by the radiologic mean-life of the plutonium, which is the half-life divided by the natural log of 2. Complications are presented by the circumstance that different plutonium isotopes have different half-lives, so that the calculation must be carried out isotope by isotope, and also by the circumstance that the decay daughter of the most important plutonium isotope, plutonium-239, is uranium-235, which is also a nuclear explosive and has a half-life of some 700 million years. We do not think, however, that an integrated inventory figure for ultimate disposition would be an informative index of security hazard even if corrected for these complications—certainly it could not be compared meaningfully with the far smaller integrated inventories associated with the earlier, much shorter phases of plutonium disposition—in part because material in this final phase would probably be much less attractive to would-be bomb-makers than would be plutonium or uranium-235 in other forms and locations, and in part because, even if it were or might be attractive, we know of no persuasive prescription for weighing a hazard distributed over so long a time period against hazards that would be experienced in the near future. Thus, while we characterize, for comparison with other phases and for comparisons among disposition options, the barriers to weapons use of the plutonium in its ultimate disposition form, we do *not* calculate an integrated inventory figure for this phase.

**APPENDIX B:
LEVELIZED ANNUAL COSTS
AND NET DISCOUNTED PRESENT VALUE**

If the real cost of money is r (a pure number obtained by dividing the annual percentage rate by 100), it is easy to show that the levelized constant-dollar annual payment needed to retire an initial investment of I dollars over a period of n years is equal to

$$I \times r \times (1+r)^n / [(1+r)^n - 1].$$

This is the size of the annual payment—levelized in the sense of being constant in constant dollars throughout the n -year period—needed to “make whole,” at the end of the period, an investor of the initial sum I who has the option of putting the money in a secure savings account with real rate of return r . (Of course, if the investment in question is less secure than a savings account, the rational investor will insist on a “risk premium” in the form of a higher real rate of return than the one that corresponds to the savings account.)

The term,

$$r \times (1+r)^n / [(1+r)^n - 1],$$

is called the (real) capital recovery factor (CRF). The nominal—as opposed to real—CRF is calculated from the same formula with r replaced by $r' = r + i$, where i is the annual rate of inflation. The product, $\text{CRF} \times I$, accounts for both interest and the repayment of principal (equivalent to depreciation)¹³. The levelized annual capital charges (LACC) associated with a project are given by the sum of the interest and depreciation term, $\text{CRF} \times I$, with any other annual payments that are proportional to capital investment (e.g., property taxes and insurance). The fixed charge rate (FCR) is the factor that, when multiplied by the capital investment as calculated at the beginning of operation, gives the LACC:

$$\text{LACC} = \text{FCR} \times I.$$

$$\text{FCR} = \text{CRF} + \text{property taxes, insurance, etc., as an annual fraction of capital investment.}$$

The total levelized annual costs (LAC) of a project are the sum of the LACC plus levelized annual operating costs (LAOC) such as fuel, spare parts, and labor for operation and maintenance:

$$\text{LAC} = \text{LACC} + \text{LAOC} = \text{FCR} \times I + \text{LAOC}.$$

The procedure for levelizing operating costs that occur nonuniformly over the operating life of the project is straightforward; it is analogous to the way the

¹³ The formula and the concept are identical to those applicable to home mortgages, except that the latter are ordinarily arranged so that the annual payment is levelized in current rather than constant dollars (meaning the formula contains r' in place of r).

The procedure for levelizing operating costs that occur nonuniformly over the operating life of the project is straightforward; it is analogous to the way the capital investment is converted into an equivalent annual payment. Other streams of payments that occur before or after the period of operation, such as the costs of decommissioning and costs of long-term waste management, are customarily accounted for by appropriate adjustments in the calculation of I and LAOC. (For example, preoperational costs other than construction—such as research and development and licensing costs—can be readily incorporated into the calculation of I , while decommissioning and long-term waste costs are customarily converted into an increment to LAOC, based on the annual contributions needed to an interest-bearing account in order to yield, at the end of the n -year period of operation, the sums estimated to be needed for these activities.)

The LAC approach to economic evaluation answers the question, “If all of the costs of a project, including the time-value of the money needed for all phases of the project’s implementation, are to be recovered through a stream of identical constant-dollar payments made annually during the operating life of the project, what must be the size of that annual payment?” This approach is customarily used in situations where one wishes to know the monetary value to attribute to (or how much to charge for) a product or service that will be delivered more or less continuously during the life of the project (e.g., electricity from a power plant) or where a set of uniform annual payments is a convenient way for the beneficiaries to pay for something they would have difficulty buying outright (e.g., a home mortgage).

The discounted present value (DPV) of an expenditure or income of C dollars at a time n years into the future is given by $PV = C / (1 + d)^n$, where d is the real discount rate (usually taken to be the same as the real cost of money, r), and DPV and C are both expressed in constant dollars of a specified year. (This need not be the same year as the one chosen as the “present” for the present-value calculation; that is, one can perfectly well calculate the present value in the year 2000 of an expenditure in the year 2020, measured in 1992 dollars.) The DPV of a stream of expenditures C_i in years $i = 0$ to n , where 0 refers to the year chosen as the “present,” is then

$$DPV = \sum_{i=0,n} \left\{ C_i / (1 + d)^i \right\} .$$

The net discounted present value (NDPV) of a project is the difference between the net present value (NPV) of the revenues or benefits from the project and the NPV of the costs associated with it, hence

$$NDPV = \sum_{i=0,n} \left\{ (R_i - C_i) / (1 + d)^i \right\} ,$$

where R_i is the revenue in the i th year and C_i is the expenditure, all expressed in constant dollars. In contrast to the LAC calculation, the n in this NDPV formula does not necessarily correspond to the end of the project's useful life but is instead the last year in which either revenues or costs associated with the project arise. The formula can be generalized to include costs and revenues that arise earlier than year 0—i.e., before the “present” in the present-value calculation—simply by letting i take on negative values.

The NDPV approach to economic evaluation answers the question, “What lump sum of money should a rational economic actor be willing to pay at a specified “present” time in order to acquire a project's future net income (NDPV positive) or to avoid its future net loss (NDPV negative)?” This approach is customarily used to compare the economics of alternative ways to accomplish a given task in a given time frame¹⁴—the way that has the highest (most positive) NDPV is the best.

¹⁴ Without the requirement that the time scale should be the same, this evaluation method would have the shortcoming of seeming to favor ways to accomplish a costly task slowly over ways to accomplish it quickly (since, all else being equal, incurring the costs later rather than sooner makes the NDPV less negative).

**APPENDIX C:
AVOIDED COST AND ASSOCIATED PITFALLS**

The avoided-cost idea is clear in concept, but choosing a figure for the avoided cost poses problems. Consider the case of a new nuclear-power plant built by the government for the purpose of plutonium disposition. On what basis should the avoided-cost credit for its electricity output be calculated? All of the obvious choices are problematic.

- (i) The national-average or even regional-average price of electricity to consumers would not be a satisfactory basis, because such prices must cover the costs of transmission and distribution as well as of generation.
- (ii) Using the national-average or regional-average cost of electric utility generation (“busbar” cost) would avoid the extraneous inclusion of transmission and distribution, but would still be problematic in that the average cost typically is lower than the avoided cost (because the new source replaces the costliest of the existing ones).
- (iii) Calculation of actual avoided costs in a given service region, although widely practiced in connection with determinations by public utility commissions of how much must be paid by electric utilities to independent power producers, entails complicated considerations of capacity credits and energy credits that are highly dependent on the circumstances of individual service regions as well as on the characteristics of the new source; thus this approach does not lend itself to use in a preliminary, national-scope assessment of the sort we are undertaking here.

The foregoing approaches also suffer from uncertainty about the future: Will other electricity costs have gone up or down by the time that reactors are in operation using WPu?

A further problem that can arise with these approaches is the misleading impression produced when avoided-cost estimates based on private-sector generating costs are credited against the costs of plutonium disposition options assumed to have been financed at the low cost of money associated with government borrowing. The result is an artificially low net cost—or an artificially high net revenue—that arises from nothing other than the government’s capacity to borrow money for power-plant construction at lower rates than the private sector can borrow. The apparent net gain to a government project from this discrepancy in private versus public cost of money is real in the sense that the government could, in principle, actually collect electricity revenues from its project that are based on the private-sector avoided costs, and in this way could reduce the apparent net costs of the project; but it is artificial and misleading in the

sense that government borrowing at lower-than-private-sector rates amounts to a public subsidy which, while regarded as appropriate for strictly governmental functions in the common interest, is regarded as inappropriate, in our society, for activities that compete with the private sector. (That is why the Office of Management and Budget (OMB) insists that economic evaluations of government projects that interact in any way with the private sector should assume a real cost of money comparable to private-sector rates rather than the lower cost of money associated with government borrowing.)

The foregoing problem is largely avoided if the government project is costed based on OMB's recommended real cost of money of 7 percent per year, approximating a private-sector rate. The fact that government projects do not pay for property taxes or insurance, however, can be considered a further subsidy, which, accordingly, confers an additional artificial economic benefit on a government project that is credited with private-sector avoided costs.¹⁵ As noted above, we here calculate fixed charges based on 7-percent per year real cost of money, both with and without an increment of 2 percent per year of initial investment for property taxes and insurance.

With the private-versus-public cost of money pitfall circumvented in this way, a temptingly simple prescription for calculation of the avoided-cost credit would be to assume that the avoided cost is that associated with an identical nuclear reactor using low-enriched uranium (LEU) rather than mixed-oxide (MOX) fuel. This assumption would reduce the new facility, multipurpose case (lower right in Table 3-9) to the preexisting facility, multipurpose case (lower left in Table 3-9), for which it is only necessary to calculate the incremental costs associated with substituting MOX for LEU. This approach would escape most of the region-specific complexity of the usual avoided-cost calculations, and it would have the further benefit of reducing the associated uncertainties about the future to a relatively circumscribed set of questions attached to the future economics of MOX and LEU. It would actually be realistic, moreover, for any region in which there is a plausible case that (1) new baseload generating capacity will be needed in the time frame at issue for plutonium disposition and (2) nuclear energy would be a reasonable choice to meet this need.

The first of these conditions—a plausible need for new baseload generating capacity over the next 5–10 years—is satisfied for most places in the United States where plutonium disposition might be contemplated. The weakness in the case for using the same-reactor-but-with-LEU approach to avoided cost is in the second condition: few analysts today would argue that nuclear plants would be likely to be chosen, in the absence of the plutonium disposition mission, for new

¹⁵ For a 7-percent per year real cost of money and nominal plant lifetime of 30 years, the fixed charge rate without allowance for property taxes and insurance would be 0.0806 per year; an allowance of 2 percent for these costs would make it 0.1006 per year, representing about a 25-percent increase in the fixed charge rate and hence in the annual capital charges.

capacity needs in the United States in the next 5–10 years. And to the extent that the options more likely to be chosen would have electricity costs lower than those of nuclear plants using LEU, basing the avoided-cost estimates on the nuclear case would overestimate those avoided costs and thus lead to an underestimate of the true cost of plutonium disposition using the MOX option.

These considerations point to the conclusion that the best way to estimate avoided costs for our purposes is to project the cost of the electricity generated by the options most likely to be chosen in the next decade for new baseload capacity, in the absence of a plutonium disposition program, in the regions where new (or newly completed) reactors for plutonium disposition would operate. These are the electricity sources on which reliance would be reduced by electricity generation in connection with plutonium disposition.¹⁶

The problem of predicting what electricity-generation technologies will be chosen for new baseload capacity becomes more difficult, obviously, the further in the future is the time frame in which one is interested. For the period 5–10 years hence, however, in which plutonium disposition using newly built (or newly completed) reactors of existing types could begin, a rather good case can be made that the baseload-generation technology of choice in most parts of the United States is likely to be combined-cycle, natural-gas-fired power plants.¹⁷

When the cost of electricity from such plants in the period 2000–2030 is estimated using the approaches and assumptions described in this section, with allowance for uncertainties in the future price of natural gas, the result of the levelized annual cost approach is \$0.04–\$0.06 per kilowatt-hour (kWh);¹⁸ a range of \$0.05–\$0.08/kWh appears in a recent review of mostly rather small projects for combined-cycle gas-fired baseload electricity generation (Kahn et al. 1993), but a recent analysis of the economic prospects of operating pluto-

¹⁶ The avoided-cost estimates obtained in this way will of course differ from those that would be obtained by the same-reactor-but-with-LEU method only insofar as the projected generation costs of the expected new capacity differ from those projected for plutonium disposition reactors using LEU instead of MOX.

¹⁷ See Kahn et al. (1993), Hudson (1993), USDOE (1993b). This is what is planned, for example, in Washington state, which would be a leading contender for exercising the MOX option for plutonium disposition by virtue of the location there of a partially completed MOX fabrication facility as well as two partially completed nuclear reactors potentially available for the plutonium disposition mission.

¹⁸ A figure of \$0.047/kWh follows from Hudson's (1993) estimate of \$600 (1992 dollars) per kWe total overnight capital cost for a combined-cycle gas-fired power plant, interest during construction at $r = 0.07$ for 4.5 years yielding a multiplier of 1.2 (interpolated in Table 3-7), capacity factor of 0.75, fixed charge rate = 0.1006/yr (corresponding to $r = 0.07$, $n = 30$ years, and property tax and insurance assessment of 0.02/yr), natural gas cost in 2015 of \$4.25 (1992 dollars) per million Btu (British thermal unit) (USDOE 1993b), heat rate 7,600 Btu/kWh, and nonfuel operation and maintenance costs of 0.004 1992 dollars/kWh (Hudson 1993). Sensitivities: natural gas price ± 30 percent yields $\pm \$0.01$ /kWh; overnight construction cost 30 percent higher adds \$0.003/kWh; real cost of money $r = 0.10$ adds \$0.004/kWh (actual cost of money employed by firms in the wholesale electricity market may be higher still).

nium disposition reactors in the Northwest United States uses a range of \$0.037–\$0.043/kWh for gas-fired combined-cycle baseload power generation in that region (SAIC 1993); the DOE PDS analysis uses \$0.030/kWh for the reference revenue from electricity sales and explores sensitivity to a range from \$0.022/kWh to \$0.060/kWh (USDOE 1993a). (All figures have been converted to 1992 dollars. For our analysis we choose a reference value of \$0.050/kWh with a judgmental 70-percent confidence interval of \pm \$0.015/kWh.

**APPENDIX D:
PREDICTED DAMAGES FROM THE DOSES
PERMITTED BY STANDARDS**

The dose limits are all below the levels that produce symptoms of acute radiation sickness, which are seen only at 0.5–1 sievert (Sv) (50–100 rem) and above. The potential consequences of exposures not exceeding these limits are therefore confined to the so-called “latent” effects, of which the ones of greatest concern are increased incidence of cancer and genetic defects. The dose-response relations governing the magnitudes of the increases to be expected from doses in the indicated ranges have been the subject of much study—and an equal amount of controversy—extending over the past half-century. The source of the controversy is that at dose rates close to those received by everyone from natural background radiation, which are in the range of 1 mSv (100 mrem) per year to the whole body, the increased incidences of cancer and genetic defects predicted from downward linear extrapolation of incidences observed at higher dose rates are too small to be unambiguously detected either in animal experiments of practical scale or in human epidemiological studies with their inescapable array of confounding factors.

Virtually all of the national and international regulatory and advisory bodies dealing with radiation hazards have taken the position for many years, however, that standards and policy should be based on the assumption that increases in the incidence of cancer and genetic defects persist in linear proportion to the dose, down to the lowest doses and dose rates experienced (the “linear hypothesis”).¹⁹ The most recent comprehensive review of this subject by the National Research Council’s Committee on Biological Effects of Ionizing Radiation (National Research Council 1990, p. 4, hereinafter BEIR V) underscores this position, finding that the latest data “do not contradict the hypothesis, at least with respect to cancer induction and hereditary genetic defects, that the frequency of such effects increases with low-level radiation as a linear, nonthreshold function of dose.” The report gives the dose-response relation for cancer as a population weighted lifetime excess risk of death of cancer of 0.8 percent (90-percent confidence interval 0.6–1.2 percent) from a whole-body dose of 0.1 Sv, quickly delivered, and indicates that some reduction in this risk—“possibly by a factor of 2 or more”—is to be expected for gamma and beta (but not alpha and neutron) radiation that is slowly delivered.²⁰

¹⁹ The question is not, as sometimes misstated, the slope of the dose-response curve for doses near zero, but rather the slope of the curve for doses near natural background, since that is the level to which any anthropogenic dose, however small, is added.

²⁰ In contrast to a view often expressed in earlier studies, moreover, that the linear hypothesis provides an upper limit to the plausible consequences of low-level radiation exposure, the BEIR V report states (p. 6) that “The Committee recognizes that its risk estimates become more uncertain when applied to very low doses. Departures from a linear model at low doses, however, could either increase or decrease the risk per unit dose.”

Expressed in terms of population dose (the product of the size of the exposed population and the average dose in that population, measured in person-sieverts or person-rem), the best-estimate dose-response relations are 800 excess cancer deaths per 10^4 person-Sv (million person-rem) for gamma and beta radiation at high dose rates or neutron and alpha irradiation at any dose rate, and 400 excess cancer deaths per 10^4 person-Sv for gamma and beta radiation at low dose rates. The incidence of excess genetic defects induced by radiation is considerably more uncertain, but the estimates given in the BEIR V report are equivalent to 30–70 excess genetic defects per 10^4 person-Sv—an incidence rate some 10 times less than that of excess cancer deaths, and spread over a considerably longer (multigeneration) period. We focus, then, on the cancer deaths as the dominant latent consequence.

The once-in-a-lifetime emergency dose limit of 0.25 Sv, whole body, to a member of the public would correspond, at the dose-response relation of 0.8-percent chance of cancer death per 0.1 Sv (appropriate to rapid delivery of the dose, as could be the case for an individual near a reactor at the time of an accidental release of radioactivity), to a 2-percent chance of dying of cancer. If the dose is delivered slowly (as from ground contamination at a greater distance from an accidental release) and consists mainly of gamma and beta radiation, use of the factor of 2 credit for dose protraction gives a 1-percent chance of dying of cancer from this dose. These figures can be compared to the 20- to 25-percent chance of dying of cancer, in industrial societies, from the sum of all causes. The increase in the chance of cancer death from the allowable one-time emergency dose, then, is in the range of 4 to 10 percent of the preexisting chance. (Of course, the probability that any given individual in a society will actually experience such an emergency dose in his or her lifetime is very low, and this would be the case even if the probabilities of reactor accidents were much higher than we believe them to be.)

A dose-rate limit of 1 mSv per year, whole body, to members of the public from routine releases of radioactivity in various contexts appears in NRC, EPA, and IAEA regulations (see “Some Relevant Standards Limiting Doses and Emissions” on p. 94). Taking the dose-response relation to be 0.4- to 0.8-percent chance of cancer death per 0.1 Sv (the lower figure for gamma and beta radiation, with credit for dose protraction, and the higher one for alpha particles and neutrons, where no such credit is applied), the added chance of cancer death experienced by an individual receiving this dose rate would be 0.004–0.008 percent per year of such exposure. The chance of death from all causes in the U.S. population is about 0.9 percent per year and the chance of death from cancer is about 0.2 percent per year (US Dept of Commerce 1992). The chance of dying eventually of cancer is currently about 19.5 percent in the U.S. popula-

tion, a figure that would be increased to 20.0–20.5 percent²¹ by continuous lifetime exposure at the 1-mSv/yr limit (National Research Council 1990). At the 0.1-mSv/yr limit allowable in NRC and EPA regulations for exposures received by an individual member of the public from a single facility, the predicted incremental probabilities of death from cancer would of course be 10 times smaller.

The occupational dose limit of 0.05 Sv per year would correspond, at 0.4–0.8 percent probability of cancer death per 0.1 Sv, to an extra chance of cancer death of 0.2–0.4 percent per year. Exposure to this dose rate continuously from the age of 18 to the age of 65 would produce an extra probability of cancer death of 15 percent—raising the preexisting probability of death from cancer from about 20 to 35 percent—according to the best estimate of the BEIR V report based on a calculation accounting for age- and gender-specific susceptibilities in a working population of half men and half women. Actual average doses in the nuclear industry are, fortunately, about 10 times lower than permitted by the standard. Downward revision of this standard is under consideration.

²¹ This is based on a 90-percent confidence interval of 0.5–1.0 percent for the increment, according to a calculation in BEIR V accounting for age- and gender-specific susceptibilities. The three-figure precision on the total is illusory but serves to indicate the size of the change.

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Reactor Options

This chapter describes each of the reactor options examined by the panel. For each option, we offer:

- (1) a description of the technology and its development status, including how the technology would be used to process plutonium;
- (2) a description of the factors affecting timing of the option, including:
 - (a) technical uncertainty,
 - (b) reactor capacity and throughputs,
 - (c) fuel fabrication, and
 - (d) licensing and public acceptance issues;
- (3) a description of issues related to safeguards and security during the processes required for the option;
- (4) a description of issues related to accessibility for use in weapons of whatever plutonium would remain in the spent fuel;
- (5) a discussion of economic issues;
- (6) a discussion of environment, safety, and health (ES&H) issues; and
- (7) a discussion of other issues that may be important to policy-makers in choosing between options.

We begin by discussing several options involving the use of current-generation reactors, then turn to more advanced reactor systems. The description of the first option—the use of light water reactors of existing designs—will be the most detailed, both because it is an option of particular interest and because

other reactor options can be considered in part by comparison to that basecase. More detailed comparative assessments of the options based on criteria related to security, cost, and ES&H can be found in Chapter 6.

U.S. PLUTONIUM IN CURRENT-GENERATION U.S. LIGHT-WATER REACTORS

Description of Technology and Status

Light-water reactors (LWRs) are the most mature of any of the proposed burners of weapons plutonium (WPU). Over 100 LWRs are operating in the United States and about 400 worldwide. LWRs have over 4,000 reactor-years of operation. They supply almost 75 percent of the electricity consumed in France and about 22 percent of the electricity generated in the United States.

U.S. LWRs use low-enriched uranium (LEU) fuel. Plutonium in the discharged fuel is not reprocessed for recycle. In Sweden and the United States, geologic repositories are being designed primarily for long-term disposal of spent fuel discharged from uranium-fueled LWRs. Other countries are considering such a direct-disposal fuel cycle as well.

It is also possible to fuel LWRs with mixed-oxide (MOX) fuel, which combines plutonium dioxide and natural or depleted uranium dioxide as a $\text{PuO}_2\text{-UO}_2$ mixture. Work on such MOX fuels for LWRs has a long history. The U.S. Plutonium Utilization Program began in 1956, and was soon followed by related work in several European nations and Japan. The development effort was motivated in part by the potential for fuel-cycle economies perceived at the time and also as a means to reduce the consumption of uranium ore. It focused on the reprocessing of LWR discharge fuel to recover and recycle the plutonium and uranium. Several tests of partial core loadings of MOX fuel were conducted in U.S. LWRs during the 1960s and 1970s. Although plutonium recovered from LWR fuel was used in these tests, the results are generally applicable to MOX made from WPU.¹

In 1963 Belgium used a partial loading of MOX fuel in its BR-3 pressurized-water reactor (PWR). After many years of experimentation, by 1986 Belgium irradiated a core with a 70-percent MOX loading. Belgium provides a significant fraction of the world's currently operating MOX fabrication capability and is considering an expansion of its MOX fabrication plant. Its MOX fabrication services are marketed in conjunction with France by the MELOX consortium. Two Belgian LWRs are licensed to burn MOX fuel.

Germany tested and demonstrated MOX fuel in LWRs from 1968–1977 and began commercial use of MOX fuel in LWRs in 1981. Seven reactors in

¹ See Chapter 2 for a discussion of the differences between WPU and reactor plutonium (RPu). Details on the U.S. Plutonium Utilization Program are found in USNRC (1976).

Germany are using MOX, five others have a license to do so, and six more have submitted license applications (Thomas 1994, Wilcox 1994). A modest-scale MOX fuel fabrication plant at Hanau operated for several years before losing its license, and a larger plant was nearly completed before encountering licensing difficulties that have so far (late 1994) prevented it from operating.

France began fueling PWRs with MOX fuel on a commercial basis in 1985, and is building a substantial MOX fabrication facility. Sixteen French reactors are licensed to use MOX, and seven of them were doing so as of late 1994 (Nigon and Golinelli 1994). Japan is considering use of MOX fuel in roughly 10 LWRs by approximately the year 2000 (Yamano 1994) and is constructing a substantial MOX fabrication facility and a large commercial reprocessing plant for recycle of plutonium as MOX fuel. Britain, while not having a domestic MOX use program, is building a large MOX plant for foreign customers to complement the reprocessing services it already offers.²

Government-funded work on commercial fuel reprocessing and plutonium recycle in the United States was terminated by presidential directive in the mid-1970s, reflecting concerns that worldwide commercial plutonium recycle might stimulate the proliferation of nuclear weapons, and was also uneconomical (see Carter 1977). While the Reagan and Bush administrations did not take a similarly negative view, U.S. industry has concluded that the high cost of a commercially owned U.S. reprocessing plant makes near-term deployment of reprocessing and plutonium recycle uneconomical in the United States. For that reason, there are now no commercial facilities in the U.S. for reprocessing or for MOX fuel fabrication, and there are no commercial reactors in the United States licensed to use MOX fuel.

From the considerable world experience there is a mature technology adequate to implement the use of WPu as MOX fuel in LWRs. The simplest concept is once-through irradiation of MOX fuel to burnups similar to those used with uranium fuel in LWRs. No reprocessing would be required, and the rate-limiting step would probably be MOX fuel fabrication. Even a small subset of U.S. commercial LWRs would suffice to absorb the nominal 50 tons of excess WPu in this way at the WPu-MOX fuel fabrication rates likely to be attainable.

The following sections present specifics on the technical issues involved in using MOX fuel in LWRs and the rates at which various LWR configurations could process the WPu.

Technical Issues in LWR MOX Fueling

Early plans for MOX fueling in LWRs, in the mid-1960s to mid-1970s, were to reprocess all fuel discharged from a given reactor and to recycle all of

² As of late 1994, the target opening date for this plant was 1997 (Wilcox 1994). For more information on civilian plutonium fuel programs, see NAS (1994, Appendix B) and Albright et al. 1993.

the recovered plutonium as MOX fuel for subsequent reloads for that reactor. This was known as self-generated recycle. The neutronic characteristics of LWRs are such that the plutonium produced would be sufficient to fuel about a third of the reactor core when steady-state recycle was reached. The remainder of the core would continue to be fueled with LEU.

The one-third core loading was also thought at that time to be a practical limit for most of the LWRs that had been designed on the basis of uranium fueling. There are several reasons why the fraction of a reactor core that can utilize plutonium fuel without compromising safety margins may be limited, unless the reactor has been designed or modified specifically for plutonium fuel use: the lower delayed-neutron fraction in plutonium fuel as compared with uranium fuel, the higher average neutron energy in plutonium fuel, and the 0.3-electron volt neutron-absorption resonance of plutonium-239 combine to put extra demands on a reactor's reactivity-control systems; the more energetic neutron spectrum and higher gamma-ray flux in plutonium fuels increase radiation damage and thermal stresses in reactor internals; and the higher radioactive-decay energy of plutonium fuels puts greater demands on post-shutdown (including emergency) core-cooling capabilities (see Chapter 2; "Some Differences Between Plutonium- and Uranium-Based Fuels"). For these reasons, and because self-generated recycle could only provide enough plutonium for one-third of the cores of the reactors involved, the designs of most present LWR cores were not intended to offer the capability to burn plutonium fuel in more than one-third of the core.

For disposition of excess separated plutonium, however, using MOX in all of the reactor core rather than only one-third would be highly desirable, as it would reduce the number of reactors or the time required by a factor of three. This could substantially decrease the amount of transportation and the number of sites required to process the excess plutonium, reducing risks of theft during transport and reducing the political and licensing liabilities of involving more of the reactor industry in the plutonium-processing operation. There are also some technical incentives to move in the direction of full-MOX cores. Because the MOX fuel rods have a higher fission cross-section than do the all-uranium rods, the two different kinds of fuel rods must be carefully distributed within the reactor core to avoid local overheating. Also, the reactivity of MOX fuel tends to change more rapidly during an irradiation cycle than that of uranium fuel. There is greater spatial self-shielding of neutron flux in fuel assemblies composed of MOX fuel rods. Design problems are aggravated by the presence of both uranium and MOX fuel rods in the same core. Thus, a reactor core fueled entirely with MOX fuel would be easier to design and to program for fuel reloads.

During the 1970s the U.S. nuclear industry did envision an alternative to self-generated recycle of plutonium with one-third MOX cores. Some utilities planned to dedicate some of their LWRs to operate as "plutonium burners," to be fueled entirely with MOX fuel. An all-MOX plutonium burner would receive

make-up plutonium produced in other uranium-fueled LWRs. The all-MOX plutonium burner would be designed with the extra control absorbers needed for reactivity control, and other modifications would be needed to compensate for the increased fast-neutron and gamma fluxes, the increased decay heat, and the smaller delayed-neutron fraction.

Such a plutonium-burner PWR was designed by Combustion Engineering (Shapiro et al. 1977, Anderson and Klinetob 1981). Called the System-80, this reactor was designed for flexibility to use large plutonium loadings, up to and including a full loading of MOX fuel. As compared to a PWR designed only for uranium fueling, the System-80 plutonium burner has additional control rods and drives and increased boron concentration in the reactor coolant. The cooling systems for the reactor and spent fuel storage are sized for higher long-term decay heat. Core internals are designed to improve cooling and reduce thermal stress that would otherwise result from increased gamma-ray heating. The thickness of the core support barrel is increased to mitigate the increased fast-neutron flux.

Three System-80 reactors, each generating 1,300 megawatt electric (MWe) (3,750 megawatts-thermal; MWt), are in operation in the United States at the Palo Verde Nuclear Generating Station in Arizona. Four 1,000-MWe units are being constructed in Korea, including a forerunner of the evolutionary System-80+ design. One of the two uncompleted reactors of the Washington Public Power Supply System (WPPSS), designated WNP-3 (Washington Nuclear Project), is a System-80 reactor. This reactor is 75 percent complete and has been kept in mothball status.

In addition, recent studies by the major U.S. reactor vendors funded by the U.S. Department of Energy (DOE) have concluded that contrary to past expectation, many (though not all) existing LWRs (both PWRs and boiling-water reactors) could use WPu MOX in 100 percent of their reactor cores with little if any modification.³ Existing margin in the control capabilities of these reactors would, it is reported, allow them to operate within existing safety envelopes with 100-percent MOX cores.

In a number of the existing-reactor cases, however, the maximum safe enrichment of plutonium in the fuel would be lower than it would be in new reactors designed specifically for plutonium use, potentially increasing total fuel fabrication costs and the time required to carry out the plutonium disposition mission. It appears likely that existing reactors could handle higher plutonium loadings if modifications were made conceptually similar to those previously believed necessary to achieve full-core operation at all. This issue—including

³ ABB-CE (1994), GE (1994), and Westinghouse (1994). Indeed, it appears that full-core MOX loading is more easily achievable with WPu than it would be with RPu. The higher concentrations of higher isotopes of plutonium in RPu cause additional neutron absorption, further reducing control-rod worth compared to the WPu case (see ABB-CE 1994, Section 5).

the trade-offs between the costs and time required to modify reactors and the extra costs and longer times associated with using fuel of only modest enrichment—has not yet been studied in detail but should be.

The importance of these recent studies is considerable, as they suggest that the job of plutonium disposition in existing U.S. LWRs—one of the main options recommended in this report and the report of the parent committee—would be significantly easier than was believed at the time the parent committee's 1994 report was written. It should be noted, however, that these analyses became available late in the panel's deliberations, and the panel has not been able to review them in detail. The panel urges that they be reviewed by the U.S. Nuclear Regulatory Commission (NRC) to ensure that the conclusion that existing LWRs could use full cores of WPu MOX without compromising existing safety margins is correct.

Whatever the conclusion as to whether significant modifications would be necessary, it appears clear that existing U.S. LWRs could be adapted for full-MOX cores. Because of the considerable worldwide experience with MOX fueling in LWRs, the panel judges the technical uncertainty of the LWR MOX option—using either one-third or full-MOX cores—to be low (by comparison to other reactor options for use of plutonium fuels).

Reactor Throughput, Once-Through Fuel Cycle

The rate at which plutonium could be processed in a once-through MOX fuel cycle is given by the product of thermal power, capacity factor, fraction of the core that is MOX fuel, and plutonium weight fraction in fresh MOX fuel, divided by the average fuel exposure at discharge. Consider, for example, the reactor characteristics once contemplated for self-generated plutonium recycle (Shapiro et al. 1977, Hebel et al. 1978): assuming a 1,000-MWe PWR at 34.2-percent thermal efficiency, a capacity factor of 0.70, a one-third core of MOX fuel, a fuel exposure of 30,400 megawatt-days per metric ton of heavy metal (MWd/MTHM), and an initial concentration of 2.5 weight percent plutonium (Anderson and Klinetob 1981), the yearly amount of plutonium supplied as make-up fuel would be 205 kilograms (kg). Assuming a once-through fuel cycle, 50 tons of WPu could be processed by 8.1 such reactors given a nominal operating lifetime of 30 years.

As noted, using MOX in 100 percent of the reactor core, rather than only one-third, would reduce by threefold the number of reactor-years required to irradiate a given initial quantity of excess WPu. Full-MOX reactors comparable to the System-80 (that is, with a capacity somewhat larger than the 1,000 MWe assumed above), operated at 75-percent capacity factor with a 100-percent MOX core, with an initial plutonium content of 2.5 percent by weight and average burnup of 31,000 MWd/MTHM, would process 828 kg of WPu per reac-

tor-year, thus processing 50 tons in 60 reactor-years (for example, two reactors operating for 30 years) (Hebel et al. 1978).

The initial concentration of plutonium in the MOX reloads can be increased beyond 2.5 weight percent, particularly by adding more burble poisons. ABB-Combustion Engineering (ABB-CE), for example, has suggested that in their System-80+ system, a loading of 6.8 weight percent WPu, compensated by the addition of erbia to the fuel (which also has the advantage of smoothing reactivity over the fuel's life in the reactor), would be attainable. In the existing System-80 reactors, however, which have somewhat fewer control rods, ABB-CE estimates a maximum loading of 4.5 percent (ABB-CE 1994).

Estimates of the average enrichment of WPu that could be used safely in full-MOX cores in existing LWRs other than the System-80s vary. Westinghouse concludes that many existing PWRs could use full-MOX cores with an average enrichment of 4.5 percent, comparable to the enrichment possible in the System-80s. ABB-CE and General Electric conclude that their non-System-80 reactors could handle full-MOX cores with enrichments in the range of 3 percent (ABB-CE 1994, GE 1994, Westinghouse 1994). While no substantial reactor modifications are estimated to be required, the various vendors do suggest measures such as including burble absorbers in the fuel, dissolving increased quantities of boron in the cooling water, or changing the material used in the reactor control rods.

What do these figures mean for how fast the plutonium disposition mission could be accomplished? A 1,200-MWe PWR using a full-MOX core with a plutonium loading of 4.0 percent, and a burnup of 42,000 MWd/MTHM would use just under a ton of WPu each year, requiring roughly 50 reactor-years for disposition of the nominal 50 tons of excess material. If a System-80 reactor with modifications could operate with an initial plutonium content of 6.8 percent, and the same average burnup, this reactor would process about 1,700 kg of WPu per reactor-year, so that one such reactor operating for 30 years would suffice to process 50 tons (see Table 6-1).

As can be seen, the use of high plutonium loadings significantly increases the rate at which the WPu can be processed in a given reactor, even when the burnup is also increased. As described in detail in Chapter 6, moreover, use of such high enrichments reduces the net cost of the operation by reducing the number of kilograms of MOX fuel that need to be produced and increasing the energy value of each kilogram of that fuel.

The use of such high plutonium loadings, however, does create some technical issues, similar in some respects to those involved in moving from one-third to full-MOX cores. In general, burble absorbers must be added to hold down the reactivity. These absorbers (erbium oxide, in the case of the ABB-CE proposal) also help preserve negative temperature coefficients of reactivity for the moderator and fuel. Because of the resonance in the neutron absorption cross-section of erbium-167 at energies just above the thermal resonance of plutonium-239

(Pu-239), erbium can counteract the tendency of Pu-239 to contribute to positive reactivity coefficients. Other materials such as gadolinium, dysprosium, or boron may be equally effective in this role, depending on the specific application. Because the effectiveness of the burble absorbers decreases with increasing burnup, as the reactivity of the fuel declines, they contribute to smoothing out reactivity changes during the fuel cycle.

Higher plutonium-loading MOX fuel will also have a higher fissile plutonium content when removed from the reactor than would ordinary LEU spent fuel. This fact will require careful attention to long-term criticality issues in preparing such fuel for geologic disposal. (See discussion below and in Chapter 6.)

To summarize, the existing System-80 reactors were designed from the outset to handle full cores of plutonium fuel. Analyses by the vendors indicate that many other existing reactors could also use full cores of WPu fuel without substantial modifications to the reactors. Further analysis and review would be required to assess this conclusion. NRC review of any proposal to use plutonium fuels in U.S. reactors is likely to be intensive, and NRC review could lead to a requirement for reactor modifications. But even if significant modifications do turn out to be necessary, the panel believes that a variety of U.S. LWRs could be adapted to handle full-MOX cores safely, with sufficient enrichments to carry out the mission in a small number of reactors.

Fuel Fabrication

Providing adequate plutonium processing and MOX fuel fabrication capability would be an important pacing factor for processing excess WPu in U.S. LWRs.

Plutonium pits would have to be shipped from Pantex (located near Amarillo, Texas), where no plutonium processing capability yet exists, to a site capable of disassembling the pits and converting the resulting metal to plutonium oxide. No facilities for carrying out the pit-processing operation on the required scale are currently operating, but facilities at Savannah River (South Carolina), Los Alamos (New Mexico), Hanford (Washington), and possibly elsewhere could be modified for this purpose—and new technologies for efficient pit conversion are being developed at the national laboratories. Using a planning figure of 4 kg of plutonium per pit (NAS 1994), processing 50 tons of plutonium in 30 years would require a capability to process more than 400 pits per year. Processing the plutonium more quickly would require correspondingly larger capabilities.

There are no operating MOX fabrication facilities in the United States. LEU fuel fabrication facilities cannot readily be modified for this purpose because of the much higher radiotoxicity and safeguards requirements of plutonium fuel. Processing 50 tons of plutonium in 30 years, at a loading of 2.5 percent by weight plutonium in the fuel, would require a fabrication facility with a 67-

MTHM/yr capacity; at a loading of 6.8 weight percent, the required capacity would be 25 MTHM/yr.

There is an existing, nearly completed MOX fabrication facility at the federal government nuclear site in Hanford, Washington, known as the Fuel Materials Examination Facility (FMEF). This facility was built in the late 1970s and early 1980s to produce fuel for the liquid-metal Fast Flux Test Facility (FFTF). It was never operated. The panel has received estimates (which may be optimistic) that this facility could be modified to produce 50 tons of LWR fuel per year or more (containing roughly 3 tons of WPu, at a 6- to 7-percent enrichment) while meeting current safeguards and ES&H standards, within roughly five years of receiving a go-ahead, for a cost in the range of \$75–\$150 million.⁴ Funds would be needed to replace or upgrade existing, outdated process computers; upgrade older facility systems such as fire protection and waste handling; upgrade other facilities such as security and radiological protection to meet current federal, state, and DOE requirements; prepare safety analysis and safety and compliance documentation; modify the facility for LWR rather than liquid-metal reactor (LMR) fuel; and provide higher throughput than was originally planned.

Alternatively, a new plutonium fuel fabrication facility could be built. Estimates provided to the panel (which appear to be optimistic) indicate that such facilities could be built for between \$400 million and \$1.2 billion, depending on their capacity. Siting, designing, building, and licensing such a facility would probably require a decade or more. These cost and schedule issues are discussed in more detail in Chapter 6.

Reactor and Institutional Options

Many specific variants of MOX use in LWRs in the United States can be imagined. The reactors used could be existing, partly completed, or newly built for this purpose. Fuel fabrication could rely on partly completed or modifiable facilities or new ones. The relevant facilities could be government-owned and government-operated (GOGO); government-owned but contractor-operated (GOCO); owned and operated privately, with subsidies from the government to make the system competitive with other sources of power; or some mix of these.

As described in Chapter 3, in choosing among these variants, the nation should seek to minimize:

- security risks (which argues for minimizing the number of sites involved and the amount of transportation of plutonium in forms vulner-

⁴ See discussion in section “Economic Comparisons” in Chapter 6. Atomic Energy of Canada, Limited, the only vendor which has so far done a detailed study of possible use of the FMEF, concluded that adapting the facility to produce MOX for CANDU (Canadian deuterium-uranium) reactors would cost \$118 million (AECL 1994).

- able to theft, and maximizing the use of government sites, where requisite security either already exists or is simpler to provide);
- costs and delays, both in construction and in gaining requisite licenses and approvals (which argues for making maximum use of existing or partly completed facilities, and again for use of government sites, where the licensing and public approval processes may be somewhat simpler); and
- risks to environment, safety, and health (which argues for making use of the safest and best designed facilities likely to be available for the purpose, and for choosing the scale and types of nuclear materials processing so as to minimize risks and waste streams).

There are no options that perfectly meet these criteria. The following are a few of the most obvious specific candidates for this role:

Currently Operating, Utility-Owned Reactors: As noted earlier, three System-80 LWRs are operational at the Palo Verde site in Arizona, which could operate with full-MOX cores without modification, with license amendments.⁵ The same may be possible with a variety of other operational utility-owned LWRs in the United States. In one possible approach, for example, if the utility agreed to participate, the federal government would cover any additional costs in using government-furnished MOX fuel and would provide the necessary additional safeguards and security at the site, while the utility would otherwise continue to operate the reactors much as they are operated today. Additional financial incentives might be required to convince the utility to undertake the additional political and licensing burdens involved. As several sites have more than one reactor on-site, the handling of “fresh” plutonium and MOX fuel could be limited to two sites—one where the MOX fuel would be fabricated (presumably a site within the nuclear weapons complex) and the reactor site. Exploration of utility and public reactions to this concept is still in its early stages, though more than one utility has privately expressed interest to DOE.

The Washington Nuclear Project Reactors: Reactor 3 of the Washington Nuclear Project (WNP-3) in Washington state is a System-80 reactor, 75-percent complete, in the western part of the state, roughly 150 miles from the Hanford nuclear weapons complex reservation. WNP-2, complete and operating, is not a System-80, but may also be capable of handling a full core of MOX fuel without major modifications. It has the advantages of being complete, licensed, and located on the federal government’s Hanford site, where the FMEF fuel fabrication facility is also located. The uncompleted WNP-1, like WNP-2, is not a System-80 but is located on the Hanford site. One or two of the three WNP reac-

⁵ It should be noted that the Palo Verde reactors have been experiencing steam generator problems, and one of the units, Palo Verde 1, has a poor lifetime load-factor record. See *Nucleonics Week* (1994), and NEI (1993), both cited in Lyman (1994).

tors could be acquired, completed (in the case of WNP-1 and WNP-3), and operated by the federal government (possibly in cooperation with a private entity) for the plutonium disposition mission. (For a discussion of the costs involved, see Chapter 6.) If the MOX fabrication capability at Hanford were used, this would have the significant advantage of confining all plutonium handling to two federal sites in the same state (or even a single large site, if only the WNP-1 and WNP-2 facilities on the Hanford reservation were used).

Two groups of private companies have put forward proposals for a government/private partnership to pursue this approach.⁶ If these options are to be preserved, action will have to be taken soon, as the WPPSS board has recently voted to cease maintaining WNP-1 and WNP-3 and to offer the components for sale.

Construction work on these reactors was halted some years ago. If these reactors were to be completed for MOX use, an intensive NRC review would be expected, which might result in requirements to upgrade some reactor systems to make them more comparable to the new reactors whose designs are now being reviewed by the NRC. Such modifications, if required, would add to the cost of completion.

DOE is also considering other options involving completion of partly-completed reactor facilities for the plutonium disposition mission, such as use of three Tennessee Valley Authority reactors on which construction has been

⁶ One team, consisting of Battelle, Science Applications International Corporation, and Newport News, calls itself the "Isaiah Project" (after the biblical prophet who admonished the world to beat swords into plowshares). In their proposal, the private consortium they would set up would acquire and complete WNP-1 and WNP-3 at its own expense (deeding ownership of the reactors to the government) and receive revenue to pay for debt service and profit. The government would pay for reactor operations, fuel fabrication, storage, and disposal, and provide a contractual guarantee of particular quantities of steam for electricity production. Advocates for this concept have emphasized the possibility that the private entity could borrow several billion dollars against the future revenues of the project, which could be provided to the government to finance other endeavors, such as assistance for plutonium disposition in Russia. This is misleading, however, as future costs assigned to the government in this concept would come to substantially more than the sums that could be borrowed. Hence, as with other approaches, the project would involve a substantial net discounted present cost to the government, not net discounted present value. Borrowing against future revenue, with the accompanying promise of large future government expenditures, would simply amount to deficit financing by other means. This point is equally applicable to other approaches involving private financing of initial capital costs in return for government promises of later subsidies.

In response to the Isaiah Project proposal, WPPSS, which built these reactors, has proposed a different concept that would use WNP-1 and WNP-2, rather than WNP-1 and WNP-3. This would keep all the operations on the Hanford reservation, which would have security advantages (because of reduced plutonium transport) and political advantages (because of the more favorable climate near Hanford than in the western part of the state). WNP-2, in the initial WPPSS proposal, would operate with only a one-third MOX core, but since it is already operating, and limited quantities of fuel might be produced in existing facilities at Los Alamos or elsewhere, it could serve as a test-bed, and begin plutonium operations before a full-MOX fabrication facility became available.

halted, but these would not have the advantage of co-locating fuel fabrication and reactor facilities at an existing government-owned nuclear site.

Acquisition of Existing LWRs: If the use of MOX fuel in utility-owned reactors or the WNP reactors faced insurmountable licensing or approval difficulties, another option is for the government to acquire existing reactors. There are several U.S. reactors that utilities may be willing to provide to the government, either because they were never completed or because their continued operation is becoming economically uncompetitive.⁷ These could be acquired, modified if necessary for full-MOX operation, and used much as the WNP reactors might be. Such an acquisition would be unprecedented, however, and could raise a substantial set of procedural and institutional questions that could take time to resolve. No reactor operating license, for example, has ever been transferred from a utility to the federal government.

Principles for Institutional Arrangements: The specifics of such institutional arrangements require further study, but several basic principles suggest themselves:

- (1) the government should have a strong role to ensure that the approach fits with broader national policies relating to arms reduction and non-proliferation, that adequate security and safeguards are maintained, that any necessary openness to international inspection is maintained, and that appropriate ES&H standards are met;
- (2) the number of sites should be minimized to consolidate monitoring and safeguards functions and reduce the risks of plutonium theft;
- (3) if the sites were federal facilities (either already owned by the government or acquired for this purpose), it could ease the task of gaining the necessary approvals and licenses and maintaining the security and international transparency mentioned above;
- (4) any increase in government competition with private electricity generation should be minimized to the extent possible; and
- (5) if private investment can genuinely reduce government costs and up-front federal capital investments, it should be encouraged. But assessments of such possibilities must include realistic appraisals of all likely future costs and revenues, including the financial risks of government commitments to future subsidies or operations.

⁷ By some estimates, more than a dozen reactors in the United States may be shut down well short of their design lives because the utilities that own them have other, more economical generation alternatives available. These reactors would be prime candidates for acquisition by the government for the plutonium disposition mission.

Approvals and Licenses

In addition to fuel fabrication, approvals and licenses will be important pacing factors for any use of plutonium fuels in LWRs in the United States. The United States initiated a licensing process for using MOX in LWRs in the 1970s (the Generic Environmental Statement on Mixed Oxide fuel, or GESMO) and obtained initial NRC staff approval, but this process was terminated when President Carter decided to end government support for the plutonium fuel cycle. While there do not appear to be fundamental obstacles to licensing U.S. reactors to handle MOX, the time to achieve the requisite fuel fabrication and reactor licenses is uncertain and might range from a few years to a decade, depending on the choice of reactors and sites as discussed above. Substantial public controversy could attend siting and construction of a plutonium fuel fabrication facility and use of plutonium fuel in U.S. reactors. There are important open questions concerning the licensing process for the various plutonium disposition facilities, and in particular whether oversight would be conducted by the NRC or the Defense Nuclear Facilities Safety Board (DNFSB). (See the appendix to Chapter 6 for more detail on the licensing issue.)

Public approval in the areas near the relevant facilities will also be a critical factor. Problems of public approval and licensing could be lessened somewhat if both the fuel fabrication facilities and the reactors handling MOX fuels were on federal sites. This is the main argument for building new reactors at existing DOE sites, rather than relying on existing civilian reactors. There is a good chance, however, that these problems could be addressed at some existing reactors.

Safeguards, Security, and Recoverability

The discussion to this point has focused primarily on feasibility and timing. Another important criterion identified in Chapter 3 is safeguards and security during the process. The panel believes that the United States and Russia should agree on a stringent standard of safeguards and security to be maintained throughout the disposition process. As described in earlier chapters, the panel recommends that safeguards and security for WPu be designed to come as close as practical to meeting the “stored weapons standard” throughout the disposition process.

An important issue is the risk of theft of materials during transport. Given the stringent security procedures and the low incidence of terrorism in the United States, this risk is likely to be greatly lower in the United States than it would be in Russia under current circumstances. (In both countries, however, it should be noted that thousands of assembled nuclear weapons, as well as other nuclear materials, are currently being transported each year as part of ongoing dismantlement programs and other activities.) The scale of transport required for disposition of excess WPu will depend to a great degree on the number of sites

involved—in particular, whether conversion of pits to oxides, fabrication of fuel, and the relevant reactors would all be at the same site or at several widely dispersed locations. The number of sites where this plutonium is handled, and the shipments of plutonium and their length, should be minimized to the extent possible to limit the risks of theft.

Once the plutonium is in the form of bulk oxide, rather than individually packaged pits, precise accounting to detect any diversion will become considerably more difficult. This will be a particular problem at the fuel fabrication facility, where the accounting system will need to have the capability for timely detection of diversion or theft of even a very small percentage of the facility's throughput. The International Atomic Energy Agency (IAEA) and EURATOM (the European Community's nuclear agency) (with assistance from the Los Alamos National Laboratory) have been working for years to develop new techniques for safeguarding such large plutonium bulk-handling facilities, as similar large facilities for civilian plutonium processing are scheduled to open soon in Europe and Japan. Nevertheless, some of these techniques are still in development, and it is doubtful that material accounting alone will be able to guarantee that diversion of enough plutonium to make a bomb could be detected within days. It will probably not be possible to achieve the "stored weapons standard" of accounting (described in Chapter 3, "Criteria for Choice") when dealing with complex, multistage processing of plutonium in bulk form. Therefore in addition to stringent material accounting, there should be extensive containment, surveillance, and security measures to ensure that no plutonium leaves the site without authorization.

Plutonium imbedded in the MOX spent fuel would be roughly as difficult to recover as the much larger and growing quantity of plutonium in commercial spent fuel worldwide, and this option therefore meets the "spent fuel standard." The MOX spent fuel, however, would have higher plutonium concentrations, so that a smaller total amount of MOX spent fuel would have to be processed to acquire enough material for a weapon.

Cost

As described in Chapter 3, in the case of reactors that would operate regardless of the mission of WPu disposition, the cost of using excess WPu in existing LWRs should be reckoned as the net additional cost compared to the cost of producing the same electricity by the means these reactors would otherwise use—that is, with LEU. As described in detail in Chapter 6, the required subsidy for using MOX fabricated from WPu rather than LEU in existing LWRs is likely to range from several hundred million to more than a billion dollars. If reactors had to be built, completed, or modified, or if the differences between LEU and MOX spent fuel involved higher disposal costs for MOX, these expenses would have to be added to this figure. In the case of the reactors that would not other-

wise operate, one must compare the cost of generating electricity in this way to the cost of generating it by the means that would otherwise be used—in most cases in the United States, either coal or combined-cycle natural gas. In effect, in these cases a twofold subsidy would have to be paid: the additional cost of using nuclear reactors rather than the least-cost means of electricity generation, and the net additional cost of using plutonium-based fuels rather than LEU in those reactors.

Environment, Safety, and Health

With appropriate modifications, the panel believes that it should be possible to operate U.S. LWRs with full-MOX cores while meeting the same safety standards that pertain to LEU fuel. Detailed safety analyses will be required, however. The plutonium processing necessary for this option (pit conversion and fuel fabrication) would inevitably result in wastes, risks of accident, and worker hazards. Careful design and the application of sufficient resources, however, should enable these facilities to comply with current regulatory standards. Commercial MOX operations are now routine in Europe and were carried out in the 1970s in the United States by several companies, but have not yet been undertaken in the current U.S. regulatory environment.

The spent fuel resulting from this option would be similar in most respects to ordinary LEU spent fuel, but there are differences.⁸ MOX spent fuel will contain more plutonium than typical spent fuel (raising potentially greater criticality concerns after eventual emplacement in a geologic repository) and will emit more heat for a longer time (which has an impact on the repository volume needed to hold a given number of fuel assemblies). The possibility that the somewhat different chemistry of the MOX spent fuel would affect long-term rates of release of radioactive materials in the repository would also have to be examined. This different spent fuel would have to be separately licensed as an acceptable waste form for geologic disposal, meaning additional costs and potentially additional delays. If the repository criticality issues are effectively addressed, however, it should be possible to store and dispose of MOX spent fuel as safely as LEU spent fuel. If the reactors used for this purpose would have operated with LEU in any case, the total amount of spent fuel to be disposed of in a geologic repository would not be increased as a result of plutonium disposition; even if reactors were operated specifically for plutonium disposition, the total amount of added spent fuel would be a small fraction of the planned capacity of the first U.S. repository. (See Chapter 6 for a more detailed discussion.)

⁸ Like LEU spent fuel, this spent fuel would be stored for a period of time and then placed in a geologic repository. How long it will be before a U.S. repository is available remains uncertain.

Other Issues

As noted in Chapter 3, policy-makers considering plutonium disposition options should be aware that use of U.S. WPu in U.S. LWRs could be perceived by some as a significant change in U.S. policy, which has been not to pursue a plutonium fuel cycle, in part because of proliferation concerns. Such a perceived shift could have an impact on decisions on civil plutonium policies in Europe, Japan, and elsewhere. If the U.S. government wished to limit this effect and maintain its current policy of seeking to minimize worldwide stockpiles of separated, weapon-usable fissile material, it could make it clear that reducing the security dangers posed by existing stocks of excess WPu by using this material as reactor fuel, once-through, was perfectly consistent with such a policy. We note that since the WPu is already separated, the choice of a reactor option would not necessarily reopen the contentious question of reprocessing in the United States. All of the "elimination" options would require reprocessing and recycle, which could raise safeguards and security concerns, and would raise contentious political issues in the United States; such an approach would require a modification of current administration policy, which is not to reprocess plutonium for civilian energy generation.⁹ (See discussion in Chapter 6, "General Considerations.")

The "Spiking" Option

Another option making use of LWRs would be to "spike" the fuel by brief irradiation, processing the plutonium more rapidly, but creating a much less substantial radiation barrier. Such spiking would require a larger fuel fabrication facility (implying a higher capital cost) and more frequent reactor shutdowns for refueling (implying a lower capacity factor and more lost revenue).¹⁰ Expanded fuel storage capacities at the reactor sites would also be required to handle the fuel between the time when it was spiked and when it was recycled into the reactor to finish burning it to "spent" fuel. Hence, the costs of the spent fuel option would increase significantly if "spiking" were used as a first step. In the panel's judgment, the security for the material that could be gained by this more rapid but less extensive irradiation could be achieved more simply by providing appropriate security at the plutonium storage site; given its substantial costs, the spiking step on the path to the spent fuel option in LWRs is probably not worthwhile.

⁹ In considering the discussions of the various elimination options throughout this chapter, it should be remembered that all of them would raise the issue of reprocessing and recycle.

¹⁰ Most spiking approaches envision achieving large increases in plutonium throughput by using irradiations much lower than those that would occur in a year of exposure in a commercial LWR. This means that the plutonium-bearing fuel would need to be changed out more frequently than the once per year or once per eighteen months common in commercial LWRs.

The “Elimination” Option

If excess WPu is used to fuel reactors on a once-through fuel cycle, the resulting spent fuel will add a small increment to the very large and growing amounts of plutonium in spent fuel around the world. Some analysts have advocated going farther, “beyond the spent fuel standard,” seeking to minimize or eliminate this small increment to the existing spent fuel problem. Given the extra costs, delays, and risks involved in such an effort (described below), the panel does not recommend undertaking such a program for the narrow purpose of eliminating WPu. The broader issue of eliminating both WPu and the global stock of RPu can only be considered in the broader context of the future of nuclear power, which is beyond the scope of our analysis here. Nevertheless, we describe below some of the issues involved in such elimination approaches, in this case using LWRs.

Two possible definitions of “elimination” in reactors present themselves. On the one hand, one could seek to fission this particular plutonium as completely as possible. Plutonium would be loaded into reactors and burned; residual plutonium would be separated from the spent fuel, fabricated into new fuel, and burned in reactors again. Any reactor fuel cycle with reprocessing and recycle, and a conversion ratio less than unity, can obtain a high degree of elimination of an initial quantity of plutonium in this way.¹¹ Complete elimination, however, is not possible, as some of the material will inevitably be lost to waste in the course of reprocessing and recycle.

Another, less demanding approach is to seek to eliminate the added increment of plutonium in spent fuel resulting from WPu disposition. If the WPu were used to fuel reactors that would otherwise have operated with LEU (and would have produced plutonium in the process), then when the excess WPu had been consumed to the point that the remainder was equal to the amount of plutonium whose production had been avoided by fueling these reactors with plutonium rather than LEU, the plutonium can be said to have been effectively eliminated. At that point, there would be no more plutonium in the world than

¹¹ Within the possible choices there are no fundamental differences in the degree of annihilation that can be obtained, although there are differences in the time to achieve a given extent of annihilation. While only fast-neutron reactors can fission all actinide isotopes, reactors with a thermal-neutron spectrum, such as LWRs, can in principle transmute most of those isotopes they cannot fission into other isotopes they can, as part of their normal operations. DOE-sponsored analyses, while considering reprocessing for other reactor types such as liquid-metal reactors, have considered only once-through MOX fueling for LWRs for the plutonium disposition mission. LWRs, however, could be used in a reprocessing and recycle mode as an elimination option—though using the traditional PUREX (plutonium and uranium recovery by extraction) reprocessing approach with its recovery of pure plutonium oxide would raise greater security issues than some of the other recycling approaches. For a more extended discussion of eliminating actinides in spent fuel by repeated reprocessing and recycle in various reactor types, see National Research Council (forthcoming).

there would have been if the excess WPu had never existed, and any added security risk from the excess WPu would certainly have been eliminated. The same amount of electricity and the same mass of fission products would have been produced as would have been produced in the absence of the WPu. The only difference would be that some uranium would remain that would have been consumed had the excess WPu not been available as a substitute. In most cases, achieving such "effective elimination" would also require some degree of reprocessing and recycle. We consider "elimination" by these two definitions in the following paragraphs.¹²

Elimination by Fission

Flowsheets for a 1,000-MWe version of a System-80 reactor operating as a plutonium burner with plutonium recycle have been presented, assuming plutonium make-up is obtained by reprocessing uranium fuel discharged from LWRs.¹³ The system would be operated by fabricating excess WPu into fuel, burning it in the reactor, separating the residual plutonium from the spent fuel, fabricating that into fuel, and burning it in the reactor again. As the cycles continued, absorption of neutrons in the plutonium would cause the amount of higher isotopes of plutonium and higher actinides to increase, eventually reaching a steady state. This increase would substantially complicate reprocessing and fuel fabrication because of the higher radioactivity of the remaining actinides. It would also affect the reactivity of the remaining actinides, as the fission and absorption cross-sections of the various actinides vary considerably. (Indeed, no more than a couple of cycles have been accomplished in LWRs to date.) Assuming a capacity factor of 0.70, in the steady state such a reactor would consume approximately 0.50 tons of plutonium per year.

This does not mean that 50 tons of plutonium could be completely destroyed in 100 reactor-years. Such a linear destruction path would last only as long as there was enough plutonium remaining to keep the plutonium-burning reactors fully fueled. Once the amount of plutonium was reduced below the amount required to sustain a single reactor, additional fissile material (such as enriched uranium fuel in part of the reactor) would have to be added, to allow continued recycling for further destruction of the plutonium. The portion of the reactor devoted to MOX fuel would shrink with each irradiation cycle, until finally the desired degree of plutonium burnup was obtained. The destruction of

¹² In particular, it can be shown that if the plutonium disposition reactor discharges more plutonium in spent fuel than an LEU-fueled reactor of equivalent power would have produced, reprocessing and recycle will be necessary to effectively eliminate the added increment of plutonium (see Garwin 1995).

¹³ Hebel et al. (1978). Because the lifetime-average composition of plutonium in the reactor is dominated by the recycled plutonium, those results can be conservatively applied for a system using WPu as make-up fuel.

the residual plutonium in such a system is proportional to the product of the neutron flux and the amount of plutonium remaining in the core, so the destruction of the residual plutonium is like exponential decay. A finite time can be reported only if the extent of depletion is specified (see, e.g., Hebel et al. 1978, Choi and Pigford 1993).

A campaign to eliminate, for example, 99 percent of the WPu using LWRs would take many decades and would involve substantial costs and risks. No facilities designed for reprocessing LWR spent fuel are currently operational in the United States. Reopening of closed facilities or construction of new ones in the United States would be costly and time-consuming, and it would be expected to be the subject of considerable political controversy and intense regulatory scrutiny.¹⁴ Alternatively, reprocessing in other countries could be considered. Facilities capable of processing and fabricating multiple recycle plutonium (which is far more radioactive than WPu) would be needed, and technologies for that purpose would have to be demonstrated and licensed. The costs of repeated reprocessing and refabrication, and the costs of fuel development and licensing for multiple recycle fuel, would add a substantial increment (in the range of billions of dollars) to the subsidy required for the once-through plutonium disposition approach.

If the traditional PUREX process for separating plutonium from LWR fuel were used, such an elimination campaign could have the effect of increasing rather than decreasing net proliferation risk, as repeated and extensive handling of fully separated weapon-usable plutonium would be required. (The steady-state material that would eventually exist, however, with its high admixture of higher plutonium isotopes and other actinides, would be even less attractive for use in weapons than ordinary reactor-grade plutonium.) It is possible that some of the reprocessing approaches that have been suggested for other reactor types, which do not fully separate the plutonium, could be adapted to LWR fueling, but such approaches would have to be developed.

The net rate of plutonium destruction would be increased if additional plutonium was not being produced by absorption of neutrons in the uranium in the MOX fuel. Uranium-free "nonfertile" fuels for LWRs have been proposed for this purpose, containing plutonium, an inert material such as zirconium, and burble absorbers such as erbium (see ABB-CE 1993, pp. III-51 ff.; and INEL 1993c). Use of such nonfertile fuels in LWRs has not been demonstrated on any substantial scale, and a substantial fuel development program would be required, involving significant cost and time. A variety of safety issues would require careful examination, and licensing of the new fuel could be expected to be difficult and time-consuming.

¹⁴ Facilities for aqueous reprocessing of LWR spent fuel, however, could be built more quickly and with less technical and cost uncertainty than could reprocessing facilities of any other type (given the extensive global experience with this reprocessing approach).

The advantage offered by nonfertile fuels for a plutonium elimination campaign may be less than is commonly thought. This is because simply increasing the plutonium concentration in ordinary (i.e., fertile) fuels would substantially reduce the amount of new plutonium produced for each unit of WPu burned or energy generated. For example, if the power level is to be kept constant, doubling the plutonium concentration (for example, from 3.5 to 7 percent of heavy metal atoms in MOX fuel for an LWR) requires, for a constant power, the same number of total fission per second.¹⁵ The rate of production of fission-spectrum neutrons is the same (for any given nominal reactor power), while the flux of fission-inducing neutrons is reduced, in order to keep the same integral of plutonium fission cross-section multiplied by flux. While the near-thermal ("epithermal") neutron flux would not be reduced by as much as a factor of two by such a doubling of plutonium concentration, it would surely be reduced, and the production of Pu-239 (via neutron capture by uranium-238 [U-238], followed by two beta decays) will be less for a given reactor power than with the lower WPu loading. This effect is slightly reinforced by the displacement of some fertile U-238 by plutonium, in consequence of the higher plutonium loading. Higher initial plutonium loadings would reduce the rate of new production still further. Since, in addition, even nonfertile fuels cannot burn their plutonium content down to zero (because at low enough concentrations of plutonium in relation to neutron-absorbing fission products, a chain reaction can no longer be sustained), it seems unlikely that the development of such fuels for reactors not already designed to use them could provide an advantage big enough to justify the required level of effort.

"Effective Elimination"

Any "effective elimination" campaign would use a reactor and fuel that would destroy a certain number of kilograms of plutonium per year. If we scale to 1,200 MWe, a comparable LEU-fueled LWR produces 253 kg of plutonium per year (see Table 6-1). Thus, for each year that a plutonium-burning reactor operates instead of a comparable LEU-fueled LWR, the amount of plutonium that otherwise would have existed in the world is reduced by the net plutonium consumption of the reactor, plus 253 kg. To "effectively eliminate" 50 tons of excess WPu would require running enough plutonium reactors long enough for this annual sum to add up to 50 tons.

To accomplish this objective using one 1,200-MWe LWR with a 100-percent MOX core would require over five decades of reactor operations, with two to three phases of separating the plutonium from the spent fuel and recy-

¹⁵ The added reactivity is compensated by additional control absorbers and neutron poisons, and the control rods are reinserted on reactor startup when the nominal power output level is reached.

cling it (see Garwin 1995). For an LWR burning nonfertile fuel, the equivalent time is nearly four decades. Counting only current market prices for reprocessing and MOX fuel fabrication (not the substantial costs of development and licensing, building a new reprocessing facility if such were done in the United States, or the increased reprocessing and fabrication costs likely to be associated with multiple recycle material), "effective elimination" with LWRs using MOX would cost some \$2.3 billion more than the once-through spent fuel option with similar reactors. (The times and costs for elimination by fission, as described above, would be substantially higher, as more cycles would be necessary.) All of the remarks above concerning required development, licensing and political issues, and security concerns would apply to this case as well.

RUSSIAN PLUTONIUM IN CURRENT-GENERATION RUSSIAN THERMAL REACTORS

Description of Technology and Status

The major differences in the case of using Russian LWRs to process Russian excess WPu include much higher security risks in the disposition process because of the current economic and political upheavals in the former Soviet Union; much lower availability of funds to finance the process; a smaller existing infrastructure of safe reactors; less experience in LWR MOX operations; and different economic conditions, plutonium fuel policies, and licensing procedures.

Russia has two major classes of thermal reactors, the VVER series LWRs and the RBMK graphite-moderated reactors. The VVER reactors use a conceptual approach similar to that of U.S. PWRs. Thus, much of the discussion in the first section of this chapter of the capability of existing LWRs for disposition of excess WPu is applicable to the existing VVER reactors and does not require repetition here.

VVERs have been built at two nominal power outputs: 400 and 950 MWe. Twenty-five VVERs are currently operable in the former Soviet Union (*Nuclear News* 1993). There are six nominally 400-MWe units in Russia and two in Ukraine. These reactors do not have containments, a major difference in safety from international standards. The early models (VVER 440-230) were not designed to withstand major earthquakes or the level of cooling water losses which Western reactors are designed to survive, have less redundancy in their safety systems, lack emergency operating procedures and training simulators to assist operation in responding to upset conditions, and otherwise fall far short of internationally accepted safety standards, such as those of the IAEA (IAEA 1992a). As a result, some of the VVER 440-230s have been shut down (in Russia and Armenia and also in eastern Germany). The later models (VVER 440-213) have incorporated some improved safety features, but still lack many key safety sys-

tems and other features and remain far from internationally accepted standards (IAEA 1992b). All the VVER-440 designs have more thermal operating margin than Western reactors, however, which has contributed to a good availability record. Since the operable VVER-440 units in the former Soviet Union have a small capacity, are old, and are substandard from a safety standpoint, the panel does not believe they should be used for the WPu disposition mission.

Seven nominally 950-MWe VVERs (model VVER 1000-320, usually referred to simply as the VVER-1000) are currently operable in Russia and ten in Ukraine. Twelve more are under construction in Russia and six in Ukraine, though construction is continuing on only a few of these. These larger VVERs have been provided with containments and have moved closer to Western designs both in increased safety and reduced thermal operating margins. The VVER-1000 reactors do not currently meet international safety standards, principally because they require thorough analysis and upgrading of (1) their instrumentation and control systems, (2) their use of redundancy to enhance safety margins, and (3) their emergency operating procedures and operator-training regimes. Nevertheless, the consensus of foreign experts is that, with the upgrades that are expected to be implemented over the next several years, they will be adequately safe. In any event, the Russian and Ukrainian governments are expected to operate these reactors for the long term.

An international program is underway to improve the safety of Soviet-designed reactors, but because the safety of the VVER-1000 reactors is so much better than that of the RBMKs and VVER-440s, the principal international safety effort today is directed at resolving the safety problems with these latter reactors except for upgrading emergency operating procedures and training; little work has been done on improving the safety of VVER-1000 reactors in the former Soviet Union, although there are plans for upgrading the plant systems in the future (EBRD 1993).

The capability of VVER-1000s to process WPu in the form of MOX fuel is likely to be similar to that of current-generation U.S. LWRs. None of the VVERs, however, have yet operated with MOX fuel, and the capability of each operating reactor in this regard is yet to be confirmed. Representatives of the Russian Ministry of Atomic Energy (MINATOM) confirm that studies of MOX use in VVER-1000s are just beginning. Preliminary Russian analyses have concluded that one-third MOX cores could be accommodated in VVER-1000s with little change in the reactors. Analyses of full-MOX cores are just beginning; as in other LWRs, use of plutonium fuel would reduce control-rod worth and the delayed-neutron fraction, which could require some modifications to ensure sufficient control capability (Levina et al. 1994, Novikov et al. 1994). More analysis is required to determine what measures would be necessary to ensure safety in an all-MOX VVER-1000, and in particular whether the VVER-1000s also have the specific design features that are reported to give many U.S.-

designed reactors enough control margin to operate with full-MOX cores of WPu without substantial modification.

The panel's general conclusion, which requires further analysis to confirm but is based on our knowledge of Western PWRs of similar core design, is that with core-reactivity-control backfits as necessary, VVER-1000s could be operated with full-MOX cores of WPu without eroding existing safety margins. If it is determined that significant reactor modifications would be required in the VVER-1000 case, it should be noted that each operational VVER-1000 is scheduled to be shut down for roughly one year for safety improvements under the ongoing program of international safety assistance. With enough lead-time for proper design and preparation, modifications that might be necessary to handle full-MOX cores could be made during this period, without substantially extending the length of the shutdown. Alternatively, VVER-1000s scheduled for completion in the near future could be modified for this purpose as they are completed.

Use of MOX in LWRs is not currently the Russian government's favored approach to dealing with excess WPu. Because of the delays in commercializing fast-breeder reactors that would consume plutonium separated by reprocessing, all other major reprocessing countries except Britain have decided to use plutonium as MOX in LWRs to limit the buildup of large stores of separated plutonium. Russia has not yet taken this route, preferring to save both military and civilian separated plutonium for eventual use in breeder reactors (see section "Advanced Liquid-Metal Reactors" below). Russia already has some 30 tons of civilian separated plutonium in storage, and more is building up every year, in addition to the excess military plutonium resulting from arms reductions.

Some representatives of MINATOM, in their advocacy of using WPu in fast reactors, have suggested that because Russia does not yet have experience with MOX in LWRs, the fast-reactor option could in fact be accomplished more quickly than the LWR MOX option.¹⁶ There are increasing indications, however, that MINATOM officials are beginning to look more favorably on the option of using plutonium fuel in LWRs that exist today, to deal with the increasing plutonium surplus in the near term.¹⁷ While Russia does not have direct experience of LWR MOX use, it appears that Soviet-designed LWRs, like their

¹⁶ Mikhailov et al. (1994). Evgeniy Kudriavtsev of MINATOM, for example, reported to the IAEA in April 1993 that in Russia there have been no major investigations of the use of WPu in VVERs, though he indicated that a future facility for fabricating MOX fuel for VVER-1000s is planned (Kudriavtsev 1993, see also Bibilashvili and Reshetnikov 1993).

¹⁷ Egorov et al. (1994, p. 2), for example, indicate that while "for a long time in Russia, much longer than in other countries, plutonium was meant to be used solely in fast breeder reactors," now "work is underway to study the fabrication technology of experimental subassemblies with MOX fuel for the VVER-1000," and a critical assembly to test such fuels is being built at Obninsk. The increasing surplus of plutonium, Egorov et al. (p. 3) note, at a time "when intensive deployment of BN type [fast] reactors in the nearest future has a low probability," requires "corrections to be made in the earlier existing strategy of plutonium utilization mainly in fast reactors."

foreign counterparts, could be prepared for MOX use on the basis of experience in other countries. And, as described below, under current economic conditions it is likely to be quite some time before sufficient funds to build the fast reactors MINATOM envisions become available.

The former Soviet Union also has RBMK reactors. Although their reactor coolant is light water, RBMK reactors are substantially different from the LWRs discussed above and are not classified as such. The RBMK reactors are of a design unique among the world's present commercial electric-generating nuclear plants. The core consists of a large block of graphite moderator interspersed with pressure tubes containing low-enriched UO_2 fuel, through which light water is pumped to remove the heat from the rods. The reactor can be refueled continually while at power. The design is conceptually similar to U.S. plutonium production reactors and resembles the Hanford N Reactor, which was the only U.S. plutonium producer that also generated electricity. The N Reactor has been shut down permanently.

RBMKs have been built at two nominal power outputs: 925 and 1,380 MWe. Fifteen RBMKs are currently operable. Thirteen of them are nominally 925-MWe units—eleven in Russia and two in Ukraine (Chernobyl). An additional 925-MWe unit is under construction in Russia. There are two 1,380-MWe units, both located in Lithuania. The RBMK designs are further below international safety standards than the VVER designs. Reactor containment is not provided, there is limited protection from loss-of-coolant accidents, and the void coefficients of reactivity lead to core instability at low power levels. It was this type of reactor that suffered the disastrous accident at Chernobyl.

This panel has not evaluated the capability of the RBMKs to process WPu , although the reactor probably can operate with partial plutonium fuel loads with suitable modification for reactivity control. Because of the accident at Chernobyl, there has been substantial pressure, from within and without the former Soviet Union, to shut all the RBMKs down. The governments of Russia, Ukraine, and Lithuania have stated that the ensuing power shortages would threaten public safety more than continued operation of the plants, however, and have declared their intention to keep operating most of them. Debates on this subject continue. Efforts have been initiated by Western countries to assist Russia, Ukraine, and Lithuania in improving the safety of the RBMK and VVER 440-230. It is generally acknowledged, however, that it will not be practical to reach a level of safety conforming to international standards, and therefore, there is a desire to shut these plants down as soon as sufficient alternate capacity can be made available. The panel does not believe that the RBMKs should be considered for the plutonium disposition mission.

Reactor Throughput, Once-Through Fuel Cycle

If full-MOX cores proved acceptably safe, with enrichments of perhaps 5 percent plutonium in the fuel, two VVER-1000 reactors could transform 50 tons of WPu into spent fuel in 30 years. As there are four operational VVER-1000s at the Balakovo site, for example, all the reactor operations for the disposition mission could be carried out at a single site. If, on the other hand, these reactors were limited to one-third MOX fuel, at a relatively low enrichment of 2.5 percent, nine reactors would be required to accomplish the same task. As there are only seven operational VVER-1000s in Russia, either completion of additional reactors or use of some of the 10 operational VVER-1000 reactors in Ukraine would then be required. The Zaporozhe site in Ukraine has five plants in operation and one partly constructed; Zaporozhe and Balakovo together could handle an entire one-third MOX campaign at two sites. Use of weapons-grade plutonium in reactors in Ukraine would involve some important political issues, however, given the changing and uncertain relationship between Ukraine and Russia concerning both Ukraine's shipment of nuclear weapons back to Russia and Russia's provision of nuclear fuel for Ukraine's reactors.

Fuel Fabrication

As in the United States, the time at which disposition of excess WPu could begin would be paced by the availability of a MOX fuel fabrication facility. While Russia has laboratory-scale MOX fabrication facilities, no production facility with the required capabilities is currently operational.

A MOX fabrication facility with an intended capacity of about 100 MTHM/yr—enough to feed four VVER-1000s using full-MOX cores (processing as many tons of plutonium annually as the percentage in the fuel)—is reportedly roughly 50 percent complete at the Chelyabinsk-65 site. This facility was intended to produce MOX fuel for the planned BN-800 series liquid-metal breeder reactors. Construction on both this plant and the first of the BN-800s has been halted since well before the collapse of the Soviet Union due to lack of funds. Completing the plant and modifying it to produce LWR rather than liquid-metal reactor fuel would require several years at a cost in the range of hundreds of millions of dollars. The standards of safeguards, security, and ES&H this plant was designed to meet—or could practicably be modified to meet—are unknown.

Alternatively, a new MOX fabrication facility could be constructed, dedicated to the excess WPu disposition mission. The German company Siemens has proposed using disarmament assistance to build a replica of the Siemens fabrication facility already built at Hanau, which has a design capacity of 120 MTHM/yr. Siemens estimates the cost of building such a facility in Russia at roughly half a billion dollars, and believes construction could be accomplished in a relatively short period (Schmeidel 1992, cited in Berkhout et al. 1993).

Similarly, the French state-owned company COGEMA has expressed interest in participating in providing MOX fabrication capability for disposition of Russian WPu.

Reactor and Institutional Options

The public versus private issues in Russia are somewhat simpler than those in the United States, since MINATOM runs both the nuclear weapons complex and the civilian nuclear reactor industry. But because of the severe economic crisis in the former Soviet Union, U.S. or international financial assistance may well be required if long-term disposition of excess WPu in Russia is to be accomplished in the foreseeable future. Just as private investment might help reduce up-front capital costs in the United States, private investment or loans from international financial institutions such as the World Bank or the European Bank for Reconstruction and Development might help finance the operation in Russia, reducing the “line-item” costs that would have to be borne by any single government (see discussion of costs below). These institutions are already considering helping Russia to complete the VVER-1000 reactors under construction and to facilitate the shutdown of older unsafe reactors.

Approvals and Licenses

The political and institutional climate for plutonium use in Russia differs from that in the United States. In Russia, the government and the nuclear industry (controlled by MINATOM) are committed to a closed fuel cycle, including plutonium fuels, emphasizing fast-breeder reactors. MINATOM wishes to save the excess WPu for eventual use as startup fuel for future breeder reactors. Some others indicate a desire to sell the excess plutonium. Virtually all Russian government officials maintain that WPu has value that must be exploited.

According to news and other reports, however, the Russian public—after decades of government secrecy and the Chernobyl disaster—has become increasingly wary of all things nuclear and distrustful of all government assurances about environment and safety. Public resistance to plutonium use may therefore be significant. The regional and local authorities in Tomsk (a major production site for WPu), for example, have gathered sufficient strength in opposing the siting of a WPu storage facility there to call into question the viability of the site. The regulatory agency empowered (in principle) to regulate reactor siting and licenses, GOSATOMNADZOR, is seeking to define its role in the new Russia, and its future powers and attitudes toward plutonium use are uncertain.

Thus, the time required to gain the required licenses and approvals in Russia is more uncertain than in the U.S. case, and could ultimately prove to be either longer or shorter.

Safeguards, Security, and Recoverability

The risks of theft in transporting and processing plutonium in Russia under present circumstances appear high. Indeed, some analysts have argued that continued storage of the plutonium under high security until the Russian political and economic situation had stabilized would pose fewer risks than the processing and transport involved in the MOX option.

There are some important mitigating factors, however. As with the Hanford facility in the United States, the unfinished MOX fabrication facility in Russia is at a major nuclear weapons site. In addition, as noted above, the four VVER-1000s at the Balakovo site in themselves have sufficient capacity to carry out the WPu disposition mission if full-MOX cores were used. Thus, it might be possible to accomplish all processing of bulk plutonium at a single existing nuclear weapons complex site and all reactor use of plutonium fuel at one additional site. As in the U.S. case, all of the disposition steps should be subject to a stringent agreed system of safeguards and security.

This option meets the "spent fuel standard," though there would be significantly higher plutonium concentrations in the spent fuel than in typical commercial spent fuel.

Cost

Russian costs are uncertain, and no detailed analysis is possible with the information available. It is clear, however, that Russia has an overcapacity of low-cost LEU available for fueling its thermal reactors, which it is trying to market in the West to earn hard currency. It is also clear that significant up-front capital would be required to provide requisite plutonium fuel fabrication capability and to modify reactors to handle full-MOX cores. Therefore, substituting WPu for uranium in Russian LWRs would require a subsidy, probably in the range of hundreds of millions of dollars. The up-front capital investment (offset in part by later revenue) would be larger.

Because both capital and labor costs are substantially lower in Russia than in Western countries, however, it is possible that MOX produced in Russia would be competitive with LEU produced in the West. Contracts for sale of such MOX to Western utilities could serve as the basis for loans to provide the capital to build and operate a MOX fabrication plant. In this case, little if any explicit government subsidy might be needed to accomplish disposition of Russian WPu.

Environment, Safety, and Health

To a large extent, the ES&H impact of plutonium disposition in Russian reactors would depend on the resources applied to mitigate these impacts, as well as on the standards set. Standards for ES&H protection in the former Soviet

Union were low, and the resulting devastating environmental legacy is now becoming clear. New ES&H policies are evolving in Russia, with uncertain prospects. As noted above, while further study is required, it appears that the VVER-1000 reactors could be modified to handle full-MOX cores without decreasing their safety. The panel believes that with sufficient resources, MOX fabrication operations could be conducted in Russia (as in other countries) while meeting stringent standards of ES&H protection.

Other Issues

As with other options, policy-makers will have to consider how this approach fits with overall fuel-cycle policies. Assistance for using MOX in Russian reactors would provide a boost to the plutonium fuel cycle in Russia. There might also be some political impact in other countries whose civil plutonium programs are controversial (see discussion in Chapter 6, "General Considerations").

In addition to excess WPu, Russia has some 30 tons of separated civilian plutonium waiting to be fabricated into fuel, and more is being separated each year. Presumably an operational MOX fabrication facility in Russia, once available, would be used for more than just the excess WPu. Some Russian officials and European analysts have suggested that Russia should fabricate the civilian plutonium into fuel before beginning use of WPu, since the civil plutonium builds up dangerous radioactivity more quickly. Thus disarmament assistance for construction of a MOX facility might in effect sponsor civilian plutonium use in Russia—and commercial competition for MOX fabricators in Europe.

The Spiking Option

The issues facing the spiking option in the Russian case are similar to those in the U.S. case, described above. The reduction in reactor capacity factor resulting from more frequent refueling would have substantially greater impact in the former Soviet Union, however, because of the lack of electricity capacity margin in Russia and Ukraine. Reduction in overall electricity capacity there would not only create economic difficulties but would increase the pressure to extend operations of the RBMKs and VVER-440s to make up for the capacity loss of the VVER-1000s, conflicting with the goal of shutting down these less safe facilities.

The Elimination Option

To implement the elimination option in the former Soviet Union would require the utilization of reprocessing facilities as well as the power-reactor and MOX fuel fabrication facilities discussed above.

Reprocessing facilities exist in Russia for the extraction of plutonium from military production reactors—the source of the WPu which now requires disposition. In addition, there is a large civilian reprocessing plant, known as “Mayak,” at the Chelyabinsk site, and consideration is being given to construction of a very large civilian reprocessing plant at the Krasnoyarsk site. Presumably, these facilities could be utilized, as in the case of similar facilities in the United States, to reprocess MOX fuel assemblies in the elimination option. The “head-ends” of the military reprocessing facilities would have to be modified to handle the MOX fuel from the power reactors. In light of the history of operations of these facilities in Russia, however, it is questionable whether they meet either adequate environment and safety standards or safeguards standards appropriate to this mission. Thus, the destruction option would probably require the construction of new reprocessing facilities, as might also be the case in the United States.

CURRENT-GENERATION CANDU REACTORS

Description of Technology and Status

In Canada, commercial nuclear power is obtained from CANDU (Canadian deuterium-uranium) reactors that use uranium fuel and pressurized heavy-water coolant confined to a system of pressure tubes in a calandria structure. The heavy-water moderator at low pressure is in the intervening spaces between the pressure tubes.¹⁸ The low neutron absorption by heavy water allows natural uranium to be used as fuel. Even so, only limited burnup of the natural uranium fuel is possible, limited by the low reactivity of natural uranium. This necessitates frequent replacement of irradiated fuel, accomplished by a pair of refueling machines that operate continuously while the reactor is at full power. Each fuel assembly is a cartridge of Zircalloy-clad rods of uranium dioxide, about 60 centimeters long. Each pressure tube contains several fuel cartridges. Discharged fuel is stored, to be later emplaced in a geologic repository. Twenty-two CANDU reactors are operating at high capacity factors and low fuel costs, at outputs varying from 515 to 881 MWe (*Nuclear News* 1993). Use of MOX in CANDU reactors is an option for WPu disposition. While this option appears technically and economically feasible (for either U.S. or Russian excess WPu), major political questions remain open. The panel notes that for this option vir-

¹⁸ Japan also has a heavy-water moderated thermal reactor at Fugen, with a significantly different design from the CANDU, which was designed for and has operated with full-MOX cores. This reactor, however, is too small (557 MWt, 165 MWe) to play a major role in the plutonium disposition mission. Construction of larger reactors of this type in the United States or other countries for the WPu disposition mission has not been proposed, and is not analyzed further in this report.

tually all of the information made available to the panel was provided by the vendor, and had not yet been reviewed by DOE or other organizations.

Atomic Energy of Canada, Limited (AECL), the CANDU vendor, reports more than 25 years of experience with experimental irradiation testing of MOX fuels, with plutonium loadings between 0.5 and 3.0 percent. An experimental MOX fabrication facility was installed at Chalk River in 1975, and operated until placed on standby in 1987; this facility is now being reopened, and a resumption of MOX fabrication is expected in late 1995. More than 350 MOX fuel elements have been irradiated, in a zero-power test reactor, two research reactors at Chalk River, and in a prototype CANDU reactor. In the 1970s, six 19-element bundles containing 3 percent plutonium were irradiated to maximum burnups of 18,000 MWd/MTHM. One bundle remained in-reactor for 14 years, resulting in an outer-element burnup of 46,000 MWd/MTHM. In the early 1980s, six 37-element fuel bundles (similar to the bundles now proposed for plutonium disposition, but containing only 0.5 percent plutonium) were irradiated to burnups in excess of 21,000 MWd/MTHM. Thus the existing MOX irradiation experience includes the plutonium loadings and burnups relevant for plutonium disposition, with performance described as comparable to that of uranium fuel. Issues identified for further study are the effects on performance of the metal-to-oxygen ratio in the fuel and of the particle size (AECL 1994, pp. 2-57 to 2-59; Boczar et al. 1994).

The CANDU fuel element contains UO_2 pellets in Zircalloy cladding. The fuel design has evolved toward the current use of thin cladding, with a minimum thickness of 0.38 mm. This thin cladding improves neutron economy (an important factor in a reactor fueled by natural uranium), but the lifetime of the cladding may limit the burnups that could be achieved with fuels enriched with plutonium.

The manufacturer reports current average discharge burnups of approximately 8,300 MWd/MTHM of uranium, with maximum local burnups in the range of 15,000 MWd/MTHM. Although both the MOX experience described above and the occasional occurrence of maximum bundle average burnups of over 25,000 MWd/MTHM during CANDU operation suggest that CANDU fuel can withstand high burnups (no performance limits resulting from the high burnup operation had been identified at the time of a 1983 summary presentation on fuel performance) (Gacesa et al. 1983), representatives of the vendor indicated that they would not have high confidence in cladding performance at burnups significantly beyond the current nominal maximum local burnup without extensive further testing (Feinroth 1993). The vendor's reference case calls for an average burnup of 9,700 MWd/MTHM with MOX fuel if the current fuel design is used, or 17,700 MWd/MTHM if the new CANFLEX design now being tested is used (AECL 1994).

CANDU reactors appear to be capable, without physical modification, of handling 100-percent MOX cores. The vendor concludes that: "There should

not be any significant changes in the design safety margins of the CANDU MOX fueled reactor systems relative to the current natural uranium fueled operations" (AECL 1994, p. 8-1).

As in the U.S. LWR case, however, no CANDU reactors are currently licensed to use MOX fuel, and favorable regulatory review of the safety of their operation in this mode would be required. While the CANDU reactor design is in principle even more easily adaptable to full-core MOX operation than most LWRs, at the same time the technical uncertainties concerning MOX use in CANDUs must be considered somewhat larger than in the LWR case, given the lack of MOX operating experience in CANDU reactors. There are also considerable uncertainties concerning the economics of this option, as no one has ever produced CANDU MOX fuel before.

The vendor suggests that the four 825-MWe units located at the Bruce A site would be particularly suitable for WPu disposition. Ontario Hydro, the utility that owns these reactors, has expressed interest in studying the plan. The utility has taken part in an initial study funded by DOE, and it sponsored a public meeting near the plants in June 1994 to discuss the idea of using plutonium there (AECL 1994, pp. 3-4 to 3-14).

AECL has also pursued preliminary discussions with the Russian government regarding this concept. The two sides signed a document calling for joint study of the use of Russian plutonium in CANDUs, if a third party (presumably the U.S. government) would provide funding. It is conceivable that production of CANDU MOX in Russia, with that country's low labor and materials costs, would be cheap enough that it could be sold to Canada at a modest profit (Feinroth 1994). (As noted above, this is also possible in the LWR MOX case.) While new CANDU reactors could also be built in Russia to consume the plutonium there, this would be slower and more expensive than using existing reactors, and would not appear to have great advantages compared to constructing reactor types with which Russia is more familiar, such as light-water reactors.

Compared to the use of U.S. LWRs, the use of CANDU reactors would have both advantages and disadvantages, discussed in the following subsections.

Advantages

Straightforward Adaptation to Full-MOX Core Operation. As noted, it appears that existing CANDU reactors, without physical modification, could operate with full-MOX cores. In normal CANDU operations with natural uranium fuel, more than half of the energy is provided by fissioning plutonium produced in the fuel as the reactor operates. As a result, adding plutonium to the initial fuel would represent a smaller change in the physics of the reactor core than in the case of LWRs. Moreover, the structure of the CANDU reactors allows plenty of space for added controls, and additional neutron absorbers could be dissolved in the heavy-water moderator used in the reactors. Thus the vendor

concludes that no physical modifications would be required to handle substantial quantities of plutonium in CANDU reactors. Several factors should be mentioned in this regard.

First, CANDU reactors offer greater flexibility for maintaining control-absorber worth with plutonium fuels. In a calandria-type heavy-water reactor most of the reactor-core volume is occupied by the relatively cool, low-pressure, heavy-water moderator. In the CANDU design the movable control absorbers are located in the moderator region. There is ample space to increase the size of each absorber, or increase the number of absorbers, as necessary to counteract the higher neutron absorption in plutonium fuel and to maintain control-reactivity worth. The CANDU design also provides for boron absorber dissolved in the moderator for shim control of reactivity. The well-thermalized neutron spectrum provides greater control worth for a given amount of boron than in a plutonium-fueled LWR. The movable and soluble absorbers are also more effective because they are located in the higher-flux region where thermal neutrons are formed. Thus, the existing CANDUs can provide sufficient reactivity control to handle full-MOX loading. The designers expect that the reactivity control would even be sufficient for a full loading of nonfertile plutonium fuel.

Second, the CANDU design features thermal decoupling between the moderator and the fuel. The heavy-water moderator is at low pressure and is thermally insulated from the hot fuel and coolant in the pressure tubes. Therefore, there is only a weak coupling between an assumed sudden increase in power and temperature of the fuel and the temperature of the moderator.¹⁹ Neutron temperature effects are, therefore, of less concern than in LWRs, where the moderator is also the coolant, whose temperature increases rapidly following a spurious increase in local fission rate. Also, in the CANDU, dissolved boron for shim reactivity control is in the moderator rather than in the coolant. In a pressurized-water reactor, with boron dissolved in the coolant, thermal expansion of the coolant accompanying an assumed sudden increase in local fission rate reduces the amount of boron near the fuel and tends to add to the reactivity.²⁰ But there is no boron in the heavy-water coolant of a CANDU reactor, and relatively little reactivity is added by moderator heating because of the weak thermal coupling between fuel and moderator.

Third, prompt neutrons have a considerably longer lifetime in a CANDU reactor than in other reactors because of the relatively large volume of heavy water with weak absorption of thermal neutrons. This results in a much longer

¹⁹ Moderator heating tends to make a positive contribution to reactivity when using Pu-239 fuel, because it shifts the thermal spectrum toward the 0.3-electron volt absorption resonance in Pu-239. The increase in the fission cross-section of Pu-239 is more important than the increase in the capture-to-fission ratio of Pu-239 in this region.

²⁰ In an LWR it is a licensing requirement that the boron concentration not be large enough to make the moderator coefficient of reactivity positive.

time-constant for power changes following an accidental insertion of reactivity sufficient to initiate a prompt-critical excursion. Thermal shock is less likely, and more time is available for safety-system actuation.

Finally, the addition of dysprosium in the MOX fuel has a beneficial effect on the reactor's performance in a loss-of-coolant accident (LOCA). CANDU reactors fueled with natural uranium will become more reactive in the event of a LOCA. For this reason, CANDUs are equipped with two independent fast-shutdown systems. The plutonium-dysprosium fuel is designed to counter this positive reactivity coefficient, so that the reactor will become less reactive in the same scenario. Analysis by the vendor shows in the event of a large-break LOCA, power output in a uranium-fueled reactor could increase to over four times its nominal value, but would not increase at all in the MOX fueled system (AECL 1994, p. 2-101).

Simplified Fuel Fabrication. CANDU fuel is produced in smaller and simpler units than those typical of LWRs, potentially reducing the fabrication cost, which is a substantial fraction of the total cost of MOX use. The specifications that CANDU fuel must meet, in such areas as granularity, pellet shape, and the like, are less stringent than those required for LWR fuel. Experience with LWR fuel, however, suggests that the need for tight specifications increases with greater enrichment, so adding plutonium to CANDU fuel might require specification tolerances closer to those of typical LWR fuel.

Continuous Fueling Offers Additional "Spiking" or High Burnup Options. Operating with natural uranium fuel requires frequent replacement of fuel, so CANDU reactors use mechanisms for continuous refueling while at power. The refueling mechanisms require relatively short (60 cm) fuel-assembly cartridges. As irradiated fuel cartridges are removed from one end of a pressure tube, fresh cartridges are inserted at the other end. Within a given pressure tube the cartridges can be moved along the pressure tube at a rate proportional to the average neutron flux in the pressure tube. All discharge cartridges can be irradiated to the same burnup, not possible in LWRs refueled by periodic batch replacement.

The ability to refuel without shutdown offers additional options for the "spiking" option. While the "spiking" approach would still require added capital expenditures for a larger fuel fabrication facility, it would not decrease revenue as a result of reactor downtime for refueling. Moreover, the remotely operated refueling machines would reduce concerns regarding possible worker exposures to radiation in reloading "spiked" fuel to finish burning it to "spent" fuel.

The small cartridges and remotely operated refueling machines also make it possible to consider recycling discharged cartridges for further irradiation, providing a possible means of greater annihilation of plutonium than otherwise obtainable in once-through irradiation. Fresh first-cycle plutonium-fueled cartridges could supply neutrons to overcome the neutron-absorbing fission-

product poisons in nearby recycled cartridges. Irradiation of individual cartridges could proceed to the material limits of the fuel material, rather than being limited by reactivity considerations. Similar operation of a plutonium-burning LWR would be difficult because of the very long (approximately 5-meter) fuel assemblies and the need to shut down and depressurize for refueling.

High Neutron Economy Potentially Contributes to High Burnup on Once-Through Cycle. The high neutron economy of the CANDU makes it possible to irradiate fuel of a given fissile concentration to much greater burnup, and to a greater degree of annihilation in one irradiation cycle, than is possible in other thermal reactors, such as the LWR or high-temperature gas-cooled reactor (HTGR). A measure of the neutron economy is the conversion ratio, defined as the number of neutrons absorbed in fertile material per fissile atom destroyed. For reactors fueled with U-235 in U-238, the conversion ratio of the CANDU is 0.75, compared with about 0.6 for a LWR or an HTGR. High neutron economy would be important for plutonium-burning fuel cycles that sought high burnup during an irradiation cycle using fertile-free plutonium fuels. (It still may not be possible, however, to design fuels that can achieve the very high burnups claimed for HTGRs, described below.)

In designing thermal reactors for efficient utilization of plutonium as a fuel, it is important to seek a well-thermalized neutron spectrum to avoid the higher capture-to-fission ratio of Pu-239 that results when the thermal-neutron spectrum is shifted to higher energies. Calandria-type heavy-water reactors have characteristically lower-energy and better-moderated spectra of thermal neutrons than do LWRs, particularly with plutonium fueling. Although Pu-239 is consumed by both neutron capture and fission, the nonfission capture produces additional plutonium isotopes such as Pu-240, 241, and 242. If disposition of WPu emphasizes the destruction of all plutonium isotopes, nonfission capture of neutrons in Pu-239 is less productive in destroying plutonium than using those neutrons for more extended fission of Pu-239 during an irradiation cycle.

Disadvantages

Uncertain Canadian Acceptance. The use of existing CANDUs would have to be approved by the Canadian government, the reactor operators (primarily the Ontario Hydro utility), and the relevant regulators (the Atomic Energy Control Board). AECL, the government-owned designers of CANDU, and Ontario Hydro both participated in the initial DOE-funded study of the concept, completed in mid-1994, though neither has made a firm political commitment to support it. The Canadian government has reportedly suggested to U.S. representatives that the two countries form an expert group to explore the idea. But further discussions between the U.S. and Canadian governments would be required before it could be determined whether this approach had enough political support to be a

practical option. Canada has previously avoided using either enriched uranium or plutonium fuels in CANDU reactors, and might reject this plutonium-use option as well. Yet Canada has also traditionally played an active role in disarmament; playing a central role in disposition of materials resulting from nuclear arms reductions might well be appealing enough to overcome the resistance to use of weapons materials. Canadian public acceptance is also an open question.

Large-Scale International Plutonium Transport. The distance over which plutonium would have to be transported to be burned in CANDU reactors would be greater than that in using U.S. LWRs, even if all the CANDU reactors involved were at a single site. The attendant controversies and risks of theft would be correspondingly larger. Possibly more important in political terms than the sheer distances is the need for the material to be shipped across international borders, to a non-nuclear-weapon state.

Lower Plutonium Loading Requires More Fuel Fabrication. As described in more detail below, the CANDU manufacturer suggests WPU loadings of only 1.5–2.7 percent. This would require larger quantities of fuel to be produced than in the case of LWRs, increasing costs and countering the advantage of simpler fuel fabrication described above.

Lower Radioactivity and Small Size of Bundles. Because of the relatively low burnup (even when enriched with plutonium) and small size of the CANDU MOX bundles, the gamma-radiation dose rates from them would be somewhat lower than those from LWR spent fuel of equal age.²¹ The spent CANDU MOX, however, would have substantially higher dose rates for several decades than the large quantities of much older LWR spent fuel that will exist at the time the CANDU MOX spent fuel would be discharged. The small size of the CANDU bundles would make them slightly simpler to steal.

Safeguards Issues of Online Refueling. Fuel can be removed from CANDU reactors at any time without shutdown of the reactor, and the fuel elements are substantially smaller and more portable than is the case for LWRs. Therefore, CANDUs require more intensive safeguarding than do LWRs. For fuel containing more plutonium, still more intensive safeguarding would be needed. Both CANDU reactors and the fresh MOX fuel in store at either an LWR or a CANDU, however, require continuous safeguarding in any case. In addition, the task of accounting for and securing complete fuel assemblies for either a CANDU or an LWR is substantially easier than that of accounting for bulk plutonium at a MOX fabrication plant. Therefore, the *net additional* security risks

²¹ The surface dose rate 10 years after discharge from a single bundle irradiated to 9,700 MWd/MTHM is about 5,500 rem/h (roentgen-equivalent-man per hour) (AECL 1994, p. C-19), compared to a surface dose rate of 18,000 rem/hr at the same time for a pressurized-water reactor fuel bundle irradiated to 40,000 MWd/MTHM (see Table 6-5). The dose rate also falls off more rapidly with distance for the CANDU fuel bundle, because of its more compact size.

of using CANDU reactors for this mission compared to using LWRs would be relatively small.

Reactor Throughput, Once-Through Fuel Cycle

AECL Technology (the U.S. subsidiary of AECL) describes two main fuel options for burning plutonium in CANDU reactors (AECL 1994). As a reference case, AECL chose the Bruce A reactors, which each have a unit core thermal power of 2,832 MWt, providing 825 MWe of gross electricity. Since the units also produce excess steam for local heating, the total gross electrical equivalent output is 904 MWe per unit. Four of these units are in operation at the Bruce A site. The reference case of natural uranium fueling (i.e., roughly 0.7 percent U-235) operates at a typical discharge burnup of 8,300 MWd/MTHM.

The first option for plutonium disposition, using fuel elements similar to those now used with natural uranium, would have an average plutonium loading of 1.5 percent. The innermost pin and the first ring of pins surrounding it (7 of the 37 pins in the bundle) would consist of 5-percent dysprosium-oxide burble absorber in depleted uranium. These absorber pins would compensate for the greater reactivity of the plutonium fuel compared to natural uranium; in a CANDU spectrum, the dysprosium absorption would be reduced at approximately the same rate as the plutonium reactivity, helping to flatten the reactivity of the fuel over its life. The next ring would have 2-percent plutonium oxide, and the outermost ring 1.2 percent, for a total of 0.23 kg in each bundle. Average discharge burnup would be 9,700 MWd/MTHM, but because the combination of MOX and dysprosium absorbers would help to flatten the power distribution through the core, peak burnup would be increased only slightly compared to the reference case (15,500 MWd/MTHM versus 15,000).

With this approach, each reactor would consume 1.05 tons of plutonium per year; two of the Bruce A reactors could dispose of 50 tons of WPu in 25 years. No hardware changes to the reactor system would be needed to operate within the existing safety envelope, according to the manufacturer.

The second concept considered by AECL Technology is to use a new fuel design still undergoing testing, known as CANFLEX, designed to achieve higher burnup. CANFLEX bundles contain 43 elements rather than 37. Demonstration irradiations of this design in power reactors are expected to begin in 1996 (AECL 1994, p. 2-77). In this case, AECL Technology again envisions the innermost seven pins being dysprosium oxide (6.0 percent in natural uranium), with the outer ring being 2.1-percent WPu, and the intermediate ring 3.6 percent, for a total of 0.38 kg per bundle—nearly twice the plutonium per bundle of the previous case.²² This much larger amount of plutonium, however, is com-

²² The panel was informed by representatives of the vendor that this figure was arrived at not by analysis designed to estimate the maximum plutonium loading that could be safely accommodated in CANDU reactors, but rather because this was the loading required to meet DOE's speci-

pensated directly by a much higher burnup (17,100 MWd/MTHM versus 9,700 MWd/MTHM), so that the total amount of plutonium consumed in each reactor each year is roughly the same, still requiring 50 reactor-years for disposition of 50 tons of WPu.

The primary potential advantage of this latter approach is not speeding the process, but drastically reducing the number of MOX fuel bundles that would have to be fabricated, thereby potentially reducing the cost of the operation. The disadvantage of this approach is that CANFLEX fuel is not yet licensed, and a more substantial delay before CANFLEX MOX fuel could be produced would be expected. The manufacturer, therefore, suggests beginning an initial program with MOX in standard CANDU bundles, switching to CANFLEX when it becomes available.

Fuel Fabrication

Like the United States, Canada has no MOX fuel fabrication capacity. Fabricating MOX fuel for CANDUs at the Hanford FMEF facility would be the most expeditious approach, with the same caveats as in the LWR case. The vendor has in fact examined fabrication of MOX fuel in the FMEF in considerable detail, and believes that large throughputs of CANDU MOX fuel (over 160 MTHM/yr) are possible, by taking advantage of additional floor space not used by the current MOX fabrication line in the facility (AECL 1994).

Approvals and Licenses

Gaining approval of the various Canadian institutions and the Canadian public would be a major hurdle for the CANDU option. Licensing reactor operations with plutonium would probably be a less difficult issue than securing agreement on the basic approach. Licensing procedures and standards for plutonium use in Canada, set by the Atomic Energy Control Board (AECB), are different from those used by the NRC. In general, the process in Canada relies more on cooperation between licensees and the board, and less on an adversarial process.

Safeguards, Security, and Recoverability

The safeguards concerns regarding fuel fabrication are similar for LWRs and CANDUs. Because of the need to transport plutonium over longer distances, transport risks would be somewhat greater for CANDUs, and because of

fied goal of consuming 100 tons of WPu (the potential combined excess stocks of the United States and Russia) in 25 years of operation, given the estimated capacity of the FMEF fabrication facility (Feinroth 1994). Additional studies should be pursued to determine the maximum safe plutonium loading; higher plutonium loadings would increase the rate of plutonium disposition and reduce the number of fuel bundles that would have to be fabricated, potentially lowering costs.

the reactor's online refueling capability and the portability of the fuel elements, the risks of theft or diversion of fabricated fuel from the reactor could be somewhat greater as well. Both of these risks could be reduced to very low levels with the application of sufficient resources.

This option would make the plutonium roughly as difficult to recover as the plutonium in commercial spent fuel. As mentioned above, however, the generally lower burnups in the CANDU case and the small size of the spent fuel cartridges mean that the radiation field from each unit of spent fuel is somewhat lower.

Cost

The cost of this option is difficult to estimate, as experience fabricating MOX fuel for CANDU-type reactors is very limited.²³ On the one hand, an argument can be made that the subsidy required would be less than in the LWR MOX case, because: (1) the fuel is simpler and probably cheaper to fabricate; and (2) at least in the case of the advanced CANFLEX fuel, the MOX fuel would have a higher energy content (and hence a longer fuel life) than the natural uranium fuel CANDU reactors normally use, so that the increased per-kilogram cost of fabricating the MOX fuel would be compensated, in part, by the reduced amount of fuel that would have to be fabricated. On the other hand, the subsidy required might also be more than in the LWR case, as the amount of natural uranium CANDU fuel each kilogram of MOX would substitute for—whose cost would be subtracted from the MOX cost in calculating the subsidy required—would be more than \$1,000 cheaper than the LEU LWR fuel a kilogram of MOX could substitute for.²⁴

The vendor estimates, probably optimistically, that the U.S. FMEF fuel fabrication facility could be modified for an overnight cost of \$118 million, and produce MOX with annual operating costs of about \$64 million per year for the reference MOX fuel and 20 percent more for the CANFLEX fuel option. As described in detail in Chapter 6, this implies total costs comparable to the middle of the range for the options that employ currently operating LWRs.

Environment, Safety, and Health

Use of plutonium in CANDU reactors would raise the same general concerns as those described for LWRs. In this case, because of the higher burnup of

²³ There is more experience fabricating MOX fuel for the Japanese Fugen heavy-water reactor, which has a similar design.

²⁴ Of course, whatever might be achieved by using a fuel enriched in plutonium, an even better economic result could be achieved by using enriched uranium fuels, which would not involve the extra handling costs of plutonium; but Canada has not chosen to use enriched fuels. Thus the subsidy for use of plutonium in the case of CANDUs arises in comparison to a fuel cycle that is currently less than optimally efficient, given current fuel prices.

the MOX compared to the natural uranium fuel it would replace, the total mass of spent fuel requiring disposal would be somewhat reduced.

Other Issues

The potential perceived impact of this approach on fuel-cycle policies would be more complex than in the U.S. LWR case. On the one hand, by providing excess plutonium free of charge to another nation, the United States would be demonstrating that it saw no economic value in the material and was encouraging its use in reactors only as an arms-control measure. On the other hand, the United States would still be encouraging use of separated plutonium as reactor fuel on a scale wider than would otherwise be the case in a non-nuclear-weapons state (see discussion in Chapter 6, "General Considerations").

The Spiking Option

The "spiking" option is also a possibility for the CANDU case. For spiking, the refueling machines would be operated at their capacity of 22 fuel bundles per day. The discharge burnup in this case would be only 5,100 MWd/MTHM. The plutonium processing rate per reactor would therefore be nearly doubled. By the same token, however, twice as much fuel fabrication capability would be required, potentially incurring significant additional capital cost. Arrangements for storing the radioactive "spiked" fuel and then eventually reinserting it into the reactor for burning to the spent fuel standard would be required.

The Elimination Option

The CANDU system, with its high neutron economy and on-power refueling offering flexibility in fuel management, could offer an opportunity for achieving substantial burnups without reprocessing if new fuels were developed for that purpose. The manufacturer suggests that development of a nonfertile fuel could allow nearly 60-percent destruction without reprocessing or fuel shuffling; higher destruction fractions could be achieved by fuel shuffling, that is by moving the burned bundles to other locations for additional irradiation.²⁵ Development of such a nonfertile fuel might be easier for the CANDU because of the lower fuel temperatures.

Because of the flexibility of the CANDU control system, as described earlier, it might be possible to control the large reactivity changes during an irradiation cycle using nonfertile fuel more easily than in an LWR. The higher neutron economy of the CANDU should make it possible to maintain criticality to a greater fraction of initial fissile atoms burned than in an LWR. This, together

²⁵ AECL (1994, pp. 2-127 to 2-140). Nonfertile fuels would have more impact in the CANDU case than in the LWR case, because of the smaller loading of fissile material (and hence higher neutron flux) in the CANDU case.

with the ability to discharge fuel to a more uniform burnup, should result in a greater fractional burnup of initial plutonium in a given irradiation cycle than in an LWR.

The relatively small fuel bundles of the CANDU and the remote refueling machines invite the possibility of recycling discharged high burnup fuel for further irradiation and burnup. The recycled discharged fuel could be located near fresh make-up fuel to maintain reactivity and promote further burnup.

As with LWRs, more complete elimination could be pursued with reprocessing and recycle, either of MOX fuels or of nonfertile fuels. From a technical point of view, there are no reasons to exclude CANDU reactors if reprocessing and recycle to eliminate WPu are to be considered for any reactor concepts. Like LWRs, however, a reprocessing cycle for CANDU reactors as currently conceived would involve substantial handling and processing of separated, weapon-usable plutonium, raising some security risks. New reprocessing facilities would have to be provided.

POTENTIAL INVOLVEMENT OF WEST EUROPEAN AND JAPANESE FACILITIES

Description of Technology and Status

Under established civil plutonium fuel programs, commercial reactors in Europe and Japan plan to process well over 100 tons of civilian plutonium over the next decade. Plutonium storage and transport arrangements, fuel fabrication capabilities, and reactors licensed to handle plutonium for this task already exist or are planned.²⁶ Technical feasibility is amply demonstrated.

One possibility for long-term disposition of excess WPu, therefore, is to substitute this weapons material for the civilian plutonium.²⁷ Pits would be processed to plutonium oxide in their country of origin and the resulting oxide shipped to Europe or Japan for fabrication and use.²⁸ In particular, such an approach would enable disposition of Russian plutonium in the near term while

²⁶ As of 1993, eight LWRs in France, seven in Germany, and two in Switzerland are using MOX fuel, and more are licensed to do so; Belgium and Japan plan to begin loading MOX fuel in commercial reactors later in the decade.

²⁷ This substitution approach was first proposed by Thomas Cochran and Christopher Paine of the Natural Resources Defense Council (NRDC 1993).

²⁸ Alternatively, rather than making use of both the reactors and MOX fabrication capabilities existing or planned in Europe and Japan, one might make use of only the MOX fabrication capabilities, shipping the resulting fuel back to the country of origin. In that case, however, another round of international shipment of plutonium would be required; and since these existing and planned MOX fabrication facilities will have a hard time handling all the projected civilian plutonium, adding WPu would mean that separated civil plutonium would build up. Some of the reactors in Europe and Japan currently scheduled to use plutonium fuels would not receive the products of these fabrication facilities as expected and would have to switch back to uranium fuels.

minimizing the safeguards and security concerns that would be involved in large-scale MOX fabrication in Russia during the current period of high risk of theft.

In this case, the initial processing and shipment step would be the only aspect of plutonium handling beyond that already planned—with the important caveat that all these facilities would now be using weapons-grade rather than reactor-grade plutonium. From the point of view of civilian nuclear-energy production, the WPu would be less radioactive (and therefore easier to fabricate) and have slightly higher energy content than the reactor-grade material it would replace.

What would happen to the displaced civilian plutonium? Three main possibilities exist: one is to expand MOX operations in these countries, involving more reactors and fabrication facilities than those currently planned, so as to process *both* the civilian and the excess WPu. This approach would, however, mean an expansion in global handling and use of separated plutonium (as would options involving MOX use in the United States and Russia). MOX fabrication capacity worldwide is in a period of rapid expansion, with facilities larger than any that now exist scheduled to come on line in the next few years in France, Britain, and later in Japan as well. These facilities are sized to provide adequate capacity to balance the plutonium to be separated by civilian reprocessing in the same countries. A large MOX plant is nearly complete at Hanau in Germany, but has not been allowed to open because of licensing difficulties arising from political opposition in the state of Hesse. Many of the German utilities have succeeded in acquiring MOX contracts elsewhere, and some have begun to cancel their post-2000 reprocessing contracts with the French state-owned company COGEMA and British Nuclear Fuels, Limited (BNFL). The Hanau plant's capacity could potentially be used to fabricate either U.S. or Russian WPu (or both) into MOX fuel, if a political arrangement allowing it to be opened for this purpose could be reached. This general type of approach could be particularly attractive if the owners of the plants with such excess capacity, having paid the capital cost of the plants with the MOX fabrication contracts already in hand, would be willing to contract for additional fabrication at the cost of operations. If capital cost were excluded, MOX could be fabricated from free WPu for less than the cost of LEU. Russian plutonium, for example, might be shipped to Germany and fabricated into MOX which would be sold to German utilities at the price of LEU; the difference between this price and the operations cost of producing the MOX could go to Russia to pay the costs of secure storage and processing the pits to oxide.

Another possibility would be to continue reprocessing and MOX use as planned, and store the separated reactor-grade plutonium displaced by the WPu. The net result would be to convert an excess stock of separated weapons-grade plutonium to an excess stock of separated reactor-grade plutonium of roughly equal size—a step the panel considers to be of too limited benefit to justify the

complications of the required international agreements and the risks of the required international transport.

The third possibility is to defer reprocessing until existing excess stocks of separated plutonium (both weapons-grade and reactor-grade) are consumed. Reprocessing plants would be kept in cold standby until then.²⁹ This approach would consume both the projected surplus of WPu and the projected surplus of separated civilian plutonium, without predetermining fuel-cycle choices for the period after the current stocks of separated plutonium are consumed. Such an arrangement, however, would require complex international agreements altering a web of existing contracts and spent fuel management policies.

Reactor Throughput, Once-Through Cycle

If the necessary agreements could be reached expeditiously, this would be by far the most rapid reactor option, since the pacing steps of building new fabrication capacity and licensing the various facilities would be avoided; as noted, over 100 tons of plutonium are expected to be processed in this way over the next decade in any case, so it would be *technically* possible to process the entire stock of U.S. and Russian excess WPu over that period. Reaching the necessary agreements could involve extended and unpredictable delays, however, with no guarantee of ultimate success.

Approvals and Licenses

Gaining the international agreements necessary to ship tens of tons of weapons-grade material from Russia or the United States to Europe would be difficult, and the amount of time required is impossible to predict. The highly enriched uranium (HEU) purchase agreement already negotiated between Russia and the United States could provide a model—though it should be noted that as of late 1994, three years after such an arrangement was first proposed, none of the material had yet been delivered.

Gaining agreement to the option involving a deferral of existing international reprocessing contracts would be particularly problematic. France and Britain share much of the world market for commercial reprocessing, and they have just completed multi-billion-dollar investments in new facilities. Any proposal to defer reprocessing for an extended period would be seen as a threat to these businesses. Even substantial financial compensation might not be suffi-

²⁹ This is the approach suggested by NRDC. An important part of this problem is that a substantial amount of the plutonium slated to be used as fuel in civilian reactors over the next decade has already been separated—some 60 tons in Europe and Japan as of the end of 1992. Thus, decisions would have to be made as to whether the weapons stock (which poses a somewhat greater proliferation risk) or the civilian stock (which will build up radioactivity more quickly in storage, requiring further processing if storage is prolonged) should be used first.

cient to overcome such objections. A multinational negotiation would be required in a forum not yet defined.

If some relevant countries were interested in pursuing this option, but others were not, the substitution of WPu for civilian separated plutonium might be only partial. Britain, for example, might agree to defer operation of THORP (Thermal Oxides Reprocessing Plant) and fulfill its contracts with WPu instead, even if France continued with its reprocessing operations as planned.

If reprocessing were deferred for an extended period, more spent fuel storage would be required. From the point of view of utility owners of nuclear reactors in countries such as Germany and Japan, the opportunity to dispose of their spent fuel is one of the primary advantages of reprocessing. These utilities might be very reluctant to agree to an additional decade's worth of spent fuel simply building up at their reactor sites.³⁰ It is also an open question whether the public in France and Britain would accept the idea of highly radioactive spent fuel continuing to be shipped from abroad to reprocessing sites in their countries for storage, with no reprocessing planned for years to come.

Convincing the Russian government to accept such an approach would almost certainly require financial compensation for the material, as in the case of the HEU deal. Even then, some opposition to sending large amounts of a key strategic material abroad could be expected.

The international controversy provoked by the recent shipment of 1.7 tons of reactor-grade plutonium oxide from France to Japan suggests the political difficulties that would be faced by the much larger shipments of weapons-grade plutonium required for the plutonium disposition mission. To displace civilian plutonium to be used in Europe with Russian excess WPu, however, would require only overland transportation, which is less controversial than recent well-publicized shipments by sea have been. The association with arms reduction should also help reduce public criticism. Shipment of large quantities of weapons-grade plutonium, rather than merely reactor-grade plutonium, to non-nuclear-weapons states such as Japan and Germany would almost certainly arouse controversy with neighboring states, however.

Safeguards and Security

Since in this case WPu would displace separated plutonium operations that would take place in any case, the *net additional* safeguards issues involved in this option are smaller than those in other cases. The net additional risks would come from the pit processing required for all options; the large-scale interna-

³⁰ Indeed, such an approach would raise a number of legal issues. Many utility licenses limit the amount of spent fuel that can be stored onsite; license amendments, therefore, could be required. In Germany, reprocessing of spent fuel was until recently required by law. The law still requires utilities, as a condition of continued reactor operation, to have specific plans for spent fuel management extending at least six years into the future.

tional shipment of plutonium, central to this option; and the difference in proliferation risk involved in the shift from reactor-grade to weapons-grade plutonium. The key potential advantage of this approach, from a security point of view, is that disposition of Russian plutonium could be accomplished quickly, with most of the bulk handling of separated plutonium required occurring in other countries, where theft risks currently appear to be lower. The need for an agreed, international approach to safeguards and security is even more obvious here than it is in other cases. The risks involved in the large-scale international transport of plutonium required in this option are difficult to judge and depend on the resources applied to reducing them.

Cost

In this option, a variety of parties would probably demand financial compensation for the materials used, the use of facilities, or the disruption of previous plans. Russia would probably insist on financial compensation for plutonium used abroad in this way, making it effectively a plutonium purchase arrangement similar to the HEU deal. MOX fabricators could be expected to charge at least the cost of operations, and possibly the full commercial rate for MOX fabrication. In the variant involving a deferral of reprocessing contracts, the reprocessors whose contracts would be delayed or canceled would probably also require compensation—perhaps by means of continued payments on the existing contracts (since those who were to receive plutonium would still be receiving plutonium without reprocessing). Delaying reprocessing of a decade's worth of spent fuel would require additional spent fuel storage either at reactor or reprocessing sites. All told, the subsidy required to financially compensate all the relevant parties in the reprocessing-deferral variant might be comparable to or higher than the subsidy required to burn plutonium in LWRs that would otherwise burn LEU, discussed above.³¹

Environment, Safety, and Health

In the variant involving a net expansion of MOX use, the ES&H impacts would be comparable to those described above in the case of U.S. LWRs. In the variant involving deferral of reprocessing, the net additional ES&H burden

³¹ Under a financing scheme put forward by the Natural Resources Defense Council, money from MOX-burning electric utilities that would have been spent, under existing reprocessing contracts, on reprocessed civilian plutonium would instead be divided between paying a fair rate of return to the investors in commercial reprocessing plants that would not be operated and paying Russia for its WPu. The option, in this concept, would not require subsidies beyond those already being paid for reprocessing and MOX use. Additional subsidies would probably be required, however, for purposes such as compensating the reprocessing workers who would be laid off, setting up the required international arrangements for management and transport, additional spent fuel storage, and the like.

would probably be smaller than that for other options, since the WPu would displace commercial plutonium that would be used in any case. As with other options, there would be some ES&H risks involved in the processing of the pits to oxide, and steps to minimize the risks of accidents during the international shipment would be required. But there might also be some ES&H benefits: workers at MOX fabrication facilities, for example, would be exposed to lower radiation doses from WPu than they would be from reactor-grade plutonium, and adding a decade or more to the time spent fuel would be stored prior to reprocessing would reduce the radioactivity of the fuel when it was eventually processed.

Other Issues

This “substitution” option sends a variety of signals, which vary depending on the specifics of the approach. Making use of potential excess MOX capacity would mean larger-scale use of plutonium fuels than would otherwise occur, which some would oppose. Critics of the use of separated plutonium fuels might see an approach that tied disposition of WPu to continued large-scale MOX operations as irrevocably confirming MOX plans that might otherwise be canceled, and as conferring the political legitimacy of disarmament on MOX operations. Parties interested in maintaining the momentum of commercial reprocessing might view the variant involving a deferral of such reprocessing until current plutonium surpluses are consumed as a fundamental threat to the economic viability of current reprocessing plants, and thus to the plutonium fuel cycle in general (see discussion in Chapter 6, “General Considerations”).

The Spiking Option

The use of West European LWRs for the spiking option is also possible. This approach would appear to have little value, however, since there is ample capacity to burn the MOX fuel to full discharge burnups. Employing the spiking option in this approach would shift the burden of the resulting reduction in capacity factor to the West Europeans or Japanese, presumably requiring additional compensation.

The Elimination Option

West European or Japanese participation in the elimination option would require the additional utilization of West European spent fuel reprocessing facilities. Capital costs to construct commercial reprocessing facilities in the United States and the former Soviet Union would be avoided, although amortization of the capital cost of the commercial facilities would probably be included in the price charged for reprocessing services in other countries. British

and French reprocessing firms are actively seeking additional reprocessing contracts for the post-2000 period.

CURRENT-GENERATION LIQUID-METAL REACTORS

Description of Technology and Status

Current-generation liquid-metal reactors (LMRs) are sodium-cooled systems, fueled with HEU or plutonium fuels, typically with an enrichment of 20–40 percent rather than the few percent used in LWRs. There are nine such reactors in the world. The development of the LMR was initiated by the U.S. Atomic Energy Commission in the late 1940s to exploit the potential of a fast reactor to generate more fissionable plutonium, by converting U-238 to plutonium, than it burned in producing power. It was thus called a “breeder reactor.” Other countries, most notably France, Great Britain, Germany, Japan, and the Soviet Union undertook major programs in breeder-reactor development. The United States, Great Britain, and Germany have now canceled their breeder programs or scaled them back to a research phase. France is planning to convert the Superphenix, the world’s largest LMR, to a research facility, focusing on plutonium burning rather than breeding, and is continuing its LMR research and development program. Japan has just started operating a new experimental LMR, Monju, but has decided to substantially postpone deployment of commercial-scale LMRs.

The current generation of LMR that has evolved from these development programs consists of a reactor core, a primary coolant system that circulates radioactive liquid sodium through the core to an intermediate heat exchanger, and a secondary coolant circuit that circulates nonradioactive sodium to a steam generator. Sodium coolant is used because of its excellent heat transfer properties and because it permits the reactor to operate in a fast-neutron spectrum—since it is not an effective moderator of neutrons. There are two types of cooling circuit configurations: (1) the loop type, in which the sodium coolant circuit is made of piping external to the reactor vessel; and (2) the pool type, in which the core and intermediate heat exchanger is placed in a large sodium-filled vessel (the “pool”), and the secondary sodium piped from the intermediate heat exchanger to an external steam generator.

The reactor core is typically made up of assemblies of oxide fuel rods—stainless steel tubes containing pellets of the oxide fuel. Some of the rods, called the driver fuel, contain either MOX pellets or medium-enriched uranium pellets, and others, the fertile fuel, contain natural uranium oxide pellets. Altering the location and relative loadings of these fuels can change the conversion ratio (the ratio of fissile material consumed to fissile material produced). Fuel burnup in the range of 100,000–150,000 MWd/MTHM is desirable to hold down repro-

cessing requirements. This burnup level has been demonstrated in test reactors, and is well above the requirements for WPu disposition to meet the spent fuel standard. The oxide fuel has proven reliable through extensive verification tests and substantial reactor operations.

Because of the need to avoid reactions between the sodium coolant and water or steam, the reactor and coolant circuits are enclosed in inerted compartments. The system operates at low pressure, making it easier to protect against loss of sodium cooling than a high-pressure system such as the LWR. Extensive testing has demonstrated that the system can be controlled effectively in a fast-neutron cycle, principally because of the negative Doppler coefficient of reactivity of the fuel.³² A variety of technical problems have been encountered in several countries, however, including sodium leaks and fires in some cases, and unexplained shifts in core reactivity in others. These have contributed to poor availability records for some LMRs, and to the worldwide delay in commercializing LMR systems.

Recycling of the plutonium has been intended for the LMR concept from the start, in order to utilize the additional fuel produced. Thus, the capability to accept MOX fuel assemblies is already a feature of the LMR concept. LMRs operating in the once-through mode instead of the recycle mode can, therefore, convert separated WPu into spent fuel. The mix of elements in the spent LMR fuel would be similar to that of spent LWR fuel, except that the fraction of plutonium remaining would be higher (because of the higher initial fissile loading), and the remaining plutonium would be closer to weapons-grade (because of the neutron-absorption properties of plutonium in a fast spectrum). The fuel could be disposed of in the same geologic repository to be used for LWR fuel (though a separate license for the fuel as an acceptable waste form for disposal would be required). The proliferation resistance of the end product would in most respects be similar to spent LWR fuel. The existing LMRs have predominantly operated in this once-through mode, so this would not be a new approach.

If desired, the LMR core mix of fissile and fertile material and reflector material can be arranged so that the ratio of fissile material created to fissile material consumed is much less than unity. This characteristic would allow the

³² The Doppler coefficient of reactivity refers to a phenomenon in which heating the nuclear fuel increases both the fission and nonfission capture of neutrons associated with "resonances" in the relevant cross-sections. Resonances are sharp increases in the magnitudes of the cross-sections that occur for specific narrow bands of incident-neutron energy. Heating the fuel imparts thermal motion to the "target" nuclei (both fissile and nonfissile neutron absorbers), and the interaction of the resulting range of target velocities with the neutron-energy spectrum has the effect of broadening the absorption resonances—that is, a wider range of neutron speeds falls into the resonance region, so more neutrons are absorbed. Depending on the relative abundances of fissile and nonfissile materials in the fuel, and the relative magnitudes of the resonances in the relevant fission and nonfission capture cross-sections, the Doppler effect of fuel heating may either increase or decrease reactivity. In typical fast reactors, the effect is a decrease—that is, the Doppler coefficient of reactivity is negative.

LMR to be operated in the recycle mode as a net destroyer of the plutonium (in order to pursue the elimination option).

Reactor Capacity and Throughput: U.S. LMRs

LMRs have higher fissile loadings than do equal-capacity LWRs. The LMR capacity necessary to implement the spent fuel option on 50 tons of U.S. WPu over a 25-year period is little more than 1 gigawatt-electric (GWe).

The United States has two operable LMRs: the Experimental Breeder Reactor-II (EBR-II), located at the National Reactor Testing Station near Idaho Falls, Idaho, and the Fast Flux Test Facility (FFTF), located at the DOE complex in Hanford, Washington. EBR-II has a capacity of only 16.5 MWe and, therefore, could not make a substantial contribution to the disposition of the WPu. In early 1994, the Clinton administration decided to shut down the EBR-II as part of a broader decision to cancel U.S. support for advanced LMR research.

The FFTF does not have a steam generator, but instead transfers the heat from its secondary circuit to an air-dump heat exchanger. It was built to test LMR fuel and gain operating experience with a loop-type LMR system. The once-through mode that would be necessary for either the spent fuel option or the reactor-spiking option has been FFTF's operating mode to date. FFTF operated with very high reliability until the early 1990s, when reactor operations were stopped for budgetary reasons. A return to power would be practicable if a decision was made to do so, however.

FFTF has a capacity of 400 MWt, which also falls well short of the roughly 1 GWe needed for a U.S. disposition campaign. FFTF is big enough to perform the reactor-spiking option, however. A recent study by Westinghouse Hanford indicates that FFTF could spike 50 tons of WPu in less than 25 years. No significant reactor modifications are indicated to be necessary. The reactor-spiking option is only an interim solution, however, so a plan to use FFTF for reactor spiking would have to include the intended permanent solution, for which FFTF could make only a small contribution. In addition, FFTF produces no electricity, and hence would produce no revenue to offset the substantial costs of MOX fabrication and reactor operation.

FFTF is located on the same site as the incomplete FMEF MOX fabrication facility described in the first section of this chapter. FMEF could be adapted to fabricate assemblies at a rate that would support the above disposition of 50 tons of WPu in less than 25 years.

Reactor Capacity and Throughput: Foreign LMRs

There are six other operable LMRs in the world with capacities in excess of 100 MWe each. However, the three smallest have operated for roughly 20 years and, therefore, cannot be expected to make much contribution to the disposition

of the WPu. Interest therefore focuses on the three newest and largest, the French Superphenix, the Russian BN-600, and the Japanese Monju.

The French Superphenix is a 1,200-MWe plant, located at Bouvesse and operated intermittently since 1986. Its poor availability record was initially due to a leaking sodium storage tank and more recently to regulatory requirements to increase sodium fire protection and to evaluate a reactivity anomaly experienced at Superphenix's forerunner, the 233-MWe Phenix experimental LMR. These problems caused such an extended shutdown of Superphenix that it lost its license. After a prolonged relicensing process, the reactor was restarted in 1994. France now plans to use Superphenix primarily as a research facility to study actinide burning. If France's current plan were changed, Superphenix is just big enough, by itself, to implement the spent fuel option on 50 tons of U.S. WPu over a 25-year period, if it operated with much higher availability than it has in the past. France also has sufficient MOX fuel fabrication capacity to support use of Superphenix for this mission. Superphenix's past record gives little basis for confidence in future performance, however, and shipping WPu to this facility does not appear to have any major advantages over the more general substitution approach described above.

The Russian BN-600 is a 560-MWe plant, located near Yekaterinburg and operated since 1981. To date it has been fueled primarily with HEU, not MOX, although MOX fuel assemblies (including some WPu assemblies) have been tested in this reactor. MINATOM officials report that "a complete conversion of these reactors [the BN-600 and the older and smaller BN-350] to MOX fuel is not possible owing to their design and physical features" (Bibilashvili and Reshetnikov 1993, p. 32). If the BN-600 could be fully converted to MOX, its capacity is big enough for the reactor-spiking option, but it is only half that necessary to implement the spent fuel option on the nominal 50 tons of Russian WPu over a 25-year period. It is questionable whether the BN-600 could operate safely over that long a period. The BN-600's predecessor, the BN-350, was initially plagued by sodium leaks and fires; reports of a substantial sodium leak and resulting fire at the BN-600 in 1993 suggest that this problem has not been entirely resolved. A larger plant, BN-800, a 750-MWe pool-type, is contemplated, but construction of the two lead units has been halted for some years (see description in section "Advanced Liquid-Metal Reactors" below).

A new experimental 280-MWe LMR, Monju, went critical at Tsuruga, Japan, in the spring of 1994. Although Monju has its whole operable life ahead of it, the small capacity would allow this reactor to implement the spent fuel option on only 10–15 tons of U.S. or Soviet WPu over a 20-year period. Monju is big enough to implement the reactor-spiking option on either the U.S. or the Russian WPu, but all the issues described above for the more general case of substituting WPu for civilian plutonium would apply in this case as well.

In short, to carry out the spent fuel option, unless Superphenix achieved high availability, essentially the total world capacity of presently operable

LMRs would be needed. Yet many of these reactors are not presently operating and the largest plant, Superphenix, had a poor availability record when it was operating. With this experience, it would not seem to be prudent to count on these systems to provide the necessary capability. Transport of large quantities of WPu outside the United States and Russia would be required. The available capacity is even less adequate to implement the "elimination" option, for which LMRs with repeated recycle might be a candidate. In short, the panel does not recommend that the use of existing LMRs be pursued further as a major option for disposition of excess WPu.

CURRENT NAVAL AND RESEARCH REACTORS

All nuclear reactors on U.S. Navy ships use HEU fuel.³³ Some Russian naval reactors also use HEU fuel. In principle, one could imagine that these HEU-fueled reactors might be modified to use plutonium fuel instead. Because somewhat more information is available, we focus on the U.S. case.

There are 158 operating U.S. naval reactors. All are low-power reactors: the average power is about 150 MWt. Naval reactor cores have long lives, with new cores planned to last the life of the ship or, at least, for decades. For example, the cores on the recently refueled aircraft carrier Enterprise are expected to last 20 years and those for the Nimitz are expected to last at least that long (Schmitt 1993b). The average use factor is only 10 percent of rated power, and naval reactor-core endurance (in full power days) and power density (in kilowatts/liter) are comparable to those of commercial reactors (Schmitt 1994). However, naval reactor cores withstand a much longer time of high temperature operation and orders of magnitude more large power transients than do commercial reactors. In addition, naval reactor cores must withstand more than 10 times the shock loading of commercial reactors.

The fuel fabrication facility for U.S. naval reactor fuel is being decommissioned because fuel is now available to support fabrication of cores through about 2001. A fabrication facility for naval fuel will not be needed until then, and the Navy expects to seek bids later this decade for such a facility. HEU is already available to cover projected requirements until about 2006. The Navy expects that beyond this time, HEU will come from U.S. weapons dismantlement (Schmitt 1993c). Given the uncertainty in out-year projections of the defense budget, future naval vessel construction and, therefore, the rate at which new reactor cores would be needed, are quite uncertain.

Since naval reactors do not now use plutonium-based fuels, any introduction of such fuels would require substantial development, not unlike that which has occurred over the last several decades in the case of plutonium fuels for commercial LWRs. Because of the extremely high reliability requirements for

³³ The information provided on U.S. naval reactors is solely from unclassified material.

naval fuel, the development time probably would be longer than in the case of developing a new commercial fuel type. The Office of Naval Reactors estimates it would take 20 years to qualify a new fuel system, develop new core designs, and develop the fuel manufacturing facility to handle plutonium (Schmitt 1993a). Although this may be an overestimate, because of the obvious reluctance of the Navy to plan to use plutonium in its cores, the required time is not likely to be substantially less.

According to the Office of Naval Reactors, the U-235 burnup per year is approximately 1.1 tons in the entire operating fleet and, therefore, about the same quantity of plutonium could be burned if the entire fleet were converted to plutonium fuel cores (a remote possibility). A larger amount of plutonium would be irradiated to the spent fuel standard each year, however. Since the Navy operation is based on extremely long-lived cores, any change to a shorter refueling cycle (so as to remove plutonium when it was sufficiently irradiated to be self-protecting, increasing the disposition rate) would require major restructuring of the Navy logistics and operations cycles.

In short, the use of naval reactors is not a promising option for plutonium disposition. One aspect of naval reactor experience that might be of use for U.S. plutonium consumption, however, is the technology that has been developed to produce very high reliability fuel. This technology may be of use if the United States chooses to develop MOX or plutonium cores for use in commercial reactors, or commercial-type reactors, for the consumption of the WPu. In particular, the naval reactor developments may be of use if the "elimination" option is selected. The technology is classified and, consequently, a high-level decision would be required for the technology to be usable in the commercial world or shared with Russia.

Current research reactors, like naval reactors, do not offer an attractive option for the disposition of WPu. Research reactors are generally small in capacity and in duty factor, they refuel only rarely or not at all, most were not designed for the use of plutonium fuel, and they are highly dispersed geographically and often located in institutional settings that would be difficult to safeguard. Given the availability of more practical possibilities, research reactors do not deserve serious consideration for the disposition mission.

ADVANCED LIGHT-WATER REACTORS

Description of Technology and Status

Advanced light-water reactor (ALWR) nuclear plants are being developed in the United States and other countries to meet future baseload electric-generation capacity needs. One or more ALWRs could be built for the plutonium disposition mission, although in general this option would appear to have higher initial capital costs and longer time-lines than the use of existing or partly

completed LWRs. A summary of these ALWR development programs, and the schedules set by the developers, is given below, followed by a discussion of the applicability of ALWRs to this WPU disposition mission.

Two main classes of follow-on ALWRs are under development. One approach is evolutionary, with designs conceptually similar to those of present LWRs, but with substantial improvements in safety, reliability, and cost, derived from experience with the current generation of plants. These plant designs tend to be in the 1,300- to 1,500-MWe power range. Another approach involves a greater technological reach, in order to emphasize the concept of passive safety. These plants tend to be smaller, in the 600-MWe range.

U.S. organizations are developing three evolutionary designs with power output of 1300 MWe: an advanced boiling water reactor (ABWR), being developed by General Electric (GE) (Redding and McGregor 1993); a pressurized-water reactor (PWR) known as the System-80+, being developed by ABB-Combustion Engineering (Turk and Matzie 1992); and another advanced pressurized-water reactor (APWR) being developed by Westinghouse, with designs of both 1,050 MWe and 1,300 MWe (*Nuclear News* 1992b). In the same power class, the French are developing a 1,450-MWe PWR, called N-4 (NEI 1985), and are collaborating with the Germans (Framatome and Siemens) to develop another 1,450-MWe PWR, called EPR (Baumgartl and Watteau 1994), for export. Great Britain is constructing a 1,300-MWe PWR, called Sizewell B (NEI 1988). The Japanese are constructing two 1,300-MWe GE-ABWRs.

In these evolutionary plants, safety is improved by: (1) reducing the probability of core-damage accidents by incorporating additional passive features and modern control-room technology that reduce the burden on the operator in handling abnormal conditions; and (2) improving severe accident mitigation through more rugged decay-heat removal systems and containment, and passive containment temperature and pressure control. Reliability is increased by utilizing improved component design and materials derived from operating experience. Costs are projected to be reduced by simplified configurations and shorter construction time through automated, integrated management systems and modern modular construction processes.

The fuel and fuel reloading systems are essentially identical to those in the present plants, since experience with LWR fuel has been quite favorable. The ABWR and the System-80+ were designed to be able to utilize a full-MOX core. Most of the other designs could handle a one-third MOX core without significant modifications, might be able to utilize a full MOX core without such modifications, and certainly could be modified for full-MOX if that proved necessary. Other evolutionary designs could be modified to do so (GE 1994, Westinghouse 1994).

Two ABWR plants have been authorized in Japan and are in the early stage of construction, with completion scheduled for 1997–1998. A forerunner of the System-80+—a System-80 with a number of the 80+ features incorporated—is

under construction in Korea, scheduled for completion in 1996. The System-80+, the Framatome, and the Sizewell B machines are being offered in competition for construction of a reactor in Taiwan. Both the ABWR and the System 80+ designs have been submitted to the NRC. Final design approval was given in 1994, and design certification is expected in 1996. The design of the APWR is being finalized in Japan. Three N-4 plants are under construction in France and are scheduled for completion by 1995–1997. The Sizewell B plant is under construction in Great Britain and is scheduled for completion in 1994.

Designs of smaller-size plants are also under development. In the United States, three 600-MWe designs are underway: AP-600 (Westinghouse) (Bruschi and Andersen 1991), the Simplified Boiling-Water Reactor (SBWR) (GE) (Redding and McGregor 1993), and the Safe Integral Reactor (SIR) (ABB-CE 1989). These systems utilize passive emergency cooling features instead of active, high-powered equipment for that purpose. Their designers argue that this will lead to improved safety, greater simplification and economy, and reduced demand for emergency response by the operators. Improved safety, reliability, and cost are also achieved through changes similar to those for the larger plants outlined above. Although the power-generating portions of the designs are conceptually the same as in the current generation of LWRs, the emergency cooling features need extensive testing and represent a new technical element in the licensing process. Both the AP-600 and the SBWR designs have been submitted to NRC for certification, and final design approvals are expected in 1996 and 1997—three years behind the evolutionary plants.

The Japanese are developing a larger unit, nominally 900-MWe output, called the Simplified Pressurized-Water Reactor (SPWR) (Sako et al. 1992), with passive safety features patterned after the AP-600. The design is at the conceptual level, and plans have not yet been formulated for detailed design and licensing. A 600-MWe ALWR with passive safety features, the VPBER-600 (Mitenkov et al. 1992), is at an early stage of development in Russia. There is no schedule for authorization or construction.

A passive ALWR design is also being developed in Sweden (ABB-Atom), called PIUS (Process Inherent Ultimate Safety) (*Nuclear News* 1992b). NRC has completed a limited portion of a pre-application review of PIUS, deferring further review until an application for design certification is received. PIUS utilizes passive safety features to a much greater degree, involving the power generation functions as well as the emergency cooling functions. This system is at an early stage of development and has limited funding, so that the time horizon for deployment is estimated to be 2010 and later. Systems similar to PIUS, at an even earlier conceptual phase, are being developed in Japan.

In general, total electricity costs for these smaller passive systems are estimated to be somewhat higher than for the large evolutionary ALWRs, and there is greater uncertainty in the cost estimate and schedule because of the newer

features in these plant designs. The licensing of the plants will probably be more time-consuming because of the newer plant design features.

Reactor Throughput, Once-Through Fuel Cycle

The reactor throughputs for ALWRs would be similar to those described above for the case of LWRs of existing design. New systems built for plutonium disposition, however, could be built from the outset with the features needed to accommodate full-MOX cores with high plutonium loadings. One of the larger class of ALWRs, using fuel with an enrichment in the neighborhood of 6–7 percent plutonium (with burble poisons), operating with a 75-percent capacity factor and an average burnup of 42,000 MWd/MTHM, could process all of the nominal 50 tons of excess WPu in 30 years. Two of the smaller class of LWRs, amounting to a comparable total energy output, would be required to process a comparable amount of plutonium over a comparable period.

ALWRs, being in the yet-to-be-built category, could not be made available to receive fuel containing WPu as early as could existing reactors. Even if high national priority were given to building them, they would not become available in less than 10 years. The delay in potential initial fuel-loading date in that case would be less than 10 years, however, because even existing reactors would need an adequately sized MOX fuel-assembly fabrication capability to be made available, and a lengthy approval and licensing process would be involved.

Fuel Fabrication

The alternatives for fuel fabrication for ALWRs are the same as those described above for LWRs of existing design.

Approvals and Licenses

Because the ALWR designs have not yet been approved by the NRC, there would be some additional licensing uncertainty compared to the use of LWRs of existing designs. The safety and environmental impact of the ALWRs, however, has received major emphasis in their design, so that they should be measurably safer than the current-generation U.S. LWRs. The major regulatory issues in the United States would be the passive safety features of the smaller plants and the utilization of these designs for processing WPu. Construction of any new nuclear-power plants in the United States, however, would be likely to raise public controversies. For the spent fuel option, reprocessing would not be required, and the existing French commercial implementation of plutonium recycle and the limited earlier demonstration in the United States could provide an experience base for this mission in ALWRs.

The above considerations apply to all ALWRs, but further evaluation requires distinguishing the different ALWR design concepts. The large ALWRs

can be classified as mature designs, with some models already under construction overseas. They have met licensing requirements (with LEU fueling) in Japan and Great Britain, and they are nearing the end of the arduous process of NRC certification with LEU loading. The more advanced passive designs, by contrast, have considerably further to go in the licensing process, and therefore face greater licensing uncertainty. NRC or Defense Nuclear Facilities Safety Board approval of the MOX fuel fabrication facility will also be a challenge.

Earlier regulatory reviews of reactor-grade plutonium recycle in LWRs, such as the Generic Environmental Statement on Mixed Oxide fuel (GESMO), provide a significant technical foundation for the licensing process, but that work was halted more than 15 years ago and focused on the one-third MOX loading option. The regulatory reviews required for the ALWR option would probably focus on full-MOX loadings and the impact of changes in safety and environmental rules since the GESMO reviews ended. The use of erbium oxide as a burnable poison to increase plutonium loading would add another new feature in licensing.

Despite these uncertainties, one possible advantage of construction of a new ALWR for this mission over use of an existing LWR is that the new system could be built on an existing government site (presumably co-located with other steps in the disposition process, such as fuel fabrication and pit processing). This could reduce problems of public acceptance, transport, and handling of plutonium at multiple sites, and the like.³⁴ Whether the overall approval process would be more uncertain and longer for ALWRs or for existing LWRs depends in significant part on the specific sites, reactor designs, and politics involved. In either case, proposals for the use of plutonium in U.S. reactors can be expected to engender a significant debate.

Safeguards, Security, and Recoverability

The issues here would be the same as those described for the case of LWRs of existing design. As just noted, ALWRs built for this mission would presumably be based at a single government site, minimizing transportation risks.

Cost

The initial capital cost of building a new reactor would inevitably be higher than the cost of utilizing existing reactors or finishing partly completed facilities. Some private vendors, however, are proposing concepts in which the initial capital cost would be covered by the private sector, in return for a guarantee of an annual plutonium disposition fee from the government sufficient to make the

³⁴ We note, however, that the WNP-1 and WNP-2 reactors mentioned above are already on a U.S. government plutonium-handling site (the Hanford reservation), which is a site with a nearly completed MOX fuel fabrication facility as well.

electricity from the reactors competitive with other sources of electricity in the area where the reactor was deployed. In general, it appears that the overall net discounted present cost of new reactor approaches would be somewhat higher than existing reactor approaches (because of the higher cost of nuclear power compared to other generating options in the projected U.S. market over the next 40 years), but there is considerable uncertainty in these calculations. (For a detailed economic comparison, see Chapter 6.)

Environment, Safety, and Health

As noted above, the ALWRs are designed to be measurably safer than current-generation LWRs. Current LWRs in the United States, however, are believed to be acceptably safe, and as noted above, if appropriate modifications were implemented, use of full-MOX cores should not reduce existing safety margins. (For a detailed discussion, see Chapter 6.)

Other Issues

The perceived signals relating to fuel-cycle policy in this case would likely be roughly the same as for LWRs of existing design.

The Spiking Option

The spiking option puts high priority on early denaturing of WPU, and therefore does not fit well with alternatives such as the ALWR, which require building new reactors for that purpose. The utilization of existing LWRs for spiking could start the disposition campaign sooner, with a campaign period no longer than one using ALWRs.

The Elimination Option

The characteristics for this option in ALWRs would not differ from those described above for current LWRs.

ADVANCED LIQUID-METAL REACTORS

Description of Technology and Status

Three alternative designs of advanced liquid-metal reactors (ALMRs) are under development outside the former Soviet Union, and another has been developed there. All of these reactors are inherently designed to utilize plutonium fuels and thus are obvious candidates for plutonium disposition. Three utilize the current MOX fuel, and one is based on a metal alloy of uranium, plutonium, and zirconium. The three oxide fuel versions are being developed in Western Europe, Japan, and Russia, and the metal fuel version in the United States. Re-

processing and recycle of plutonium is an inherent part of the operating concept of these reactors. Such reprocessing and recycle is applicable to the elimination option, but not to the spent fuel option. If operated in a once-through mode, however, ALMRs could be used to transform WPU into spent fuel, just as other reactors could. The capital costs of these liquid-metal reactor concepts are expected to be higher than those of LWRs, however, and they are much farther from being licensed in the United States than are evolutionary ALWRs. Hence these reactors are of greater interest for the long-term elimination option than for the spent fuel option.

The U.S. design is being developed by General Electric, under the sponsorship of the U.S. Department of Energy, and is based on the metal-alloy fuel pin design being developed by Argonne National Laboratory in Illinois.³⁵ In 1994, government support for development of this reactor concept was canceled, but it could be restarted if this concept were chosen as a plutonium disposition option or a future electricity source. The overall concept, called the Integral Fast Reactor (IFR) (Till and Chang 1988), is a significant advancement over current LMR designs. The plant design is based on the pool configuration described above in the section "Current-Generation Liquid-Metal Reactors."

The use of metal fuel results in lower reactor outlet temperature (500° C), steam temperature (430° C), and net plant efficiency (36 percent) compared to the other two ALMR alternatives, but it is projected that this loss in efficiency will be balanced by superior fuel-cycle economics. A smaller reactor power output, about 303 MWe, permits the incorporation of a passive decay-heat removal system. Each reactor drives a single steam generator, and two reactor/steam generator units would be coupled through their steam loops to drive an economically sized, 606-MWe (net), turbine.

The most significant advance in the U.S. IFR program is a pyroprocessing approach intended to significantly reduce the costs, wastes, and proliferation risks of reprocessing (compared to the conventional aqueous process used with oxide fuel).³⁶ In this integrated reprocessing approach, the plutonium is never fully separated in a form that could be used directly in nuclear weapons, thereby reducing safeguards concerns.³⁷ The metal fuel lends itself to this pyroprocess-

³⁵ General Electric uses the name PRISM for its reactor design for the ALMR program.

³⁶ Till and Chang (1990). The most complete available assessment of the proliferation issues involved in the IFR fuel cycle is Wymer et al. (1992). For an independent assessment of the state of development of the pyroprocessing system and the challenges remaining, see National Research Council (forthcoming). See also OTA (1994).

³⁷ This approach would mitigate concerns regarding theft of plutonium or covert diversion of material under safeguards. Possession of such a facility, however, would still offer a country the technology needed to produce separated plutonium for weapons, should it choose to do so openly. As the United States and Russia already possess large nuclear arsenals, this is not a special concern in the context of this report. It must be remembered in considering the potential implications of a worldwide breeder economy employing such technologies, however.

ing approach. To further increase proliferation resistance, the power plant, reprocessing facility, and MOX fabrication facility are co-located.

The design employs modular factory fabrication to minimize capital cost in an effort to offset the loss of economy of scale. Other distinctive features include seismic isolators supporting the nuclear steam supply system and electromagnetic pumps driving the primary sodium. A comprehensive test program has been defined for demonstrating the newest features of the plant design. A test program has also been underway for several years at the Argonne National Laboratory to demonstrate the performance characteristics of the metal fuel pin and of the pyroprocess that would be used for processing the spent LMR fuel in the recycle mode that ALMRs would operate in commercial deployments. Licensability reviews have been completed favorably with the NRC. NRC design certification has not yet started; prior to program cancellation, design certification was scheduled to be completed by 2005. Construction of a full-size prototype reactor co-located with a reprocessing and fabrication facility was planned if the demonstration tests were successfully concluded and after completion of preliminary and final design.

The West European design, known as the European Fast Reactor (EFR) (Ebbinghaus et al. 1992), is the result of a collaborative effort, mainly between France, Germany, and Great Britain. Here, too, government decisions have put the program's future in doubt, as both Britain and Germany have withdrawn their support. The design is for a large, 1,450-MWe plant with the pool configuration, of the same concept as the current generation of oxide-fueled LMRs, incorporating substantial improvements in safety, reliability, and economy derived from construction and operation of Western Europe's existing LMRs. The newest features of the design have been the subject of a comprehensive test program. The oxide fuel assemblies are to have an in-core residence time of six years and to achieve a peak burnup of 200,000 MWd/MTHM. The core is 1 meter high and 4 meters in diameter. The large-capacity primary sodium circuit drives six parallel secondary sodium circuits each containing a steam generator. A reactor outlet temperature of 545° C permits a steam temperature of 490° C and a resulting net plant efficiency of 40 percent.

The Japanese design, known as the Demonstration Fast-Breeder Reactor (DFBR) (Miura et al. 1992) is for a medium-size, 670-MWe plant. The Japanese government has recently indicated that construction of such a commercial-scale breeder will now be postponed for a decade or decades. The DFBR is a sodium loop design. The loop components are interconnected by top entry piping that projects down into free sodium surfaces in the components, thereby precluding a siphoning risk that would otherwise exist with the loop configuration. This medium-capacity primary sodium circuit drives three parallel secondary sodium circuits, each containing a steam generator. The reactor outlet temperature, steam temperature, net plant efficiency, and fuel burnup are slightly higher than those described above for the EFR. The details of the DFBR design are substan-

tially influenced by the high seismicity of Japan. The newest features of this design also have been the subject of a comprehensive test program. In addition, some Japanese officials have suggested that an international group fund a special-purpose LMR to be built in Russia to consume excess WPu. The issues associated with this concept are similar to those of other ALMR concepts, described in this section.

The Soviet Union also designed a follow-on LMR, known as the BN-800 (Minkov et al. 1990). MINATOM hopes to build four of these 750-MWe plants, three at the Mayak complex at Chelyabinsk. Construction on two of these plants started in the 1980s, but has been stopped for several years because of lack of funds. The BN-800 is an important special case of follow-on LMRs, because use of plutonium fuel in these reactors is MINATOM's preferred option for disposition of both excess WPu and civilian separated plutonium (now building up in storage as a result of continued reprocessing at the Mayak site) (Mikhailov et al. 1994). It is therefore worth describing this design in some detail.

The BN-800 design is similar in many respects to the BN-600, described above. The major BN-800 reactor components are of the same size and comparable in design to those of the BN-600 reactor, with the major differences being in the secondary systems. The most significant difference is the intended fuel. The BN-800 reactor is designed to use a full core of MOX fuel, while the BN-600 uses primarily uranium oxide, and, as mentioned above, is described by MINATOM as being incapable of shifting to full-MOX fueling. Raising the output of the reactor design from 560 to 750 MWe required an increase of the core volume; this was accomplished by increasing the core height from 0.75 to 0.95 meters, and by increasing the number of fuel assemblies, which increased the core diameter from 2.05 meters to approximately 2.25 meters. Despite the increase in core diameter, the radial dimensions of the central section of the reactor were kept unchanged, thanks to a reduction of the thickness of the radiation shield and a decrease in the thickness of the radial blanket. The BN-800 core is designed to have three radial zones of differing enrichment, one more than in the BN-600 core.

Some important changes were made in the reactor safety provisions, resulting in part from new guidelines on nuclear safety adopted by the Soviet Union in 1982. The most important changes were:

- Increasing the number of control rods from 27 to 30, making it possible to create two independent and redundant reactivity control systems.
- Introducing technical means (based on acoustic and neutron noise) to detect the onset of sodium boiling (which could lead to overheating because of the positive sodium void coefficient).
- Providing operational guidance and procedures to the operator on what to do to prevent accident development.

- Placing the entire equipment of the primary and secondary circuits on a monolithic concrete slab to improve seismic safety. This includes the steam generators, the refueling equipment, the auxiliary systems for operating the components with liquid metal, the heat transport equipment for the primary and secondary circuits, and the safety subsystems.
- Adding a new emergency cooling system involving auxiliary sodium-air heat exchangers connected in parallel with the steam generators in each secondary circuit. This system provides for emergency residual heat removal during accidents initiated by the loss of power supplied to the reactor subsystems or the loss of feedwater (for example, in the case of a rupture in the water-steam equipment or piping). In all other accidents, the emergency cooling of the reactor is carried on through the steam generators.
- Adding a redundant standby control room allowing the reactor to be shut down, and its main neutronic, thermal, and fire-safety conditions to be monitored, in the event the main control room becomes unusable.
- Setting up the reactor scram-rod system as two independent subsystems, each being able to actuate independently the insertion of all the rods by the force of gravity. The independence of the two scram-rod subsystems is ensured by placing their components in different rooms, by laying communication lines along different cable passages, and by connecting each of the two subsystems to a different power supply source. Each subsystem operates on three independent information channels.

Thus the safety of the BN-800 design is significantly better than that of the BN-600 that is now operational. MINATOM officials report that the design has received licensing approval. Whether the design incorporates adequate redundancy in its safety features to meet international safety standards, however, requires additional information and further analysis.

In addition to this sodium-cooled design, Russian institutes are carrying out conceptual design studies of ALMRs cooled by molten lead, or a molten lead-bismuth mixture, rather than molten sodium. Conceptual designs have been developed for both a 300-MWe modular unit patterned after the PRISM-IFR (Gleukler and Quinn 1994) and a 1,000-MWe unit patterned after the evolutionary ALMRs. A significant advantage of the lead-cooled reactor is the elimination of the risk of sodium leaks and fires, since lead is inert to both air and water. This allows the secondary coolant loop and the intermediate heat exchanger to be eliminated, simplifying the system. Lead has heat transfer properties that are superior to those of sodium, including a larger margin between the operating temperature and coolant boiling temperature, increased natural convection potential, and a larger system thermal inertia. The proposed fuel is a high-density mixture of uranium and plutonium mononitrides (PuN-UN) with a ferritic steel

cladding. The gap between the cladding and fuel is filled with molten lead to reduce fuel-clad interaction and increase heat transfer.

This lead-cooled ALMR is still in the conceptual design phase. Corrosion and structural properties of the plant material will have to be defined and adequate reliability demonstrated. The proposed nitride fuel form will require extensive testing and qualification. It is highly unlikely that this concept could be developed and deployed in less than two to three decades.

Reactor Throughput, Once-Through Fuel Cycle

As described in the above section "Current-Generation Liquid-Metal Reactors," the ALMR capacity necessary to implement the once-through spent fuel option would be about 2 GWe to process a combined U.S./former Soviet Union total of 100 tons of WPu over 25 years. The precise capacity necessary would depend mainly on the reactor design and on the degree of fuel burnup necessary to achieve the desired level of proliferation resistance.

ALMRs, being in the yet-to-be-built category, could not be made available to process WPu as early as could existing reactors, and the costs and uncertainties involved in their use would be higher. Capital costs in the billions of dollars would be needed to provide the capacity to process 100 tons of WPu in the spent fuel option. For ALMR designs such as the IFR, a prototype plant would probably have to be built first. In the United States, prospects for gaining approval for construction of a new LMR must be judged to be highly uncertain. Even once the requisite licensing and political approvals had been gained, a new LMR would take a minimum of 10 years to complete in a "crash" program. An ALMR design such as the IFR would take longer.

The time required to complete the BN-800 reactors in Russia is even more difficult to predict. The fundamental barrier to completion of these reactors is lack of available funds; safety reviews and the need to gain political and licensing approval could also delay completion. Particularly given the substantial supplies of low-cost uranium and enrichment services available in Russia, the electricity cost from reactors of this type is likely to be significantly higher than the cost of electricity produced by LEU-fueled LWRs, or the cost of other sources of electricity; in the current economic environment in Russia, such a large-scale subsidy may be difficult for MINATOM to justify. Completion of these reactors in the near term appears unlikely in the absence of substantial Western assistance. Because of the decline in industrial production (which will probably not recover completely for a substantial period), moreover, electricity demand has declined, and it is by no means clear that new electrical capacity is needed in the areas planned for these reactors. Because of factors such as these, some top MINATOM officials have acknowledged that the first BN-800 is un-

likely to be operational for at least 10–15 years.³⁸ Even this estimate appears optimistic; it is difficult to rely on the availability of these facilities on any set schedule. Thus the BN-800 does not appear to meet the important criterion of minimizing the total time before disposition of plutonium could be accomplished.

We note that in saying it may not be desirable to wait for the BN-800s for plutonium disposition, we are neither endorsing nor rejecting the BN-800s as an option for Russia's future energy supply. The 30 tons of civilian separated plutonium already in stock is more than sufficient to operate the four BN-800s, regardless of what is done with the WPu: each reactor only requires 2.3 tons of plutonium as startup fuel (meaning that less than 10 tons is required to start all four that might be built), and each produces more plutonium than it consumes thereafter.³⁹ To use both the 30 tons of civilian separated plutonium and the nominal 50 tons of excess military plutonium in these reactors would mean continuing to add more fresh plutonium as spent fuel is removed, rather than allowing the reactors to fuel themselves through reprocessing and recycle of the plutonium they produce, as they were designed to do. The net result would simply be a much larger quantity of spent fuel awaiting reprocessing in the fuel cycle for these reactors.

Fuel Fabrication

Given the long schedules for providing ALMR reactor capacity, the availability of fuel fabrication capacity would probably not be a limiting factor governing the schedule for initial fuel loading in the ALMR option. As noted above, the United States already has an incomplete LMR MOX fuel assembly fabrication facility, the FMEF at Hanford, which could be completed, modified for current safeguards and environmental standards, and used to produce fuel at a rate that would support the disposition of 50 tons of U.S. WPu in less than 25 years in the spent fuel option. A facility comparable to the U.S. facility could be provided in Russia by completing the Chelyabinsk MOX fabrication plant, which was designed to provide fuel for the BN-800s.

³⁸ Boris Nikipelov, remarks reported in Hibbs (1993). It is notable that Egorov et al. (1994, p. 2), while on the one hand supporting BN-800 deployment as the best approach to plutonium disposition, at the same time notes that "Construction of the first two BN-800 units and of Complex-300 [the associated MOX fabrication plant] has been suspended due to financial difficulties," and that "intensive deployment of BN type reactors in the nearest future has a low probability," requiring "corrections to be made in the earlier existing strategy of plutonium utilization mainly in fast breeder reactors."

³⁹ Mikhailov et al. (1994). Annual make-up fuel is 1.6 tons.

Approvals and Licenses

As noted earlier, gaining political and licensing approval for construction and operation of an ALMR in the United States would be difficult, because of the domestic politics surrounding reprocessing and fast reactors. The current government policy is to terminate the fast-reactor program. This option therefore has substantial licensing and approval uncertainties. Indeed, consideration of the institutional issues and uncertainties involved suggests strongly that the U.S. government is the only viable organization that could sponsor and fund an ALMR deployment in the United States.

In Russia, the BN-800 design is reported to have received licensing approval. The national and local politics that would surround a decision to complete the BN-800s are difficult to predict, given the fast pace of political changes in that country. It seems likely, however, that if money was available, gaining the needed approvals would be less difficult in Russia than in the United States.

Safeguards, Security, and Recoverability

The safeguards and security issues in the use of ALMRs on a once-through cycle for the spent fuel option are not fundamentally different from those for other reactor types. Because of the high fissile loading in the fuel, a smaller total amount of fuel would have to be fabricated, but each unit of that fuel would contain more plutonium. In the case of the IFR, there are options for incorporating the WPU with various minor actinides and other radioactive species in the course of fuel fabrication, which could reduce the risk of theft of the material during the process, though it may at the same time complicate accountancy and require a somewhat modified safeguarding approach (Wymer et al. 1992).

As noted in the discussion of current LMRs, the mix of elements in the spent LMR fuel would be similar to that of spent LWR fuel, except that the fraction of plutonium remaining would be higher (because of the higher initial fissile loading), and the remaining plutonium would be closer to weapons-grade (because of the nonfission capture properties of plutonium in a fast spectrum). The proliferation resistance of the end product would in most respects be similar to spent LWR fuel.

In MINATOM's concept for plutonium disposition, the plutonium in the BN-800 spent fuel would ultimately be reprocessed and reused. Thus, while the WPU would initially be embedded in highly radioactive spent fuel (as in other spent fuel options), it would then be separated again. Only a few tons would exist in separated form at any one time, however. The BN-800 as currently conceived does not incorporate the integral reprocessing approach envisioned for future U.S. LMRs and thus raises greater safeguards and security concerns.

Cost

The capital costs of ALMRs are expected to be significantly higher than the costs of comparable LWRs. Since capital charges are a large fraction of the total cost of nuclear-generated electricity, the overall electricity cost from such reactors is also likely to be higher.

Plutonium would be a less costly fuel for ALMRs than uranium (in contrast to the LWR case), because of the higher costs for uranium purchases and enrichment for their more enriched fuels. But that would not make the total subsidy required for plutonium disposition (compared to the production of electricity by other means) less than it would be in the case of LWRs.

Environment, Safety, and Health

Few of the ES&H aspects of ALMRs are fundamentally different from those of other reactor types. The ALMRs outside the former Soviet Union are designed to offer high reactor safety and significantly lower operational exposures than LWRs. Confirming the safety of the BN-800 design would require more information than is available to the panel and further analyses.

If ALMRs were operated on a once-through fuel cycle, the spent fuel would ultimately have to be disposed. LMR spent fuel has not yet been certified as an acceptable waste product for disposal—in part because these systems and their fuels were largely designed with reprocessing rather than direct disposal in mind. The higher fissile content of LMR spent fuel would lead to greater long-term criticality concerns for repository disposal than in the case of LWR spent fuel. It is doubtful whether the metal fuel used in the IFR would be an acceptable waste form for the oxidizing environment in the proposed U.S. Yucca Mountain repository. A waste package could in principle be designed to attempt to protect the fuel from the oxidizing environment, but to date the U.S. waste disposal program has avoided placing primary reliance on such engineered barriers.

Other Issues

The U.S. decision to cancel government support for the IFR program was based in part on a desire to send a signal that the United States did not support a near-term transition to the plutonium fuel cycle for which fast reactors were designed. Construction of a fast reactor for plutonium disposition would be interpreted by some as sending the opposite signal, which would be contrary to current U.S. policy. Assistance for completion of the BN-800 reactors in Russia would send a similar signal and would provide a boost to the plutonium economy there. If operated for plutonium disposition on a once-through cycle, however, ALMRs would not inherently need to raise the issue of reprocessing and recycle. (See discussion in Chapter 6, “General Considerations.”)

The Spiking Option

The spiking option puts high priority on early denaturing of WPu and therefore does not fit well with alternatives such as the ALMR, which require building new reactors for that purpose. The panel would not recommend using ALMRs for the spiking option.

The Elimination Option

LMRs, with their fast-neutron spectrum, can fission all isotopes of plutonium and are frequently put forward as a prime candidate for nearly complete plutonium elimination. Several countries are examining their potential as actinide burners. As noted above, some ALMRs, such as that being researched in the United States, employ an integral reprocessing technique in which the plutonium is never fully separated, mitigating some of the safeguards concerns that would otherwise arise from the repeated reprocessing and recycling required for the elimination option.

The fast-neutron spectrum of ALMRs requires a large fissile loading to achieve criticality, given the smaller fission cross-sections in a fast spectrum. As noted in the above section "U.S. Plutonium in Current-Generation U.S. Light-Water Reactors," after the initial period of reactor burning in an elimination option, the stock of plutonium would eventually be reduced to the point where it would no longer be sufficient to maintain criticality in the reactor. The remaining inventory would then have to be burned down exponentially by adding additional fissile material to maintain criticality in the reactor. For an ALMR, the in-core inventory that would have to be burned down in this manner is quite large, so longer times would be required to achieve very high destruction fractions than would be the case in a system with a smaller in-core inventory.⁴⁰

As with other elimination concepts, pursuing this option would require an additional large capital investment to provide reprocessing capability. Although the pyroprocessing approach proposed for the IFR might prove to require lower capital costs than aqueous reprocessing, the uncertainties of cost estimates are particularly large since all previous production experience has been with aqueous homogeneous reprocessing. There are commensurate uncertainties in when such a capability could be provided.

Many of the other concerns regarding such elimination options described in the section on U.S. light-water reactors would apply in this case as well. In the case of ALMRs using an integral reprocessing approach in which the plutonium was never fully separated, such as was proposed for the U.S. IFR program, safeguards and security concerns would be substantially reduced compared to an elimination campaign based on the PUREX process.

⁴⁰ For a more detailed discussion of an elimination campaign using LMRs, see Garwin (1995); see also National Research Council (forthcoming).

MODULAR HIGH-TEMPERATURE GAS-COOLED REACTORS

Description of Technology and Status

The high-temperature gas-cooled reactor (HTGR) is a unique thermal-reactor approach. In the HTGR (or MHTGR, as some recent modular designs are called), the fuel would be encased in tiny particles or pellets, which would provide the first line of containment against release of fission products. This fuel pellet design offers the possibility of higher burnup on a once-through cycle than other reactors. These pellets would be within a graphite structure that would serve as moderator. The whole system would be cooled by helium. The hot helium could be used to produce steam, or, in the most recent designs, to drive turbines itself, in a so-called "direct-cycle" approach. In July 1993, this less mature direct-cycle system, known as the gas-turbine modular helium reactor (GT-MHR), was chosen as the basis for further commercial development in the United States, and it is the system now proposed by the developer, General Atomics, for plutonium disposition as well.⁴¹

In the General Atomics design, the pellets would be bonded together to form fuel rods, called compacts. The compacts would be inserted into vertical holes in hexagonal graphite blocks, called elements. (In the German and Russian HTGR designs, the graphite blocks are in the form of tennis-ball-sized spheres, and the reactor can be refueled online without powering down.) The General Atomics MHTGR core is an assembly of these blocks arranged in an annular shape. The center and outer portions of the core would be made from unfueled reflector blocks. Within the core, center reflector blocks are surrounded by 102 fuel blocks, which in turn are surrounded by outer reflector blocks. Each block can be removed by a fuel-handling machine (a feature that becomes important in one of the proposed plutonium-burning modes). The helium coolant flows downward through holes in the fuel blocks to a plenum at the bottom of the core. Metal-clad, borated graphite control rods control core reactivity. The reactor core is housed in an uninsulated steel reactor vessel, approximately 24 feet in diameter and 73 feet high, which is located below ground (GA 1994).

The kernel contains enriched uranium in the commercial design. For the plutonium disposition program, it would contain a mixture of plutonium oxides corresponding to 1.61 oxygen atoms for each plutonium atom. The kernel is surrounded by successive layers of a porous carbon buffer, an inner isotropic pyrolytic carbon layer, a silicon carbide barrier coating, and an outer isotropic pyrolytic carbon layer. These four layers together are called a TRISO coating. The TRISO plutonium particles are estimated to be 0.675 mm in diameter. This

⁴¹ For general discussion of the MHTGR, see National Research Council (1992) and *Nuclear News* (1992a). For discussions of the possible MHTGR role in plutonium disposition, see GA (1992, 1993a, 1993c, 1994) and USDOE (1993).

is smaller than the particle previously planned for the version of the HTGR design developed for the New Production Reactor competition, a U.S. Department of Energy program (now canceled) to develop a reactor to produce tritium for weapons production. This particle had three additional layers, but experienced failures in testing.

The fuel designed for commercial application consisted of two types of kernels: fuel particles with 19.9-percent enriched uranium oxycarbide and fertile particles of thorium oxide (*Nuclear News* 1992a, p. 78). The plutonium-burning fuel would replace the uranium with weapons-grade plutonium. A high plutonium loading (88 percent) was used for fuel particles irradiated in the Peach Bottom MHTGR in the late 1960s and early 1970s. Performance was similar to that for HEU fuel, although sensitive to the initial oxygen-to-plutonium ratio (GA 1992, p. 4).

The MHTGR proposed by General Atomics in Phase I of DOE's study of reactors for plutonium disposition is based on gas-cooled designs that have been under development and, in some cases, in operation over the last 30 years. Two commercial plants have operated in this country: a helium-cooled 40-MWe demonstration unit at Peach Bottom, Pennsylvania, which operated from 1967–1974, and a 330-MWe plant at Fort St. Vrain, Colorado, built for commercial generation of electricity. Consistently poor performance led to the closing of that plant in 1991. A thorium high-temperature prototype reactor was run in Germany, but closed in 1989 due to technical problems.

The MHTGR design was developed initially for application as a commercial reactor to produce electricity. The plutonium burner was based on a reference design that was being developed by General Atomics as one of the two competing designs for the New Production Reactor. Both the commercial MHTGR and the plutonium-burner version used steam generators. General Atomics now has proposed that a direct Brayton cycle be used for both versions. The commercial reactor is labeled GT-MHR and the plutonium disposition reactor proposed in Phase II of DOE's study is labeled PC-MHR (plutonium-consumption modular helium reactor). This design also is the basis for a 1993 joint General Atomics/MINATOM agreement to work together to develop an HTGR for plutonium disposition, with the hope of funding for the project from the U.S. government.

The primary differences between the MHTGR proposed in Phase I of DOE's study and the PC-MHR proposed in Phase II are that the PC-MHR utilizes (Blue 1993):

- direct cycle, using the heated helium in the reactor to feed directly into a gas turbine, with inlet temperature of 849° C;
- magnetic bearings for the turbines;

- compact plate heat exchangers, leading to a significant reduction in volume and an increase in efficiency; and
- improved turbines.

With these features, General Atomics estimates that this design could achieve a very high thermal efficiency, in the neighborhood of 46–47.7 percent (Schleicher et al. 1992; GA 1994, p. 2–52).

A significant difference between the GT-MHR and the PC-MHR proposed for plutonium disposition is that the commercial version has a vented, low-pressure containment, while the PC-MHR has a high-pressure, low-leakage containment. The MHTGR that had been under review by the NRC also differed from the MHR concept in that the MHTGR was air-cooled and the MHR is water-cooled.

General Atomics concludes that a plutonium TRISO fuel particle development and fabrication program would add a year and a half to the schedule that had existed for the New Production Reactor program. Since particle behavior for the large number of particles required for HTGR cores has been the critical safety issue (National Research Council 1992), however, significantly longer than 18 months might be necessary for successful demonstration of the plutonium particle fabrication process. Since the fuel remains the same for the PC-MHR as for the Phase I concept, General Atomics' previous discussion of the fuel remains relevant. General Atomics notes that: "The critical path activity on the schedule for deployment of the PC-MHR is qualification testing and fabrication of coated plutonium fuel particles" (GA 1994, p. 13-2).

According to General Atomics, the main elements for the plutonium fuel development program are (GA 1992, pp. 19-20):

- Design of a coated plutonium fuel particle.
- Demonstration of plutonium fuel particle and compact fabrication capability, including process development, equipment design, and production scaleup.
- Model development to predict plutonium fuel performance during normal and off-normal reactor service conditions.
- Plutonium fuel irradiation performance and fission product behavior testing to validate fuel design, qualify fuel fabrication capability, and provide data for validation of fuel performance/fission product transport design methods.

In its review of the General Atomics proposal, DOE noted that six of the ten areas identified by General Atomics as needing development are related to the use of plutonium fuel. DOE concluded that the plutonium-burning HTGR technology "is not yet mature, and considerable development and testing remains to be completed" (USDOE 1993, Vol. II, p. SC 4&5-21). The Phase II report pre-

pared by General Atomics lists 22 plutonium fuel-design development needs. This report also identifies 28 facilities in the United States, Russia, France, and Japan that could in principle be used to support the design of the PC-MHR, and 7 new development and test facilities that would be required (GA 1994, pp. 3-48 to 3-57, 3-59 to 3-67).

The Lawrence Livermore National Laboratory (LLNL) examined the MHTGR as a plutonium burner and considered burnup cycles of 356,000, 560,000, and 730,000 MWd/MT plutonium, provided by General Atomics. Lack of information on the control characteristics for the pure plutonium particle cores led LLNL to examine a modified version, which would contain a neutron absorber or fertile material in the core. According to LLNL, this design would be within the current commercial design and safety envelope. LLNL concluded that this design could be deployed before 2010 if a decision was made in 1995, since the design is at a more advanced stage of development than the MHTGR plutonium burner sponsored by General Atomics. LLNL estimated the MHTGRs proposed by General Atomics, using a pure plutonium kernel and high burnup, would take an additional 5–10 years (Omberg and Walter 1993).

The principal advantage of the new PC-MHR approach, if the design holds up under detailed analysis (including safety reviews), is the possibility of lowering the cost of the plant while increasing the output, thus lowering the cost of electricity generated by the plant. While important commercially, this feature does not affect the plutonium-consumption characteristics. The MHTGR itself still has substantial licensing review ahead. This direct cycle system has much more.

Basic problems with the MHR include getting the funding for a new reactor (a problem common for any of the basic reactor approaches); developing and testing a new plutonium fuel; redesigning the core for higher burnup, in particular, if the fuel shuffling approach is to be used; and building and licensing a plutonium fuel fabrication facility. In addition, although the designs are different, any HTGR program will have to overcome utility concerns relating to the poor operational performance of the Fort St. Vrain reactor in Colorado, and the THTR (thorium high-temperature reactor) in Germany.

Reactor Throughput

The reference PC-MHR facility has 14 modules, each at 600 MWt (286 MWe).⁴² Analysis has been done on several burnup programs. The reference design with a two-year fuel exposure is estimated by General Atomics to achieve 90-percent burnup of Pu-239 and 63-percent burnup of initial total plutonium, with an average burnup of 590,000 MWd/MT plutonium (GA 1994). At

⁴² General Atomics notes that 14 modules are required if the DOE-specified 75-percent capacity factor is used. General Atomics believes that an 85-percent capacity factor can be achieved, meaning that only 12 modules would be required.

590,000 MWd/MT, using fourteen 600-MWt MHR modules, 50 tons of plutonium could be processed to 90-percent Pu-239 destruction in 25 years from project start.

The MHTGR regular fuel cycle is 24 months long, with half of the core being replaced every 12 months. To achieve higher burnup, another proposal in the General Atomics Phase I report was to move fuel blocks to the reflector area after 24 months, for an additional 12 months of exposure. Shuffling fuel blocks to reflector locations for an additional one year of irradiation was reported to lead to 97 percent burnup of Pu-239 and 73 percent burnup of total plutonium, at 677,000 MWd/MT. Similarly, the Phase II report includes a brief mention of an alternative that would have a "slightly lengthened" refueling interval and would obtain 95-percent Pu-239 destruction and 72-percent total plutonium destruction (GA 1994, p. 1-20).

These once-through burnups are substantially higher than those projected for the LWR, CANDU, or ALMR systems because of the unique pellet design of the MHTGR and use of nonfertile fuel. Nevertheless, substantial quantities of plutonium would remain in the spent fuel (see below for a discussion of the difficulty of recovering this material for use in weapons). These burnups are therefore more than is needed for the "spent fuel" option, but less than needed for a true "elimination" option, which would require reprocessing and recycle.

In its Phase I report, General Atomics examined the concept of using two cycles of reprocessing and reuse of the plutonium to produce a higher final burnup. In this General Atomics concept, 80 percent of the fuel blocks loaded each year would be fresh blocks, 15 percent of the blocks would have been recycled once, and 5 percent twice. Using a simple model, this three-pass program is estimated to achieve 99.9-percent Pu-239 burnup and 90.7-percent total plutonium burning (GA 1993b, pp. 7, 16). At the end of a six-year residence time, the average fuel burnup would be 813,000 MWd/MT plutonium (INEL 1993a, p. 23). Additional recycling would be needed to eliminate the remaining material.

LLNL estimates it would take 10–15 years for deployment of a plutonium burner, based on the commercial variant using plutonium-uranium fuel, and 15–20 years for the plutonium fuel variants. The Idaho National Engineering Laboratory estimates 10–20 years for an MHTGR deployment after decision. Given such times, the MHTGR is not competitive with current-reactor designs for the "spiking" mission, with its emphasis on irradiating the WPu as quickly as possible, nor with the utilization of current LWRs for the "spent fuel" mission.

Fuel Fabrication

As noted above, fabrication and performance of plutonium HTGR fuels are among the biggest technical hurdles to be overcome for a plutonium-burning HTGR. Unlike the previous reactors discussed, in this case there are no substantial facilities anywhere in the world with the capability to undertake this fabri-

cation on the scale required. Licensing and construction of such a facility would be expensive and time-consuming.

Approvals and Licenses

The basic MHTGR design has been available for many years, although it has not been commercially successful. The MHTGR that had been proposed both for a commercial reactor and for the New Production Reactor had been under design for at least five years, although a completed design had not been developed. The MHTGR had been undergoing licensing review at the NRC,⁴³ and the NRC has adequate experience for licensing review based on its work with Fort St. Vrain in Colorado.

The sponsor's principal concern for the commercial reactor appears to have been to make the case that no containment would be needed, because the pellets themselves would provide the primary containment, and therefore no credible scenario exists for release of radioactivity from the fuel. If this argument were accepted, the MHTGR would gain two principal advantages: cost and siting. A low-pressure confinement structure can be significantly less expensive than a full containment. And if the NRC agreed that there were no credible radiation release scenarios, the vendor argues that the NRC should drop the requirement for emergency planning and should allow the reactor to be sited close to industrial areas. Siting in industrial areas is a particular advantage for the MHTGR, which is a high-temperature reactor whose 700° C exit temperature (National Research Council 1992, p. 119) could be used to provide process heat.

Plutonium-kernel fuel will require extensive testing for licensing approval if the pellet integrity is to be relied on for a major element of the safety analysis. (As noted above, however, the plutonium-burner design is provided with a high-pressure, low-leakage containment, unlike the commercial design.) In addition, the direct cycle concept is new, leading to additional licensing issues. General Atomics notes (GA 1994, p. 13-1): "because the MHR is significantly different in technology and design philosophy from that with which the regulatory community is familiar, extra effort will be required . . . to familiarize the reviewers with the technology."

Safeguards, Security, and Recoverability

The MHR has a few unique features that affect the safeguards issue. The fuel form itself provides a barrier against ready access to weapon-usable plutonium. Once the plutonium has been placed in the kernel of the MHR fuel pellets, the silicon carbide coating makes recovery of the plutonium more difficult

⁴³ USDOE (1993) reports that the licensing review had been terminated (p. 64). The review has not been terminated, however, only put on hold until the Clinton administration and Congress reach a decision on what efforts will be continued for the HTGR (Miraglia 1994.)

than from normal fuel. Thus, manufacturing MHR fuel as soon as possible, well before it could actually be used in a reactor, would provide a more significant safeguard than would fuel fabrication for most other reactors.⁴⁴ Overall safeguards risks during the fuel fabrication process would probably be comparable to those for other reactor types. Security risks in transportation of the fabricated fuel would be lower because of the greater difficulty of retrieving the WPU from the fresh MHTGR fuel.

Because of the higher burnup possible in an MHR, and the unique fuel characteristics, MHR spent fuel would pose a less attractive target for a potential proliferator than most other types of spent fuel. The argument should not be overstated, however. Once the WPU has been converted to spent fuel—and thus becomes only one small part of the much larger global stock of plutonium in civilian spent fuel—whether that small part of the total is significantly more proliferation-resistant or not, does not have a major bearing on overall safeguards and security concerns. That is the rationale for the “spent fuel standard.”

In particular, while an MHR could potentially destroy a large fraction of the plutonium and leave the residual with high proportions of the isotopes above Pu-239, this does not greatly change the security picture when this spent fuel’s place in the global stock is considered: whether the amount added to a global stock of 1,000–2,000 tons of plutonium in spent fuel is 50 tons or only 5 tons does not make a dramatic difference. And while the plutonium from high burnup MHR fuel would be more troublesome to use in explosives than standard reactor-grade plutonium, the fact is that nuclear explosives can be constructed from virtually any combination of plutonium isotopes (other than nearly pure Pu-238; see Chapter 2 and NAS 1994).

The volume and mass of the MHR fuel that would have to be stolen to acquire enough plutonium for a weapon is about 10 times larger than the equivalent volume for LWR fuel. On the other hand, because of the absorptive properties and small size of the graphite blocks, the radiation dose rate from MHR spent fuel would be substantially lower than the dose rate from standard burnup LWR fuel of equivalent age,⁴⁵ making the MHR fuel somewhat easier to handle.

Separating the remaining plutonium from the MHR spent fuel would be more difficult than in the LWR case. The several layers of coating around each tiny particle would pose some barrier to recovery of weapons material. Methods to reprocess MHR fuel have been proposed, but there is no worldwide experi-

⁴⁴ Note, however, that in the case of the ALMR, proponents have suggested a fuel form in which the WPU would be mixed with some minor actinides and fission products even before irradiation in the reactor, offering a significant safeguard against recovery of the plutonium. This concept is addressed in Chapter 5.

⁴⁵ Ten years after discharge, the gamma dose one meter from the surface of a 100-kg MHTGR WPU fuel block that had been irradiated to 580,000 MWd/MTHM would be about 180 rem/hr, compared to about 940 rem/hr 1 meter from the surface of a 660-kg PWR WPU-MOX fuel assembly that had been irradiated to 40,000 MWd/MTHM (see Table 6-5).

ence with actual implementation of these methods comparable to the experience with reprocessing of LWR or LMR fuels. (This lack of experience is desirable for the proliferation resistance of once-through fuel, but undesirable if an elimination approach involving reprocessing of this fuel is to be pursued, as described above.) Since the kernels begin as pure plutonium oxide, however, there would be more plutonium per kilogram of initial heavy metal—once the kernel material was separated from the particles in the HTGR waste—than there would be in the spent fuel from other reactors.

Cost

The MHTGR is one of a set of advanced reactors designed to overcome the current utility- and public-acceptance problems seen as preventing expansion of nuclear power in the United States. Estimates of comparison costs indicate, however, that the MHTGR capital costs would be approximately 30 percent higher than those for ALWRs (National Research Council 1992, p. 139). Cost estimates for any unbuilt reactor design are uncertain. Cost estimates for development and fabrication of the plutonium fuel are even more uncertain. Nevertheless, numerous studies have concluded that the electricity from an MHTGR would be more expensive than that from ALWRs.

DOE concluded in Phase I of its Plutonium Disposition Study that, of the reactor options examined, only the HTGR did not produce a positive net value over its life cycle, using DOE's economic assumptions (USDOE 1993, Vol. I, p. 5). General Atomics' calculations are more favorable to the MHTGR in some cases and less so in others; once discounting is applied, the General Atomics calculations also would not produce a positive net present value for the plutonium disposition campaign.

It is to address this cost differential that General Atomics has proposed the GT-MHR. The direct cycle is estimated by the vendor to lead to a 25 percent increase in thermal efficiency, as well as reducing the capital cost of the reactor. The vendor also proposes to increase the reactor to 600 MWt, also reducing the cost per megawatt. Further development and analysis will be necessary to validate the approach and cost improvements projected by the vendor.

There is no current infrastructure remaining for MHTGRs. General Atomics has maintained a cadre of technical staff who could begin design and testing immediately, however.

Environment, Safety, and Health

The MHTGR offers a potential advantage of improved safety against accidents compared to LWRs. If "containment-in-a-pellet" passes licensing approval, the reactor would be described as having an inherent safety substantially greater than LWRs. Such proof would require demonstrating, however, that the

number of fuel pellets that could deteriorate and provide a channel for radioactive product release would be insignificant.

The used fuel is taken out of the reactor with the fuel remaining in the graphite elements, thereby producing a large volume of high-level waste. Although it would be possible, in principle, to extract the fuel compacts, Oak Ridge National Laboratory (ORNL) has examined potential waste-disposal options for the MHTGR and has concluded that the preferred option would be disposal of the spent fuel still embedded in the graphite blocks. ORNL believes the fuel blocks could be placed in spent fuel waste containers similar to those for LWR spent fuel. General Atomics states that HTGR spent fuel could be packed more closely than LWR spent fuel because the HTGR waste has lower heat generation per unit volume, and that therefore the necessary repository size should not be increased significantly (see GA 1994, p. 13-2, referencing Lotts et al. 1992; GA 1993b, p. 2-335). DOE stresses, however, that the HTGR waste volumes are expected to be much greater than for the other plutonium disposition concepts. Current repository regulations bar emplacement of potentially flammable materials, including graphite. General Atomics argues that flammability is not a problem for nuclear-grade graphite (GA 1994, pp. 3-28 to 3-29).

Other Issues

Construction of a new MHR for burning WPu would be comparable to construction of a new LWR for the same purpose in terms of the signal relating to fuel-cycle policy that some parties might perceive such an action to send (see discussion in Chapter 6, "General Considerations").

MOLTEN-SALT REACTOR

Description of Technology and Status

The molten-salt reactor (MSR) would use a liquid fuel to carry the plutonium or other fissionable material into the reactor and to carry both heat and fission products out. The proposed approach has a spherical reactor vessel. The liquid fuel is a mixture of lithium, beryllium, and plutonium fluorides (possibly including some zirconium fluorides as well), which only attains criticality in the graphite moderator of the reactor core. Criticality excursions are limited by the large negative temperature coefficient of the fuel, which means that as criticality increases and more heat is produced, the reaction immediately slows down. The fuel circulates to an external heat exchanger. The heat-exchanger loop, which also uses a salt, goes to a steam generator for power production. A side stream is drawn from the reactor cooling loop, processed to remove fission products, and then returned to the reactor. In addition, the fuel salt would be in contact with helium for removal of volatile fission products (Gat undated). By this method of

continuously recycling the fuel, an MSR could, in principle, achieve near-total elimination of plutonium added to the system.

As an advanced reactor requiring a long period of development and demonstration before a commercial-scale system could be built, the MSR is not competitive for the spiking or spent fuel options, so these are not discussed further in this section. It could be a candidate, however, for the elimination mission.

Liquid-fuel reactors have two main advantages: ease of heat removal and fuel management. They also have the potential for achieving high power density. The MSR began as part of a program to develop a nuclear-powered aircraft. The reactor development effort was shifted to a commercial power design in 1956, and to a breeder program in 1960 (McPherson 1985). In the early 1950s, the liquid-fuel reactor design envisioned uranium fuel dissolved in a molten salt. The early view of the utility of molten salts was mixed (Glasstone 1955, p. 527; Lane et al. 1958). Supporters of molten-salt systems (located primarily at the Oak Ridge National Laboratory) believed, however, that the Atomic Energy Commission decision to concentrate on the LWR and the sodium-cooled reactor was a mistake, and they have maintained interest in the MSR ever since, although basic funding for the program ended in the mid-1970s. Because development of this concept was abandoned two decades ago, the available base of expertise in molten-salt concepts is minimal. Recent interest in molten-salt systems has focused primarily on subcritical reactors driven by accelerators, an approach known as accelerator-based conversion (ABC, described below), though some study is now being devoted to molten-salt systems without accelerators.

Current MSR concepts are based on the Molten-Salt Reactor Experiment (MSRE), a low-power reactor (8 MWt), which operated with various salt compositions that included both uranium and plutonium.⁴⁶ The MSRE was initially operated with U-235 at 35-percent enrichment and operated for nearly three years, from 1965–1968. A U-233 fuel was added after removal of the U-235. The plutonium produced remained in the salt and several additions of PuF₃ were made, though uranium remained the dominant fuel. Later work concentrated on developing an MSR thorium breeder, using a ⁷LiF-BeF₂-ThF₄-UF₄ salt.

Several problems were identified during the operation of the MSRE and in subsequent analysis before the program was terminated. These included developing graphite with higher radiation resistance and lower permeability to the salt and to xenon produced during fission, so that the graphite would not have to be replaced every few years. A second problem was that the creep-ductility of the material that had been used for the vessel and piping, Hastelloy N (a nickel-base

⁴⁶ For a discussion of the molten-salt system in general, see the series of papers edited by Weinberg (1970) and McNeese and Rosenthal (1974). For discussions of the MSR specifically for disposition of WPu, see Gat et al. (1992, p. 391) and Omberg and Walter (1993, pp. 17-18).

alloy), was reduced by neutron radiation. A solution under investigation at program termination was to add 2 percent titanium. An environmental problem is the production of tritium, produced in lithium in the fuel salt; tritium diffuses readily through metals at the MSRE operating temperatures (McNeese and Rosenthal 1974, pp. 55, 57-58). Use of a mixture of sodium, fluoride, and sodium fluoroborate to capture the tritium had been proposed as a solution (McPherson 1985, p. 377). Further discussion of molten-salt issues can be found below in the section "Accelerator-Based Conversion of Plutonium."

Compared to other systems discussed in this chapter, little information is available on the MSR. The MSR sponsor recently wrote: "there is no active program on molten salt reactors and, hence, we cannot make any additional calculations, no matter how simple they are" (Gat 1993, p. 1). The sponsor noted that "there is no current cost estimate for fuel processing" and "the estimated duration of [full] development has such wide margins that any number or even range is meaningless."

Reprocessing in the MSR concept is part of the integral fuel management. A fuel loop would carry a portion of the fuel continuously through a reprocessing station, where fission products would be removed and the actinides and salt would be returned to the reactor. The fissionable material would be recycled until consumed. The basic processing steps were demonstrated by the MSRE, with the important exception of demonstrating how to separate plutonium from the salt in the presence of uranium and fission products, which would require development effort (Omberg and Walter 1993, p. 18).

MSR advocates have always stressed potential safety features of a molten-salt reactor, including possible passive safety. Proponents argue that the MSR design has a low source term due to its small radioactive inventory and a lack of energy to drive an offsite release, and that its simplicity and passive safety features add to its overall safety (Gat 1986). Some safety issues remain unresolved, however, including possible reactivity excursions, containment of radioactive tritium (although the intermediate-loop proposal may solve that problem), and the containment of the hot, radioactive liquid fuel in the case of a pipe break.

Timing and Other Issues

The MSR exists only as a conceptual design. Considerable development effort would be needed before any MSR system could be fielded. A study by the Lawrence Livermore Laboratory estimated that 23–27 years would be required for deployment after decision (Omberg and Walter 1993, p. 28). No calculations have been done concerning how long it would take to eliminate specific fractions of the input plutonium (e.g., 99 or 99.9 percent) in an MSR system.

The sponsor of the MSR has noted the licensing problems associated with such systems, arguing that "perhaps the biggest availability issue is the licensing of MSRs. Present licensing practices are oriented solely toward LWRs and solid

fuel systems. . . . The licensing of MSR, keeping their advantageous property intact[,] by NRC will require a large effort” (Gat undated, p. 8). Given the novelty of the reactor concept, public acceptance is an open question, and it is highly unlikely that any utility would be interested in trying this reactor.

The economics of an MSR system are too uncertain to estimate with any confidence. The long development effort required to bring the MSR to the level of technical maturity of the ALWR, ALMR, or MHTGR would be costly, however; and it appears unlikely that an advanced fluid fuel reactor system with reprocessing can be competitive economically with LWRs in producing electrical power while uranium prices remain low.

Estimating the ES&H impacts of an MSR system is nearly impossible at this early stage. The MSR may offer a reduced risk of large accidents, but also may have a larger number of scenarios that could lead to radioactive contamination within the reactor building, leading to greater worker hazard. The intensely radioactive molten fuel would lead to high radioactivity levels throughout the reactor primary circuit, requiring development of remote-handling equipment and potentially posing risks of significant worker exposures.

Safeguards and security risks for an MSR system should be low, once the plutonium is mixed with the radioactive molten fuel. If the plutonium is recycled until it is nearly all consumed, the risk of recovery after disposition would be effectively eliminated.

PARTICLE-BED REACTORS

Description of Technology and Status

The particle-bed reactor (PBR) concept evolved from designs for nuclear-powered rockets for the Strategic Defense Initiative program, a concept code-named Timberwind. The PBR fuel elements consist of two concentric cylinders with fuel in tiny particles packed between them. The fuel particles (diameter 0.8 mm) would be similar to those of the MHTGR. The PBR fuel particles have a central graphite kernel that, in the version under development, contains uranium carbide. The plutonium burner would have a kernel of a graphite matrix with PuC₂, a graphite layer, and an outer coating of pyrolytic graphite. If fuel tests indicate it is necessary, a final layer of silicon carbide would be added (Brookhaven 1992, pp. 2, 40; Ludewig 1993b).

The inner and outer cylinders are constructed of porous tubes called frits. Gas coolant flows through the center core and outwards through the cool inner frit, across the fuel pellets, and then through the hot outer frit. Fuel elements are arranged in hexagonal patterns and surrounded by a moderator, vital to the operation of the system, which could be graphite, beryllium carbide, heavy water, or other materials. The proposed design uses helium coolant, with the cold frit at approximately 300° K and the hot frit at about 1,000° K. The hot frit would be

made of Incalloy and the cold frit made of Zircalloy. Very high power densities are estimated for this design—approximately 5 megawatts per liter.⁴⁷ Therefore the reliability of coolant flow must be high.

The basic PBR design, developed for nuclear rockets, has been discussed for many years. The proponent of this concept is Brookhaven National Laboratory (BNL), whose work has been done for the Air Force. The space nuclear propulsion program has been terminated, however, and the BNL work on it ended at the end of fiscal year 1993. A low level of effort continued, aimed at developing proposals to use the PBR for plutonium and actinide burning (Ludewig 1993c).

The BNL program has developed the uranium fuel particles, the frits, and control designs, and has analyzed the thermal hydraulics and neutronics. The design also has been proposed as a means of transmuting actinides and long-lived fission products from LWR spent fuel. In that case, target elements containing fission products and minor actinides would be introduced along with fuel elements.

The reactor would be shut down during loading and unloading operations, estimated to be required at least every two to four weeks.⁴⁸ The proposed design would have a hydraulic loading and unloading operation, a concept that has not been tested on PBR nuclear fuel elements.

Since the design was being developed for a rocket, no significant effort had been devoted to considering how the design could be used as a power reactor. BNL estimates that a conventional steam cycle based on MHTGR experience would be used. The use of helium as a coolant has been tested at low pressure and with electrically heated particle beds. The design uses beryllium carbide, Be_2C , in a graphite structure as the moderator, which would require developing manufacturing techniques and raises some potential ES&H issues. The basic fuel particle, PuC_2 , has not been fabricated or tested (Koopman et al. 1992, p. 4-36).

Potential problems that have arisen in testing include the following (Brookhaven 1992, pp. H2-H5; INEL 1993a, pp. 10-11):

- Chemical compatibility of the fuel, coatings, fission products, and coolant.
- Liquefaction of the hot frit, either because temperatures were hot enough to melt the hot frit or because of thermal interaction between the hot frit and the fuel.

⁴⁷ Brookhaven (1992, pp. 1, 4). The power density for the nuclear rocket version was several times higher.

⁴⁸ Brookhaven (1992, p. 2). BNL states, however, that it may be necessary to load and unload more often than weekly to minimize reactivity swings.

- Significant fuel swelling due to retention of gaseous fission products, which has reached 20–30 percent and could lead to extensive fuel failure.
- Potential complications in fuel fabrication from impurities in the plutonium. Extrapolation from the uranium fuel fabrication technology to plutonium is not warranted.
- Shortening of the hot frit with cycling, which can lead to fracture and loss of fuel pellets. Nuclear testing has resulted in hot frit shortening and numerous cracks in one element, which may limit the number of times the PBR fuel elements can be thermally cycled.
- Thermal and hydraulic flow stability in the particle fuel bed.

Timing and Other Issues

As with the MSR, the recent LLNL study concludes that deployment of a commercial-scale PBR would require at least 20–25 years after a decision was taken (Omberg and Walter 1993, p. xiii). Hence the PBR also would not be competitive with other approaches for the spiking or spent fuel missions.

Because of the extremely high power density in the PBR, consumption of actinides in the system would be rapid. BNL estimates that the PBR could reduce a small initial plutonium inventory by 95 percent in 20 days, based on a burnup of 500,000 MWd/MTHM (Ludewig 1993a, p. 2). Reprocessing would then be required if a true elimination option was to be pursued. BNL states that techniques exist to reprocess the spent fuel particles for this purpose. As in the case of the HTGR, however, reprocessing approaches for this fuel require considerable development, as would fabrication technologies for multiple recycle materials.⁴⁹

As with the MSR, licensing and public acceptance of an advanced and unfamiliar reactor type would be time-consuming and uncertain. The NRC has not licensed a similar type reactor. Many questions would need to be answered before a licensing estimate could be made. Serious problems are to be expected.

ES&H impacts of a PBR system are also uncertain. There would be a low volume of radioactive waste in the fuel, but the reactor components may need to be discarded on a much more frequent cycle than that for current reactors, leading to a higher total volume of radioactive waste. Current repository rules bar emplacement of potentially flammable materials, including graphite; as with the MHTGR, this problem would have to be addressed. Frequent refueling would increase the risk of worker exposures. The design has not been developed far enough to provide an estimate of safety. Because of the small core, the total amount of fission products contained at any one time is small compared to a

⁴⁹ For further discussion of the PBR as an actinide burner, see National Research Council (forthcoming).

normal LWR. In that sense, the hazard may be lower. Nevertheless, given the high power density, careful design will be required to provide adequate emergency cooling to prevent excessive heating of core materials in the event of a loss-of-coolant accident. Moreover, because of the high burnups projected on a single pass, reactivity swings in the reactor will be substantial. Uncertainty masks any safety estimate.

Similarly, it is too early to guess what the cost of a PBR system would be. Given the early stage of development, and the uniqueness of the concept, developing and deploying a PBR for WPu disposition would probably be significantly more expensive than developing and deploying any of the concepts already well into the development cycle.

A DEDICATED PLUTONIUM-BURNER REACTOR

To ensure that the full range of possibilities was appropriately explored, the panel considered what a dedicated plutonium disposition reactor designed for no other purpose would look like. The mission of such a dedicated disposition reactor would be to transform plutonium into spent fuel as quickly, safely, and cheaply as possible—without the accompanying mission of producing electricity in the process.

Removing the mission of producing electricity would simplify the design of a reactor substantially. There would be no need for high temperature to operate a heat engine, or for the high pressure that accompanies high temperature in the case of water coolants. Capital costs would be much reduced by eliminating turbines, generators, many of the support facilities, and the thick-walled pressure vessel needed to contain a high-pressure system. Reactor design would be simpler and more flexible. Operation at low temperature and pressure would minimize the required strength of reactor vessels, piping, and the like. In fact, the reactor could operate in a warm pool of water and rely on natural circulation for removal of decay heat. The fuel temperature would also be low, allowing one to use fuel that could be more cheaply fabricated (an important part of the cost of operations in a plutonium-fueled reactor). Safety could be significantly increased, in part because the low operating temperature would result in little or no steam energy with which to deal in the case of an accident.

At the panel's request, the Idaho National Engineering Laboratory (INEL) outlined a conceptual design of such a reactor (INEL 1993b). They envisioned a reactor with a core of comparable size to those of modern LWRs, but with a power density 10 times lower, and a maximum fuel temperature in the range of 400° K, rather than over 2,000° K in the case of a PWR. The core envisioned would have a thermal power of 1,000 megawatts.

A fuel with particularly low-cost fabrication would be plutonium-aluminum alloy extruded in aluminum cladding. Since the goal would be to secure plutonium by getting it into the reactor as quickly and cheaply as possible, fuel with

very high plutonium loadings would be highly desirable. This could be achieved with the use of large quantities of burble absorbers, such as erbium, to offset the extra reactivity. It would be quite feasible to design such a core to be loaded with 10 tons of plutonium at a time.⁵⁰ This would mean, however, that when discharged, the fuel would be substantially richer in the remaining plutonium than ordinary LWR fuel (by a factor of 10). This would have to be balanced against the reduced fabrication cost. Detailed study of whether the cheaply produced aluminum-clad fuel would be an acceptable waste form for geologic disposal would also be required. Such fuels have historically suffered corrosion problems in storage, leading to their being reprocessed rather than stored in most cases, and certifying such a fuel type for repository disposal would be difficult, time-consuming, and costly. Stainless steel cladding is an alternative proposal that would also have relatively low cost.

All told, the capital cost of such a reactor would be expected to be hundreds of millions of dollars less—perhaps more than a billion less—than that of a typical LWR, and operations costs would also be expected to be less.⁵¹ But since this reduction in cost would come with the sacrifice of many billions of dollars in revenue that an electricity-producing reactor would provide, it appears extremely unlikely that the net discounted present cost of developing, licensing, building, and operating such a system for plutonium disposition would be competitive with the costs of using MOX in existing or new LWRs. Design, development, and licensing of the reactor and its fuel in the current regulatory environment in the United States would be expected to take many years, moreover. Thus the panel does not recommend further pursuit of the option of a specially designed plutonium burner without electricity production.

ACCELERATOR-BASED CONVERSION OF PLUTONIUM

Description of Technology and Status

Accelerator-based conversion (ABC) systems have been under study as a means of eliminating plutonium and of fissioning actinides and transmuting fission products in order to reduce the longevity of radioactive wastes. Despite the name, these concepts do not involve bombarding the plutonium directly with an accelerator. Rather, these concepts are a variation on reactor burning of plu-

⁵⁰ The INEL core physics analysis was done for a loading of 1 atomic percent plutonium in aluminum, corresponding to about 48 grams per fuel pin, or nearly 10 tons of plutonium per core. This is low enough for fabrication by extrusion.

⁵¹ By one very rough estimate (Sauerbrun 1993), such a plutonium burner might have a discounted present cost over a 40-year life of \$5.5 billion compared to \$6.6 billion for a demonstration ALWR. But the additional cost of the ALWR is more than offset by many billions of dollars in electrical-generation revenues.

tonium: a subcritical reactor—meaning one that is not capable of sustaining a chain reaction without outside neutron input—would be driven by neutrons produced by a beam of particles from an accelerator hitting a target. Typically the proposed reactors would have a multiplication factor, k , of about 0.95, meaning that 19 out of every 20 neutrons in the system would come from fission, not from the accelerator. A portion of the power provided by the reactor (typically 10–20 percent) would be used to power the accelerator.

The accelerator would act, in effect, as a negative control rod: shutting off the accelerator would, if everything worked as intended, bring the reactor below criticality and shut down the reaction. This could be done more rapidly than mechanical control rods can be inserted in an ordinary reactor, which advocates argue would lead to improved safety—particularly in the case of plutonium fuels, with their smaller delayed-neutron fraction. Proposed ABC systems, however, raise some reactor safety issues of their own, which have not been fully resolved.

Analysis of ABC systems is hampered by the existence of a wide array of fast-changing proposals. Virtually any type of reactor could be arranged in a subcritical state, to be driven to criticality by spallation neutrons from an accelerator hitting a target.⁵² Liquid-fueled reactors with both aqueous-slurry and molten-salt fuels have been examined, as has a gas-cooled particle-bed system. LWRs, LMRs, or HTGRs could also be considered. During 1994, the approach being most energetically examined was a molten-salt system; this system is the focus of discussion in this section. Just as the reactor subsystem is uncertain, the power level of the accelerator has not yet been optimized, nor has the spallation target that would produce the neutrons when bombarded by the accelerator been fully developed.

ABC concepts are only at the early paper-study stage, and such devices could not be operational on a large scale for decades. Both the proposed fluid-fuel subcritical-reactor technology and its fuel-cycle technology are unproven and extremely challenging. As such, these advanced concepts are a potential long-term option for the elimination mission, but are not competitive with nearer-term reactor types for the spiking or spent fuel missions.

Nevertheless, ABC research efforts at Los Alamos have shifted, at DOE direction, from an initial focus on developing an option for nearly complete elimination of both WPu and the much larger global stocks of RPu, to a current focus on a once-through approach for burning of WPu, without explicit consideration of recycling. Under this approach, a significant fraction of the plutonium would

⁵² Indeed, such concepts have been discussed for several decades, and there is currently some interest in them in many of the major nuclear-power-producing nations, including the United States, Russia, Japan, and several countries in Europe. One of the more recent and more vigorous proponents of the accelerator-driven subcritical reactor is Carlo Rubbia of CERN (the European Laboratory for Particle Physics in Geneva, Switzerland), who with his colleagues is proposing thorium burners using rather conventional sheafed oxide pins.

remain in the spent fuel. A recent review has concluded that “this narrowing of mission” from the total world plutonium stock to burning a fraction of the WPu stock is “unwise,” and that “the ABC concept as refocused under guidance is not worthy of support” (JASON 1994, pp. 17, 18). In this section, therefore, we examine an ABC refocused on the broader elimination mission, which would require reprocessing and recycle. As argued elsewhere in this report, incurring substantial additional delays, costs, or risks to pursue the elimination option for WPu should only be considered if the much larger global stocks of plutonium in the civilian cycle are also to be consumed.⁵³

In a number of the proposed ABC approaches involving reprocessing, both actinides and long-lived fission products would be separated and recycled. Excess neutrons, either from the subcritical reactor itself or from a separate accelerator system, could be used to transmute the long-lived fission products (particularly technetium-99 and iodine-129) into short-lived species.

ABC advocates claim that such systems will ultimately be able to transmute all the long-lived species produced in the reactor, providing large-scale nuclear electricity generation with virtually no long-lived radioactive wastes, while simultaneously transmuted the wastes from previous generations of nuclear power. In principle, resource extension could also be achieved, through use of a thorium-uranium breeder cycle. This concept of vast resources of virtually waste-free power, however, remains little more than a vision at present. A wide array of key technologies involved remain to be demonstrated.

In particular, the panel notes that if the cost of power produced by such a system were even a few mills per kilowatt-hour higher than that of electricity produced by other means—which appears very likely, given that the system involves advanced reactors with extensive and complex reprocessing and with a significant fraction of the electricity produced being siphoned off to run the accelerator—then the total subsidy needed for transmutation of all the actinides and long-lived fission products in the spent fuel already produced in the world would amount to many hundreds of billions of dollars.

A Baseline System

As currently envisioned,⁵⁴ a baseline ABC system would consist of the technologies discussed in the following paragraphs.

⁵³ For a comparison of ABC to other systems for the broader role of transmutation of actinides and other long-lived radioactive wastes, see National Research Council (forthcoming).

⁵⁴ Except where otherwise noted, descriptions and numbers used in this section are for the variant of ABC reviewed in JASON (1994).

Accelerator

The accelerator power required depends on how far the reactor is from criticality (that is, the value of k), and this parameter has not yet been optimized. Similarly, the current and proton energy of the accelerator could cover a broad range. As of January 1994, the Los Alamos team advocating the ABC approach envisioned an accelerator with a beam energy of 800 million electron volts and a current of 75–100 milliamperes.

Reactors

The accelerator beam would be split between four to eight target modules, each consisting of a beam target and a subcritical reactor. A variety of material choices are possible for the target of the proton beam; the target would be designed to produce of the order of 25 neutrons per proton.

These neutrons would go into a subcritical reactor system, which would consist of molten salts (lithium, beryllium, and zirconium fluorides), with plutonium dissolved in them, circulating within graphite blocks that would serve as the moderator. Each module might have a power of roughly 500 MWt, so that an eight-module system might have a net output power of 1 GWe (after subtracting 10–20 percent of the electrical power to run the accelerator). The neutron flux in these modules would be extremely high—initially $3\text{--}8 \times 10^{14}$ n/cm² (neutrons per square centimeter). The reactor would be cooled by circulating the fuel through heat exchangers immersed in the molten salt. As envisioned in January 1994, each module designed for the WPu destruction mission would begin with roughly 20 kg of plutonium in the fuel. The fuel would continue to reside in the reactor throughout the 10-year expected module life; additional plutonium would be added over time to make up for the loss of reactivity resulting from plutonium consumption and the buildup of fission product poisons. All told, 2.3 tons of plutonium would be added to each module over its 10-year life, of which approximately 90 percent would be fissioned during that period.

Reprocessing and Enrichment

As noted above, in some current concepts there would be no reprocessing. The actinides and fission products that remained in the fuel at the end of the 10-year module life would simply be vitrified and disposed of in a geologic repository. This would be radically different from earlier ABC and accelerator transmutation of waste (ATW) approaches, which advocates claimed could minimize or eliminate long-lived wastes by reprocessing and recycling the actinides and long-lived fission products. In the no-reprocessing approach, there would still be significant quantities of plutonium in the waste. Thus this approach is another, and expensive, way to burn a fraction of the plutonium while

leaving the rest embedded in radioactive waste. It cannot be considered a genuine "elimination" option.

The fraction of the plutonium consumed would be higher than in the case of LWRs (and certainly higher than in the vitrification approach, where none of the plutonium is consumed), but this difference would offer little security advantage, since the plutonium in the waste, whether it was 5 tons or 50 tons, would be only a small part of the many hundreds of tons of plutonium in spent fuel worldwide. Given the long time, high cost, and high uncertainty that would be involved in ABC development, the panel does not believe that ABC would be competitive with other systems unless it were designed for the elimination mission, which would require reprocessing and recycle.

Reprocessing has been examined for these concepts and demonstrated on a limited scale in the MSRE. If both actinides and long-lived fission products are to be transmuted, these would both have to be separated from the short-lived fission products, which would be stored onsite until they decayed. Originally, advocates of fluid fuel ABC concepts proposed an extremely challenging online reprocessing approach in which small amounts of fluid fuel would be continuously removed from the reactor, reprocessed with very little delay (meaning that the fuel would be orders of magnitude more radioactive than solid fuels in past reprocessing experience), and the actinides, long-lived fission products, and molten salt would be returned to the reactor. The average time each atom spent in the reactor between reprocessing cycles would have been only a few weeks, and on average, each atom of fissile material would have gone through more than 100 reprocessing cycles before being fissioned. Recovery of the important radioactive species requiring further transmutation would have to be extremely high to avoid these species building up in the waste during these repeated reprocessing cycles. In addition to chemical separations, enrichment of cesium isotopes would be required, because of the need to transmute long-lived cesium-135 (Cs-135) without burdening the reactor with comparatively short-lived Cs-137. The challenging separations needed for this concept have not been demonstrated and are likely to pose larger technical difficulties than would the other reprocessing concepts discussed in earlier sections of this report.

In the more recent concept, reprocessing and enrichment would be required only after the fuel's 10-year residence in the reactor module. This would allow greater cooling time for the fuel before reprocessing. Still, the required molten-salt processing would pose substantial challenges, and the problem of enriching intensely radioactive cesium isotopes—which has never before been done on the required scale—would remain.

Because the plutonium would not be separated from the minor actinides and long-lived fission products in this reprocessing approach, the risk of plutonium theft or covert diversion would be substantially lower than in reprocessing concepts in which the plutonium is fully separated—an advantage that ABC shares,

to varying degrees, with other reactors using similar reprocessing approaches, such as the IFR.⁵⁵

Balance of Plant

As with other reactors, the heat from the ABC would be used to drive turbines to produce electricity. As noted, 10–20 percent of this electricity would be used to power the accelerator.

Alternatively, the ABC could be supplied with external power for the accelerator and use a cool slurry rather than one which is hot enough to provide (through a heat exchanger) steam to drive a turbogenerator. In this way, the accelerator and subcritical reactor could be divorced from the cost and constraints involved in providing electrical power. Using ABC systems to transmute a large fraction of all the actinides in the world, however, would not be economically practical unless electricity were produced to cover at least a portion of the cost.

Challenges

The accelerator part of the ABC system is close to the existing state-of-the-art, but every aspect of the reactor and reprocessing systems involves important unknowns.

The Accelerator

The average power of the accelerator would be well beyond that of any accelerator previously built, but the peak power, particle energy, and current required have all been demonstrated. While the accelerator would require considerable development, there is no reason to believe that an accelerator with the required capabilities could not be built on a reasonable time scale. Similarly, technology appears to be available to build an accelerator target that would produce the required spallation neutrons.

The Reactors

The proposed fluid fuel subcritical reactors are based on immature technology that still faces major challenges. The reactor technology is based on the much smaller (8-MWt compared to 500-MWt) Molten-Salt Reactor Experiment (MSRE) which operated from 1965–1968 (see description in section on “The Molten-Salt Reactor” above). As noted earlier, the MSRE program was terminated in the 1970s, with other reactors chosen for development, meaning that

⁵⁵ As in the case of the IFR, however, a country which had been provided the separations technology needed for this approach would be able to adapt that technology to separate plutonium for nuclear explosives, if desired.

only a modest current knowledge base is available to answer questions regarding molten-salt systems.

The MSRE identified several major issues that were not resolved when the program terminated. In addition, several aspects of molten-salt ABC systems would differ substantially from the MSRE, posing new challenges.

Loss-of-Coolant Accidents. One of the most significant safety issues in power reactors is the need to deal with fission-product heating after shutdown. The proposed ABC system relies on heat exchangers for cooling. Systems will be needed to ensure adequate cooling in the event of heat-exchanger failure. Design for such an accident would require means for dumping the contents of the reactor into a volume that maintains the fuel in a subcritical geometry, prevents the intensely radioactive fission products from escaping or otherwise posing a safety hazard, and provides reliable, effective means for removing hundreds of thermal megawatts of fission product decay heat. Earlier aqueous-slurry ABC concepts would not have met the basic safety criterion of being able to survive a pipe-break accident, as such an accident would have led to intensely radioactive slurry fuel spraying out into the containment.

Reactivity Excursions. The choice of $k = 0.95$ reflects an implicit assumption that there will be no reactivity excursions of more than 5 percent. This assumption may or may not be correct and must be thoroughly demonstrated if the reactor is to be judged safe. Fluid fuel reactors, for example, tend to have very large negative temperature coefficients of reactivity (because of the expansion of the fluid fuel as it heats up). Therefore going to cold shutdown would greatly increase the reactivity of the reactor—possibly by an amount larger than the capacity of the accelerator to control. Control rods similar to those used in more conventional reactors may be needed for safe shutdown.

Another problem is potential xenon oscillations. High neutron-flux thermal reactors tend to experience spatial oscillations in power density, caused by xenon-135 produced during the reactions, which can cause local overheating. Xenon oscillations are a significant safety issue in LWRs and even more so in heavy-water moderated plutonium production reactors; they could be more severe in an ABC system, with its higher neutron flux. (Such fluctuations were not a major problem for the MSRE, because of its small size.) Moreover, they could be exacerbated by xenon encroachment in the graphite moderator, which was observed in the MSRE. (Some approaches to sealing the graphite appeared to limit xenon encroachment substantially in the MSRE experience, but graphite permeability increased under neutron bombardment; this problem was not solved before the program was terminated.)

A further problem arises from the very large neutron absorption cross section of samarium-149 (Sm-149), a daughter of the fission product promethium-149, which has a 53-hour half-life. Shutting down a high-flux reactor can

lead to the growth of such large amounts of Sm-149 that it may not be restartable until the Sm-149 has been removed.

If the ABC were partly fueled with thorium, as envisioned in some concepts, another potential problem is created by the isotope protactinium-233, which decays to U-233; when power is reduced from equilibrium, the amount of U-233 in the system will increase, causing an increase in reactivity. The higher the neutron flux, the larger this reactivity swing would be, raising a concern for the high-flux ABC.

In short, considerable work needs to be done before there can be high assurance that the risk of potentially dangerous reactivity excursions in the proposed subcritical reactors can be eliminated.

Use of Plutonium. One of the important differences between the proposed ABC molten-salt system and the MSRE is the use of plutonium as the fuel. Only a small amount of plutonium was ever introduced into the MSRE, which ran primarily on uranium. A technology development program would be needed to demonstrate that the different neutronic and chemical characteristics of the system with plutonium are adequately understood.

For example, fission of uranium tends to make the chemistry of the molten salts more oxidizing, while fission of plutonium tends to make it more reducing; a reducing environment could lead to precipitation of certain metals, which could have serious effects (such as blocking flow and thereby leading to local overheating).

Miscellaneous Issues. Other issues identified in the MSRE include surface cracks and ductility changes in the alloy used to contain the salt, graphite permeability to the salt, and shape stability of the graphite under intense irradiation. Warping of the graphite blocks in an ABC system could create voids where fuel might collect and stagnate, causing local overheating. The production of tritium from bombardment of the lithium in the salts is a potential ES&H concern, particularly given the permeability of the containment metals to tritium at the relevant temperatures. Similarly, the reliance on highly toxic beryllium fluoride in the MSRE might not be acceptable under today's ES&H standards.

The Reprocessing

The reprocessing involved in a molten-salt system, particularly one designed to transmute long-lived fission products as well as actinides, would be quite different from the aqueous reprocessing used around the world today. There is no base of experience available comparable to the experience with aqueous reprocessing. As noted above, enrichment of intensely radioactive cesium would also be required. To achieve the hoped-for goal of having the reprocessing waste qualify as Class C waste (not requiring repository disposal), unprecedented separation factors would be required. (It should also be noted

that even if this goal is met for the fuel itself, the irradiated system components and similar wastes will almost certainly require repository disposal.) The necessary processes require development and demonstration.

System Optimization

Considerable further study will be required to optimize the various components of the ABC concept. One important question in that process will be: Is the accelerator that gives the concept its name needed? An alternative would be operating a similar reactor system (whichever reactor system is ultimately chosen for ABC) at full criticality, as current reactors operate (and as the MSRE operated). Does subcritical operation provide a unique safety advantage sufficient to justify consuming 10–20 percent of all the power produced by the system (meaning 10–20 percent of all its potential revenues in electricity sales) to operate the accelerator? If an accelerator is needed, would a lower-power accelerator (and a reactor closer to criticality) be sufficient?

In short, the ABC technology remains quite immature. Technical feasibility of a number of the fundamental concepts, and engineering feasibility of essentially every aspect of the system, remain to be demonstrated. A development program costing several billion dollars and lasting for decades would probably be needed before a commercial-scale ABC system could be built with confidence.

Other Issues

The ABC concept is too immature to allow detailed treatment of issues such as licensing, ES&H, and cost, but some discussion is in order.

As with other fundamentally new reactor designs, gaining required licenses and political approvals for construction and operation of an ABC system is likely to be difficult. The NRC has no experience regulating systems comparable to proposed ABC molten-salt reactors. Both the new-concept reactors and the associated reprocessing may face substantial difficulties with public acceptance. Even if all long-lived wastes could be transmuted, the storage of tanks of intensely radioactive liquid wastes would be required at the ABC site for decades or centuries, a further complication for public acceptance. If, however, claims that the system can transmute all long-lived wastes are proven, prospects for acceptance could be improved.

ES&H impacts of ABC are claimed to be low. ABC proponents point in particular to the hoped-for nearly complete elimination of long-lived species that must be disposed of in geologic repositories, which would reduce the potential ES&H impacts of repository disposal. Success in a wide array of as-yet-unproven technologies will be required, however, to ensure that ES&H impacts of ABC-system operations (including reactor operations and reprocessing) will

be acceptably low. As noted earlier, while molten-salt systems may offer a reduced risk of large accidents, there may also be a larger number of scenarios that could lead to radioactive contamination within the reactor building, leading to greater worker hazard. The intensely radioactive molten fuel would lead to high radioactivity levels throughout the reactor primary circuit, requiring development of remote-handling equipment, and potentially posing risks of significant worker exposures.

The cost of ABC systems is quite speculative, given their current state of development. If past experience is a guide, costs will be substantially higher than now envisioned. It appears unlikely that an advanced-reactor system requiring a large accelerator and advanced reprocessing will be cost-competitive with proven LWR technology as a power producer in the next several decades, while abundant low-cost uranium continues to be available. If ABC were not cost-competitive, so that a “transmutation subsidy” of, for example, a few mils per kilowatt-hour would have to be paid to make the system economically viable, the cost to transmute the world supply of spent fuel could be very large.

Some Conclusions

If ABC existed and were in use for transmutation of high-level wastes and actinides, then it could readily handle the WPu. The development and validation program for ABC would take longer than for most other options, however, and it is impossible as yet to have high confidence that this approach will succeed.

The role of the ABC is far from assured. First, it must be proved feasible—technically, politically, economically, and institutionally. If it is feasible, there seems no reason why its rare abilities should be wasted on the relatively easy task of converting WPu into spent fuel. Accelerator-based conversion might be of use after the WPu problem has been transformed into a small part of the civilian plutonium stock. Only if that large global stockpile were to be transmuted by ABC systems would it make sense to commit the WPu to treatment by accelerator-based conversion.

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Disposal of Plutonium Without Irradiation

INTRODUCTION

This chapter discusses options for immobilizing weapons plutonium (WPu) without irradiation, in ways that would make it significantly less accessible for reuse in nuclear weapons and also less hazardous to the environment. The chapter begins with an overview of the technology and a discussion of several of the key technical issues facing vitrification. This is followed by an assessment of this approach on the basis of the criteria developed in Chapter 3 and applied to reactor options in Chapter 4—including timing; safeguards and recoverability; environment, safety, and health; and cost. (A more detailed comparison of the options against the key criteria is found in Chapter 6.)

The goal of the immobilization options, enunciated elsewhere in this report, is to make the WPu roughly as inaccessible for use in weapons as the much larger and growing quantity of plutonium in spent fuel worldwide. (Lesser goals may be of interest as interim steps.) While there are a variety of materials into which plutonium could be embedded to achieve this goal, the primary case we examine in this chapter is incorporating the plutonium in borosilicate glass—a process known as vitrification. We focus in particular on glass that also incorporates radioactive high-level waste (HLW) or other fission products, so as to create a major radiation barrier to handling the material.

This case was chosen as the baseline, not as the result of a detailed comparison of alternative waste forms by the panel, but because, after decades of such comparisons for the mission of disposal of HLW, borosilicate glass has been chosen as the waste form of choice for the HLW disposal mission in the

United States and most other countries. While the plutonium disposition mission is different in some important respects, it appears desirable (to minimize costs, delays, difficulties of gaining approvals, and the like) for the plutonium disposition mission to make use of existing processes, approaches, and facilities to the extent practical, a logic that focuses attention on the borosilicate glasses already scheduled to be produced. (A brief discussion of a few of the alternative waste forms that have been proposed for this mission is also provided.) As discussed in this chapter, plutonium could also be incorporated in glass without fission products, but we do not believe this would provide a large enough barrier to reuse in weapons to be satisfactory as a final disposition option.

Much engineering development work has been done over several decades on vitrification of radioactive materials, both in the United States and in other countries. Based on this work, the general features of the technology are well known: several vitrification facilities have been built and operated around the world using different glasses, different radioactive species, and different throughputs. While glasses containing substantial quantities of plutonium have never been produced on a large scale, the key engineering parameters that would govern a large-scale WPu vitrification operation are believed to be understood. In this sense, the technical feasibility of vitrifying WPu has been adequately demonstrated.

Several important technical issues must be resolved before vitrification of WPu could move from a technical possibility to an operational reality, however. Some of these issues stem from the fact that plutonium has never been vitrified on a large scale before. In addition, further work is required to determine the best mix of plutonium, glass, and fission products for this purpose. The principal objective of vitrifying WPu is to deter its potential reextraction for weapons use. To make this deterrent most effective, it would be desirable to have: (1) small amounts of plutonium in each "log" of glass; (2) large and heavy logs that would be difficult to steal; and (3) large quantities of fission products and other contaminants in the logs to make reextraction difficult. But to minimize cost, take maximum advantage of existing vitrification programs, and meet other criteria, there may also be reasons to increase the amount of plutonium in each log, decrease the amount of fission products in each log, or make the logs smaller. Hence the selection of how much WPu to put in logs of what size, with what composition of other contaminants, requires a systematic exploration of the parameter space, taking into account engineering, handling, and cost aspects. For WPu vitrification, such a comprehensive evaluation has not yet been done.

Some of the technical issues associated with this approach are discussed below. In general the technical uncertainties associated with this approach are somewhat greater than are those surrounding the use of mixed-oxide fuel (MOX) in light-water reactors (LWRs). Nevertheless, the panel believes that *WPu vitrification represents a feasible technology that could meet the "spent fuel standard," could be available in the relatively near future (within about a*

decade), and could potentially immobilize all of the nominal 50 tons of excess WPu in glass in a relatively short time (a few years, very likely less than 10) once the vitrification campaign had begun.

Carrying out a vitrification plan such as this would require the U.S. Department of Energy (DOE) to make a major commitment to it, involving not only financial support but broad institutional support from the highest levels. Among the needed elements of such a commitment would be actions in order to retain the key technical personnel, recruit others, upgrade facilities, provide high-level “protection” from the budgetary and other institutional threats that will inevitably occur, and so on.

OVERVIEW OF THE TECHNOLOGY

The technology of plutonium vitrification is simple to explain even though it is complicated in detail. Basically, the final product is a glass in which plutonium, and other radioactive wastes in the approach examined here, are dissolved or suspended as impurities while the glass is in the melted state, after which the glass is cooled and solidified. The final form is a glass “log” usually weighing in the range of tens to thousands of kilograms, although the glass can also be made in the form of a powder or in small pieces. Once produced, the glass logs incorporating plutonium and fission products would be stored until a nuclear waste repository became available, at which time they would be emplaced in the repository as waste—without making use of the energy value of the plutonium.¹ Once the glass is produced, it is well within current technical capabilities to handle and store the plutonium-laden glass safely from the perspectives of worker safety, environmental contamination, and criticality.

The basic waste vitrification device is a melter, into which a glass powder, known as glass frit, is continuously fed, along with whatever is to be dissolved in the glass. The nonglass material can be a liquid slurry or a dry feed. The melter melts the frit and dissolves (or suspends) the nonglass material in the

¹ Plutonium-laden glass could also be stored for later recovery of the plutonium for use as reactor fuel, or the decision could be postponed. Two factors should be kept in mind, however: (1) both the initial vitrification of the plutonium with fission products and its eventual separation from that form would be costly, creating a significant disincentive to vitrifying the material if the intent is to recover it in the foreseeable future; and (2) the proliferation resistance provided by the fission products in a glass combining plutonium and HLW will decay with time (with the radiation barrier declining by roughly half every 30 years), so the glass would not be a desirable form in which to store the plutonium at readily accessible locations for many decades or centuries. (This is also true of spent fuel.) As with the reactor options, a guarantee of near-term availability of a geologic repository is not critical to the viability of the vitrification option (because the glass could be stored safely for decades, offering a proliferation resistance comparable to that of spent fuel stored for a comparable period). For the reasons outlined above, however, the panel would not recommend the vitrification option if the intention was to recover the material in the near term for use as reactor fuel, rather than to eventually dispose of it in a geologic repository.

glass (Marples 1988). The glass is heated and kept in a molten state, usually by joule heating in a ceramic melter (in which a large alternating current is passed through the molten glass itself) or inductive heating in a metallic melter. Glass product is continuously or intermittently extracted in molten form from the melter as more material is fed in, and is poured into a mold where it cools into a solid form—typically a large glass log. It is beyond our scope here to discuss the several different approaches to heating the melter, to feeding in the frit and the impurities, to accomplishing effective dissolution, and to cooling the final glass log.

Melters to produce the requisite glass can be large (several meters in diameter and in height), small (substantially less than a cubic meter), or in between. Technologies exist for small-scale modular melters that could be built and shipped to the sites where the plutonium now resides (including sites in Russia). To incorporate large quantities of fission products (whether in glass or other waste forms), however, would require a remote-handling facility with adequate protection for workers against the intense radiation—meaning that whether the melter itself is large or small, it will be part of a facility which is large, expensive, and complex. Thus it is likely to be highly desirable to make use of existing facilities to the extent practical. In some options, plutonium might be incorporated in HLW glass logs already scheduled to be produced for HLW disposal, using the same facilities (with some important modifications required). In other options, plutonium might be incorporated in glass logs in addition to those previously scheduled to be produced, either using existing vitrification facilities or new ones constructed for this purpose (probably within an existing remote-handling facility such as a reprocessing canyon or vitrification plant). The issue of what facilities might be used for the WPu disposition mission is an important one, discussed later in this chapter.

The answers to several critical technical questions will determine the feasibility of safely and economically vitrifying WPu. The principal issues can be grouped into three main categories:

- designing a form that would meet the “spent fuel standard” with tolerable production costs, schedules, and risks, including options for the physical, chemical, and radiological characteristics of the glass;
- difficulties of producing the chosen glass form, including plutonium handling and criticality issues in both preprocessing and during the vitrification process itself, and how much plutonium can in fact be loaded into the glass; and
- the suitability of the resulting glass forms for deep geologic disposal.

Before discussing each of these categories of issues, it is necessary to discuss briefly borosilicate glass and several of the possible alternatives to it for the WPu vitrification mission.

THE CHOICE OF WASTE FORM

After decades of study of scores of different possible waste forms, most of the international community has settled on borosilicate glass as the waste form for immobilizing high-level radioactive wastes. Borosilicate glass is being used for this purpose, or is planned for use, in the United States, France, Great Britain, Japan, Germany, and other countries. The choice of borosilicate glass is based on several favorable properties (Marples 1988): it can incorporate almost all of the important radioactive fission products dissolved as oxides; it can contain waste at levels as high as 20 or even 25 percent by weight; it is tolerant of widely varying waste compositions; it is reasonably resistant to leaching by water; it is relatively resistant to radiation damage; it can accommodate the chemical changes that occur when the waste impurities decay radioactively; and the production process is relatively simple and reliable, with a reasonably low formation temperature, and with a glass product that is not corrosive to the process equipment, unlike phosphate and lead phosphate glasses.²

Providing a sufficient radiation barrier to meet the spent fuel standard for 50 tons of plutonium will require tens or hundreds of millions of curies of radioactivity, a small but significant fraction of the total amount of separated radioactive fission products currently stored in the United States. Hence, incorporating these fission products with plutonium into waste forms *other than* the borosilicate glasses on which the HLW disposal program is now centered would represent a substantial modification of that program, with the attendant potential for delays and uncertainties for both the HLW disposal program and the WPu disposition program. Nevertheless, it is worth briefly considering a few of the most prominent of the alternate waste forms that have been proposed for the WPu disposition mission—some of which are unique to the WPu disposition mission, and some of which have been considered for the HLW disposal mission in the past. DOE's Office of Fissile Materials Disposition is studying a wide array of possible waste forms for immobilization of WPu, including those discussed below and numerous others.

Phosphate Glass

Phosphate glass is used in the Russian HLW vitrification operation at Chelyabinsk, which raises the question of whether this waste form might be appropriate for vitrification of WPu in Russia (if such an option were ever seriously pursued by the Russian government). Russia is essentially alone in the world in choosing phosphate over borosilicate glass for disposal of HLW; the choice was apparently made in part because of the lower formation temperature

² There are other waste forms that appear to have leaching characteristics superior to borosilicate glass, but there are no waste forms that have been demonstrated to be superior overall, considering all these criteria.

and therefore lower cost of production associated with phosphate glasses. Phosphate glass is less appropriate than borosilicate glass for the WPu disposition mission because it is less durable and less resistant to criticality if WPu is embedded in it (it does not include large quantities of neutron-absorbing boron as a basic constituent). Neutron absorbers could be added to the phosphate glass to control the latter problem, but this possibility has not been examined in detail. The panel believes that if WPu were to be vitrified in Russia, switching to a borosilicate glass would probably be a better approach than continuing to rely on the phosphate glass currently produced (Diakov 1992). Although borosilicate glasses have been studied in Russia, the panel is not aware of any Russian plans to switch to borosilicate glasses, or of any estimates of the cost and schedule for modifying the Russian facility to produce borosilicate rather than phosphate glasses.

Synroc

A synthetic rock known as "synroc" was developed as a possible HLW form in the United States years ago, but was ultimately abandoned in preference for borosilicate glass. Some work on the concept has been pursued in Australia in the intervening years. The choice of borosilicate glass was based on a number of technical issues related to synroc that have not been resolved, including the larger amount of hot-cell processing required to produce the synroc, and the greater flexibility of glass in incorporating a wide range of wastes. The latter concern might not be a serious problem in the case of plutonium disposition, if the radiation barrier were to be provided by fission products such as the cesium-137 stored at Hanford, rather than by HLW combining a range of products. Nevertheless, it does not appear that synroc has any unique advantages for incorporating plutonium compared to borosilicate glass that would suggest that it would be clearly superior for the WPu immobilization mission.

Cements

Some authors have proposed a variety of cement compositions for disposal of radioactive wastes. For disposal of low-level wastes (LLW), this approach has considerable promise, but the consensus of the international community is that for containment of HLW, glasses are superior to cements.

Pyroprocessed Metals

Some analysts have proposed that the "pyroprocessing" approach that was to have been used to prepare fuel for the U.S. integral fast reactor (IFR) program be used instead to combine WPu with spent fuel into a waste form that would meet the spent fuel standard. As part of a redirection package in the wake of the

cancellation of the IFR, DOE plans to provide several million dollars for examination of this approach by the Argonne National Laboratory (ANL).

For the IFR, the pyroprocessing approach (based on molten-salt dissolution and precipitation) would have been used both to recycle IFR fuel and to reduce spent oxide fuels from LWRs to a metal form, from which the short-lived fission products would have been separated. This metal, containing substantial quantities of uranium, plutonium, other actinides, and long-lived fission products, then would have been introduced as fuel for the IFR. For WPu disposition, the concept is to use the pyroprocessing approach to reduce a mixture of oxide spent fuel and WPu to a metal form; in this case, for maximum proliferation resistance, the fission products from the spent fuel would not be separated but would remain in the metal product.³

This approach has several disadvantages that the panel believes effectively rule it out as a serious competitor for the near-term plutonium disposition mission:

- First, to meet the spent fuel standard would require the plutonium to be mixed with a large amount of material (roughly 1,000 tons for 50 tons of plutonium, if the product was to be 5-percent WPu by weight); this would require building, in effect, a substantial reprocessing plant. The costs, delays, and approval difficulties involved in building such a facility would be substantial.

³ For a detailed discussion of the pyroprocessing flow sheet that was planned for the IFR, see National Research Council (forthcoming). Given the very different chemistry of cesium and actinides such as plutonium and uranium in the molten-salt system, it may be difficult to design the process so that a large fraction of the fission products are retained. Some of the experiments done on pyroprocessing LWR fuel, however, found that as much as 90 percent of the intensely gamma-active cesium was retained in the metal precipitate. ANL researchers describe this result as surprising and inconvenient for recycle but desirable for the immobilization concept (Argonne 1994). For a discussion based on previous ANL reported results in which only about one-third of the cesium was retained, which concludes that the radioactive barrier from pyroprocessed materials would be substantial (though less than in the case of spent fuel or vitrified glass comparable to planned Savannah River Site waste glass), see Lyman (1994). Lyman advocates consideration of mixing plutonium with spent fuel using the AIROX (Atomics International Reduction Oxidation) process, in which oxide spent fuel assemblies are punctured and subjected to a series of oxidation and reduction steps, which effectively disassemble them and convert the pellets back to powder; this highly radioactive powder could then be mixed with WPu. Lyman estimates that a construction of a plant for this purpose would require 5–10 years, at a cost in the neighborhood of a billion dollars, and could combine 50 tons of WPu with oxide spent fuel in a decade of operation. A number of the arguments advanced here, however—the need for a time-consuming development program, the need for a large remote-handling facility that does not currently exist and whose licensing would be highly uncertain—would also pertain in this case. And as Lyman (p. 21) points out, “much research and development” would be needed before it would be clear whether processes were available to safely fabricate the resulting product into fuel or into an acceptable waste form for geologic disposal.

- Second, at the time the IFR was canceled, reduction of LWR oxide spent fuel to metal was the least developed part of the pyroprocessing scheme, with only a few experiments completed. Feasibility at an engineering scale had not been demonstrated (National Research Council forthcoming). Demonstration and validation of this technology would involve additional costs and delays, with no guarantee of success in providing an economical and effective WPu disposition option.
- Third, it is doubtful that it would be desirable to process the WPu in this way unless the resulting intensely radioactive product were suitable for geologic disposal, for if it were not, the material would eventually have to be processed yet again to prepare it for such disposal or for use as fuel (a step that would also require a large remote-handling facility). As the output of the pyroprocessing was never intended to be a waste, it remains uncharacterized as a waste form. Characterization and certification of waste forms for radioactive isotopes that will last many thousands of years is a lengthy and painstaking process that would almost certainly introduce additional delays. It appears unlikely that a metal matrix such as that produced by the pyroprocessing would be a suitable waste form for the chemical environment of Yucca Mountain: the metal, once exposed to water, would be expected to undergo both hydration and oxidation reactions, breaking down its structure and releasing the radioactive materials it contained.

For all these reasons, the panel believes this approach is not competitive with either vitrification in borosilicate glass or the use of MOX in existing reactors, both of which would be likely to involve lower costs, lower technical uncertainties, and shorter delays.

Plutonium-Beryllium Combinations

Specialists at the Los Alamos National Laboratory (LANL) have proposed that plutonium be combined with beryllium in such a way that the alpha-n reactions, followed by multiplication of the neutrons in the plutonium, would create enough neutron radiation for the material to be "self-protecting" by Nuclear Regulatory Commission (NRC) and International Atomic Energy Agency (IAEA) standards (100 rads/h at 1 meter) (Toevs and Trapp 1994). In effect, what is proposed is to create extremely large plutonium-beryllium neutron sources. The materials envisioned would be between 10 and 30 percent plutonium by weight. (Higher plutonium percentages, combined with compact geometries, would bring the plutonium closer to criticality, resulting in more neutron multiplication and a higher intensity radiation field.)

While such a plutonium-beryllium combination would be more self-protecting than pure plutonium metal or oxide, it appears extremely unlikely that it would be possible to design such materials in a way that would fully meet

the spent fuel standard without coming perilously close to criticality. As currently envisioned, the radiation field from such materials would be far lower than that from spent fuel until the spent fuel is many decades old, while the plutonium content by weight would be far higher.⁴ Thus these materials would appear to be a substantially more attractive target for theft by a potential proliferator—or for reincorporation into weapons by a weapons state—than plutonium in spent fuel. Moreover, as with the pyroprocessed waste-form concept, it appears unlikely that such a material would be considered an acceptable waste form for ultimate disposal. Combining plutonium and beryllium would result in a mixed waste, creating very difficult regulatory issues in the United States. If these materials were not suitable for disposal, the additional costs and complexities of eventually processing this material to some other form would eventually have to be borne. The cost of producing these materials, however, might be significantly less than the cost of WPu disposition using the vitrification or MOX fuel options.

TECHNICAL ISSUES FACING VITRIFICATION

Options for a Proliferation-Resistant WPu Glass

The principal objective of WPu vitrification is to place barriers in the way of any party wishing to reuse the WPu for nuclear explosives. Nevertheless, just as with the reactor options (except those designed for near-complete elimination) reextraction of the plutonium is not precluded from any of the glass forms under consideration. The ease or difficulty is only a matter of technical skill, access to facilities, money, and time.

As with spent fuel, the chemical processes needed to extract plutonium from glass are not especially difficult or obscure. The primary difficulty arises from coping with the radioactivity of the fission products also embedded in the glass. To meet the spent fuel standard, the amount of radioactivity would have to be sufficient to require remote operations, such as those used in reprocessing plants. The chemical processes for extracting plutonium from glass would be conceptually similar to those for extracting plutonium from spent fuel. Most types of glass can be easily dissolved in suitable acids, after which separating plutonium and the other impurities requires a series of chemical processing steps that are well known. The difficulty of subsequent steps to purify the plutonium itself would depend on what other impurities, radioactive and nonradioactive, were present in the mix (see below). Other options for recovering the plutonium

⁴ In the very long term, after geologic emplacement, the radiation field from these materials would in fact be substantially higher than that from spent fuel, as the radiation in spent fuel is primarily caused by species with half-lives of the order of 30 years, while the radiation from these materials originates from the 24,000-year half-life plutonium.

from the glass are also widely known, including electroprocessing and plasma-arc processing.

The difficulty of diverting the plutonium-laden glass, transporting it to where it could be processed to extract the plutonium, and conducting that processing would depend on the size of the glass logs, the amount of plutonium in each log, and the radiological and other contaminants also incorporated in the glass. As noted above, no comprehensive trade-off analysis among these variables to select the optimum glass form for cost-effective proliferation resistance has yet been done. The difficulty of these various steps, and the barriers that particular features of the product could pose to use in weapons, would also depend importantly on the skills and resources available to the party that wanted to extract the plutonium for weapons—in particular whether that party was a major weapons state like the United States or Russia, or whether it was a non-weapons-state or nonstate group.

There are several different glass-product options that need to be differentiated. Some differ in their chemical or radiological composition and some in their physical size.

Size

Three general classes of physical size have been discussed with the panel by vitrification experts:

- Small glass frit, beads, powders.
- Small glass logs of 35- to 70-liter size (one liter of glass weighs about 2.8 kilograms; kg), typically produced by a small melter. These 100- to 200-kg logs are heavy enough so a single person cannot carry away a log, but small enough for hijacking with a forklift.
- Large logs, about 3 meters (m) long, 60 centimeters (cm) diameter, weighing about 1,700 kg, plus 200 or more kg for the surrounding canister; too large for forklift handling and regularly available transport. This is the glass form currently planned as the output from U.S. HLW vitrification programs.

Of course, other options are available. Obviously, the larger the size the more impediment to theft and easy post-theft handling. The large glass logs to be produced in vitrification operations at the Savannah River and Hanford sites, about 3 m and over 2 tons each, cannot easily be moved. Especially when combined with radioactivity sufficient to require remote handling, size can be a substantial handling problem for at least some potential parties wishing to reuse the WPu. As an example, the large HLW-laden logs to be produced at Savannah River will be handled individually by a specialized vehicle weighing over 100 tons.

Plutonium Concentration

It would be desirable if the plutonium concentration in the logs were low enough that more than one log would have to be stolen to recover enough material for a single weapon. This would imply a very large number of logs, however. If the logs were limited to 2 kg of plutonium each, for example, disposing of 50 tons of WPu would require the production of 25,000 plutonium-bearing logs, more than the entire amount scheduled to be produced in the U.S. HLW disposal program. By contrast, if the logs contained some 20 kg of plutonium (roughly 1 percent by weight for a 2-ton log like those to be produced at the Savannah River Site [SRS], comparable to the percentage of plutonium in spent fuel), only 2,500 logs would have to be produced. In that case, incorporating plutonium into a fraction of the logs already scheduled to be produced at SRS (see below) would be sufficient, without requiring production of any additional logs. Higher concentrations of plutonium may also be possible (see below), but do not appear to be necessary for most options.

Radioactive Spiking

Three general classes of glass compositions vis-à-vis their radioactive content have been discussed with the panel by vitrification experts.

- *Glass with plutonium and HLW.* This glass, spiked with high-level radioactive waste that is the detritus of defense processing activities, can be made sufficiently radioactive that handling it would be extremely hazardous to life (lethal external dose in minutes). The nominal case for the HLW glass logs scheduled to be produced at SRS is a radiation field of 5,200 rem/h (roentgen-equivalent-man per hour) at the container surface (roughly 900 rem/h at 1 m), but in fact nearly all of the logs will be less radioactive than this, many of them having roughly half this dose rate (Westinghouse 1994).
- *Glass with plutonium and specific fission-product spiking,* such as spiking with the cesium-137 (Cs-137) now stored at Hanford. This would create a similar radioactive barrier, but being a single chemical constituent might allow the use of simpler chemistry and preprocessing than use of the complex HLW in storage at Hanford and SRS. Sufficient Cs-137 (about 50 million curies) is stored at Hanford to produce 500 two-ton logs (which could incorporate 50 tons of WPu if the concentration were a high 5 percent by weight) with a radiation field of 2,000 rem/h at 1 m. DOE currently has no definite plans for disposition of this Cs-137.
- *Glass with plutonium only.* This glass is not very radioactive, and the radioactivity presents little if any impediment to theft or post-theft

handling (except for the hazards of the plutonium itself, particularly if released in an aerosol form).

When spiked with either defense HLW or fission products such as Cs-137, the glass can be made so radioactive (hundreds or thousands of rem/h at 1 m) that lethal doses would arise quickly (McKibben et al. 1993). This would then require remote-handling operations, not only in the handling of the logs but in the initial processing stages as well, substantially increasing the proliferation resistance of the product. The radioactive decay rates of both the defense HLW and the Cs-137 spiking seem to follow roughly the 30-year half-life of Cs-137 (Wicks 1993). Thus a 2,000-rem/h dose rate would decay to some 60 rem/h (below current regulatory standards for being "self-protecting") in 150 years. In another 100+ years the radioactive-handling problem would be small.

It is important to note that intense radioactivity is a major barrier not only in the handling of the logs by a potential reextraction team, but also in the initial chemical processing of the glass and of the plutonium-laden chemical product of glass dissolution. The need for complex remote-handling technology, the problems with equipment becoming and remaining intensely radioactive, and the storage difficulties all contribute to substantially greater costs and greater occupational risks. Many of these costs and risks, however, will have to be borne in any case to dispose of the existing fission products at DOE sites; the important questions in this case are the net additional costs and risks that would be associated with adding plutonium to the HLW vitrification campaigns that will be required in any case.

Chemical (Nonradioactive) Spiking

The plutonium-laden glass could be spiked with nonradioactive chemicals similar to plutonium (actinides, lanthanides) that would make chemical extraction of the plutonium more difficult. (See, for example, Makhijani and Makhijani 1994, Simonson et al. 1994.) After discussion of this possibility with a number of experts in plutonium chemistry, however, the panel concludes that the benefit of available chemical spiking techniques would be modest. Well-known chemical processing steps—within the capability of nearly all states and some nonstate groups—are available that could separate the plutonium from essentially any elements that might be added to the glass.

In particular, if the glass had no fission product contaminants to create a radiological barrier, extracting WPu from it would be substantially easier, whatever the mixture of chemical contaminants added, than extracting WPu from highly radioactive glass requiring remote handling. For states such as the United States and Russia, a chemical barrier alone would be insignificant. For others, it is the panel's judgment that most potential proliferators with the technical expertise, personnel, and the organization required to produce an operable weapon

from separated plutonium—a substantial technical task in itself—would also be able to extract plutonium chemically from a glass log not spiked with radioactivity. Having to do so would not substantially increase the overall time and cost of the project of building a weapon.

Thus the principal advantage of vitrifying plutonium without an added radiation barrier would be to put the plutonium into a form that was large and heavy, and therefore more difficult to steal without detection.⁵ If the plutonium was nevertheless diverted, the fact that it was in vitrified form would be only a modest additional obstacle on the path of its use for production of a nuclear explosive.

Some analysts have pointed out that Russian officials strongly oppose options that would throw away the plutonium's energy value, and have suggested that vitrification in a plutonium-only glass from which the material could be readily reextracted for fuel at a later time might be a more acceptable alternative for Russia. This approach would only be viable, however, if the plutonium-only vitrification provided a major improvement in proliferation resistance, and the panel has not been presented with evidence that would suggest that this is the case. Thus, the panel believes that *putting WPu into such a simple glass form would not constitute enough of a barrier for this to be a serious contender as a long-term disposition option*. Therefore, this simpler option is not discussed extensively here.

Nevertheless, the panel recognizes that there might be scenarios in which carrying out such an approach might be considered worthwhile as an *intermediate step* before revitrifying WPu with fission products. For example, substantial quantities of plutonium are now stored in forms that are not suitable either for long-term storage or for transport; vitrifying these forms of plutonium at the sites where they are now stored in order to produce a safe form for transport to another site where they could be revitrified with fission products could be an attractive disposition approach for these materials. Indeed, in some circumstances it might be judged desirable to undertake a relatively rapid campaign to vitrify all of the WPu without any other radioactive species, thereby inexpensively and rapidly transforming the existing metallic WPu pits and ingots into a glass form, but with a second follow-on vitrification step clearly in mind in which the WPu-laden glass would be *revitrified* later with highly radioactive species included to provide the desired deterrent.

⁵ A significant part, though not all, of this advantage could be achieved more quickly and cheaply, without the safeguards issues involved in processing plutonium, by storing plutonium in similarly large and heavy containers, designed to make it difficult to remove the plutonium without detection. For example, some analysts have proposed storing plutonium in multi-ton dry casks similar to those planned for spent fuel storage.

Summary of Proliferation Resistance Issues

Clearly a wide range of factors would affect how difficult it would be for a particular party to acquire the glass logs produced in this process and extract WPu from them for weapons, including characteristics of both the party concerned and the logs themselves. The intense radioactivity of the logs would pose the greatest barrier. This barrier does not scale linearly with the dose rate: at a certain dose rate (not well defined), remote-handling operations become necessary to process the glass, drastically increasing the cost and complexity of the operation. Beyond that point, increasing radioactivity would continue to pose additional difficulties, increasing the difficulty of acquiring and transporting the logs and so on. To meet the spent fuel standard, enough radioactivity would have to be incorporated into the glass to require remote operations—but how much benefit would be derived from specific further increases in dose rate beyond that point requires further study.

It is important to note that the WPu in this option would remain weapons-grade, unlike the plutonium that could be extracted from most types of spent fuel. Potential bomb-makers would prefer weapons-grade material to reactor-grade material. As noted elsewhere in this report and in the report of the parent committee (NAS 1994, pp. 32–33), nuclear weapons that would have assured yields in the range of 1 kiloton can be constructed with reactor-grade material, using technology similar to that required to produce simple weapons with weapons-grade material. With more sophisticated designs, higher assured yields could be achieved.

Overall, the panel judges that the plutonium in glass comparable to that scheduled to be produced for HLW disposal in the United States would be approximately as inaccessible for weapons use as plutonium in commercial spent fuel.⁶

**Difficulties of Producing Glass
with WPu and Fission Products***Plutonium-Loading Capacity of the Glass*

A significant technical question is how much plutonium and other waste impurities can be placed into the glass. For those options in which WPu glass logs would be produced above and beyond the HLW glass logs already scheduled for production, the overall cost will be quite sensitive to the number of logs that must be produced (determined by the amount of plutonium that can be put in each). By contrast, if the plutonium is to be added to HLW glass logs already

⁶ A security advantage enjoyed by the glass logs over much spent fuel is that the major vitrification operations—and the interim storage of the glass logs prior to emplacement in a geologic repository—would be at major nuclear weapons complex sites.

scheduled to be produced for HLW disposal, the cost of the overall operation will be only modestly sensitive to the number of logs that must incorporate plutonium, unless the level of plutonium that can be added to each log is so low that the overall number of logs to be produced must be significantly increased.

The amount of plutonium that can be put into glass is limited by several factors: the solubility of plutonium in glass, the need to prevent formation of a form that could sustain an accidental chain reaction (known as criticality), and the suitability of the glass product for geologic disposal.

Westinghouse Savannah River Company (WSRC) has quoted experimental work by Plodinec and Wiley (1979) in which 7 percent plutonium was successfully dissolved in a bench-scale test in a glass composition somewhat different from that now planned for use at SRS. (All percentage loadings quoted herein will be percent by weight.) SRS personnel and analysts at the Lawrence Livermore National Laboratory studying the issue for DOE's Office of Fissile Material Disposition believe that even higher loadings may be possible. (The HLW-laden glass to be produced at the Defense Waste Processing Facility [DWPF] at SRS will contain 20–30 percent fission products by weight.) Further experimental work would be required to demonstrate that loadings of several percent by weight could be achieved with current glass compositions in large-scale production, while producing a glass meeting acceptable repository performance criteria (see discussion of repository issues, below).

Criticality also does not appear to place restrictive limits on the amount of plutonium that can be dissolved or suspended in the glass because of the large quantity of neutron-absorbing boron in the glass. WSRC has published simple calculations suggesting that *if the glass were homogeneous*, it would not be critical at plutonium loadings below 15 percent (McKibben et al. 1993). Given the extreme importance of preventing criticality, however, it is essential to take into account possible inhomogeneities in the glass, as well as possible accident scenarios. The issue of possible criticality during preprocessing and in the melter is discussed in more detail below.

In the case of options in which the WPu would be added to already planned vitrification campaigns, concentrations in the range of 1 percent by weight plutonium would be sufficient to allow all of the nominal 50 tons of excess WPu to be incorporated in the logs already scheduled to be produced at a single site (either Hanford or Savannah River), without any increase in the number of logs. Given that lower concentrations are better from the point of view of the proliferation barrier, there does not appear to be any substantial incentive to move in the direction of higher plutonium concentrations for these options. Only in the case of options in which the plutonium would be incorporated in a separate vitrification campaign, where the processing costs would scale directly with the number of logs produced, would there be any incentive to pursue higher plutonium loadings. Even in that case, it is likely that for reasons of both proliferation resistance and repository performance, loading in the range of one to a few per-

cent will be desirable. Hence, it appears that neither solubility nor initial criticality are likely to be major constraints on achieving the plutonium loadings of most interest.

Handling Plutonium in Upstream Processing and in the Melter

The chemical processing involved in preparing HLW for vitrification is quite complex, and the intense radioactivity of the HLW complicates the problem further. (Indeed, discovery of a series of difficulties with the flow-sheet for chemical preprocessing of wastes at Savannah River has delayed the DWPF project there by several years.) Adding plutonium to this process would raise additional complications, both because of the environment, safety, and health (ES&H) hazards if plutonium were to be accidentally released in aerosol form, and, even more important, the need to design all the relevant processes to ensure against possible criticality.

Plutonium metal would in most cases have to be converted to other forms to prepare it for vitrification.⁷ Two principal approaches are possible: (1) oxidation (roasting, calcining) of plutonium into an oxide that is fed dry into the melter, or (2) creation of an acidic plutonium solution (McKibben et al. 1993). The oxide form has a significant biological hazard potential if released in aerosol form. The dry-feed approach associated with the oxide form, however, makes it easier to transport the material. The panel believes that the pros and cons of each approach need to be explored.

There are also several approaches for how the WPu might be combined with the HLW and with the glass. The WPu (in oxide or acid solution form) might be mixed with the liquid HLW solution prior to its introduction into the melter, or the two might both be added to the melter separately, to be combined as they each dissolve in the glass. Alternatively, the WPu might be "previtrified" without HLW, and the resulting plutonium-bearing glass added to the melter as a frit. Whatever the approach, detailed engineering work will be required to ensure against criticality, and against aerosolization of the plutonium, throughout the process.

Another unanswered question is whether the design specification for the plutonium-laden glass will require total plutonium dissolution in the glass or can accept some fraction of undissolved plutonium. In some cases, allowing some undissolved plutonium to remain could simplify mixing, and upstream process-

⁷ Specialists from the Oak Ridge National Laboratory, however, have applied for a patent for a process, known as the Glass Material Oxidation and Dissolution System (GMODS), by which plutonium metal could be vitrified directly without preprocessing (see Forsberg et al. 1994). While such a process could potentially eliminate one expensive step in WPu vitrification, the concept is still at an early stage of development. It uses a lead-glass material for the dissolution, which does not seem to be a suitable form for permanent disposal in a geologic repository.

ing, allow melter operation at lower temperatures, and potentially increase throughput.

These issues in the upstream processing, particularly criticality, require careful engineering that has not yet been done. But the panel believes that resolving these issues is within current technical capabilities.

It is also essential to avoid criticality in the melter as the WPu glass is being produced. This depends on the glass composition (including the fraction of plutonium being added to the glass), the melter design, and possible inhomogeneities in the glass.

A variety of different factors in melter design affect the criticality risks that could arise from vitrifying substantial quantities of WPu. Principal among these are the size and shape of the melter and what provisions are made for ensuring the glass in the melter is well mixed (particularly whether the glass is stirred or unstirred).

Currently both the DWPF and, apparently, Hanford's not-yet-built Hanford Waste Vitrification Project, plan to use very large, unstirred melters (Omberg 1993). Most of the major U.S. engineering work on melter technology in recent years has gone into designing and building the DWPF facility at SRS, where a major (20- to 30-year) campaign to vitrify defense-produced HLW is scheduled to begin in 1996 (McKibben et al. 1993). The DWPF melter, remote-handling facilities, and all the supporting auxiliary equipment are designed for this application. Based on material presented to the panel (McKibben and Wicks 1993, 1994; Gray 1994), it appears that the existing first-generation melter at DWPF is *not* well suited to vitrifying plutonium for a variety of reasons including both criticality concerns and offgas-system questions, issues which we will address in turn.

Large melters like the DWPF have the advantage of large throughputs for a single melter and single feed stream. They have the disadvantage, for the HLW disposal mission, that if a problem arises with the single large melter, production is stopped completely. The amount of glass in the melter at any one time is very large, moreover, and the residence time in the melter is very long, increasing the potential for inhomogeneities in the glass (such as settling of heavier elements or precipitation of some chemical species). In France, by contrast, several smaller melters are used in parallel. The consensus of the vitrification community appears to be moving in the direction of smaller melters. One key process advantage seems to be that these smaller melters can be shut down, cleaned out, and restarted in a matter of a day or two, whereas the shutdown-restart time for the larger DWPF melter system is much longer (Wicks 1993).

For the WPu disposition mission, large melters have the additional major disadvantage that a very large amount of plutonium would be in the melter at any one time, increasing criticality concerns. In the DWPF, for example, if the plutonium were 1 percent by weight in the glass, the melter would contain well over 100 kg of plutonium at any one time. In addition, the DWPF is an unstirred

melter. In such a melter, the potential exists for several problems such as multiple-glass-phase separations, glass-impurity separations, physical stratification (within the glass, as well as heavy material collecting by precipitation at the bottom), and so on—though it is expected that convection in the molten glass will provide significant mixing. Although some information exists on stratification of other heavy species, the panel learned that almost nothing is now known about potential plutonium stratification in any of the melter designs now under active exploration or already in use. Finally, the DWPF is also a “siphon-pour” melter, rather than a “bottom-pour” melter: the glass leaves the melter through a pipe roughly a meter above the bottom of the melter, rather than through a hole in the bottom, so that there will be a meter of glass at the bottom within which heavier species such as plutonium might build up and be drawn off only partially. In short, it appears that the current DWPF melter would not be appropriate for the plutonium vitrification mission.

Each DWPF melter, however, will have a useful life of only a few years and must then be replaced. A second melter similar to the first has been built for this purpose. If a requirement were developed to design a subsequent melter (the third or fourth for DWPF) for vitrifying plutonium, the panel believes that such a design would be feasible. The issues in developing such a design are technically challenging and have only begun to be explored, but there do not appear to be any inherent technical obstacles.⁸

A melter designed for WPu vitrification would probably be smaller (perhaps with more than one melter operating in parallel) and designed specifically to ensure criticality safety (perhaps with an “inherently safe” geometry that would prevent criticality). A WPu vitrification melter might also incorporate stirring of the molten glass. Stirred melters would be expected to have less serious difficulty with criticality via stratification than unstirred melters. The common practice in criticality assurance, however, is to avoid relying principally on mechanical devices such as the mechanical stirring itself as the means for avoiding criticality; indeed, if a mechanical stirrer that could fail were the only barrier to criticality in the melter, the design would not provide sufficient assurance of criticality safety, and it is the panel’s judgment that it would not pass regulatory review.⁹ Stirred-melter technology is relatively new, and several issues require additional work before it could be proposed for actual production use for WPu vitrification. Small stirred melters, which have been demonstrated at smaller scales under test conditions (although not with plutonium), can operate in a glove-box-type environment with smaller throughputs, about 30 kg/h compared to about 100 kg/h for the larger first-generation DWPF melter. One

⁸ It is possible, however—though we believe it to be unlikely, if sufficient ingenuity is applied—that the throughput of a melter designed for safe vitrification of WPu would be so compromised as to make the operation impractical.

⁹ A design that offered additional protection against criticality—such as stopping the addition of material if the stirrer failed—could potentially offer sufficient assurance of safety, however.

important advantage for plutonium vitrification would be that, if plutonium is to be incorporated in the glass without any other radioactive species in an interim step, a glove-box-type operation could be used without the complication of the remote-handling equipment that makes the DWPF operation so complex.

The stirred-unstirred trade-off issues (and the large-or-small melter trade-off issues) are complex and require further investigation, but there is a basis for confidence that some form of melter technology can be made available that can provide options for critically safe vitrification of WPu. Four small melters in parallel would provide the capacity of the proposed large melter, would probably be cheaper and quicker to develop and perfect, and might, by virtue of modularity, be less costly in operation. We recommend, therefore, that small melters be pursued for future replacement of the large melter now in place at DWPF, particularly if plutonium vitrification is to be carried out in that facility.

Another criticality concern arises in the system that filters the gases released from the melter—known as the “offgas” system—where the potential exists for the accumulation of plutonium in the ducting, filters, scrub solution, etc. The volume and character of the offgas depend critically on whether the melter feed is dry or a slurry: with dry feed the offgas volume is much lower and the problem therefore likely to be easier to resolve (McKibben et al. 1993). The design specification for the melter offgas system for DWPF, assuming the current plan for a DWPF campaign to vitrify HLW (but not plutonium), has concentrated on Cs-137 in the offgas because it is believed by the designers to be the most difficult species that drives the design (Sullivan 1993). The panel believes that a major design and testing effort for a modified offgas system is likely to be required before plutonium vitrification could proceed. This will include developing information on plutonium volatility as a function of various parameters.

Repository Issues

As of today, no waste-form criterion or standard exists that would govern the properties of a plutonium-laden glass destined for deep geologic disposal in the proposed Yucca Mountain repository.¹⁰ Some (interim) DOE agreements exist between different DOE offices, but the panel believes that nothing now exists that is well founded on a thorough examination of all of the issues. Several issues are involved, including waste leachability aspects, waste physical form, heat generation rates, chemical properties of the waste, and criticality issues.

All of these issues must be, and are being, addressed for the spent fuel that would be the Yucca Mountain repository's principal waste form, including criticality. In that sense, there is nothing unique about plutonium-laden glass. But

¹⁰ Standards for disposal of wastes in the Waste Isolation Pilot Plant (WIPP) have recently been issued.

plutonium-laden glass has never previously been studied. Serious criticality issues could arise for plutonium-laden borosilicate glass, in part because the boron—needed for criticality control in the glass—leaches out much more rapidly than the plutonium (see Plodinec and Wiley 1979, Bibler et al. 1985), leaving significant potential for plutonium criticality in the waste over subsequent millennia even for modest plutonium loadings in the glass.¹¹ The idea of mixing into the glass an effective neutron absorber with a very low leach rate, such as gadolinium, to assure criticality control even in the absence of boron, has been proposed but not yet explored in detail (McKibben 1993, Omberg 1993, Simonson et al. 1994). The repository criticality issue is described in more detail in Chapter 6.

The panel believes that other glass leachability issues besides criticality are also important and require research. As mentioned above, it is likely that the migration of plutonium from the glass in the repository into underground water is solubility-limited for both steps of what is thought to be a two-step process: first, plutonium leaching from the glass to form a separate precipitate, and second, the dissolution of that precipitate in the groundwater. The various rates are not yet well enough understood, however. The migration of plutonium into the environment may not be a problem, for example, if the overall transport time is longer than several times the 24,000-year half-life of Pu-239.

If the glass is loaded with other radioactive species (HLW or Cs-137, for example), the process governing the leachability of each important species needs to be understood. The panel believes that there may be glass-design approaches, to modify these leach rates if necessary, that remain unexplored. (The answers to these questions will vary, perhaps significantly, depending on the specific glass composition.) As noted in Chapter 6, there are reasons to believe that the addition of plutonium, which appears to act chemically to strengthen the chemical network of the glass, may even modestly improve glass performance in containing other radioactive species.

Questions exist as to the design and performance specifications for the outside canister that would surround the glass in the repository. Because consideration of WPu-laden glass is only recent, no final canister design has been specified yet. Indeed, the standards that such a canister must meet are also in a state of flux now due to the current reevaluation of the Environmental Protection Agency (EPA) standards for Yucca Mountain (see Section 801 of the Energy Policy Act of 1992).

An issue that has not yet been sufficiently explored is whether the alpha-loading of heavily plutonium-laden glass might compromise the glass's physical

¹¹ If the glass were not intended for disposal in a geologic repository, this would not necessarily be an issue. But as noted earlier, if the glass is not an acceptable waste form for geologic disposal, some form of further plutonium processing would be necessary at some unknown time in the future, involving additional costs and risks.

integrity: helium from alpha decay can, at sufficient concentrations, cause internal pressure buildup, cracking, physical expansion, etc. Alpha-loading experiments to date (Weber 1991) seem to be limited to alpha-loadings corresponding to plutonium loadings only in the 0.1-percent range—an order of magnitude or more below the range of interest for WPu disposition. Even if alpha-loading cracking occurs for plutonium loadings of interest, it is not known whether this issue will matter if the glass is destined for a deep geological repository like Yucca Mountain. Plutonium migration from the glass to surrounding groundwater is likely to be solubility-limited, in which case the additional surface area provided by severely cracked glass may not significantly affect the total plutonium leaching rate into the water (see discussion below). Migration of other radioactive species might be affected, however. Some dose-rate effects of the alpha-loading may also exist and would require exploration.

If it were ultimately determined that substantial modifications to the glass composition were required for a WPu-HLW glass to be an acceptable waste form for geologic disposal—which the panel does not believe is likely—modifying the vitrification process could require unknown extra resources and time. Even with approximately the current glass composition, the addition of substantial quantities of plutonium (and possibly new neutron absorbers) will inevitably require laboratory tests and engineering analyses to validate the glass's repository performance.

Applicability to Other Forms of Plutonium

Both the United States and Russia face severe problems with plutonium in forms such as scrap and residues that are not suitable for either long-term storage or transport to appropriate processing sites. Many tons of plutonium exist in these forms. In addition to handling excess pits from dismantled weapons, vitrification may have an important role to play in immobilizing these unstable forms of plutonium for disposal. Small melters that could be set up onsite to vitrify these scraps and residues—and thereby both stabilize them to reduce the hazards of near-term storage and prepare them for ultimate disposal—deserve consideration. Once vitrified onsite, this plutonium would be in a form sufficiently stable to ship elsewhere for revitrification with fission products if desired.

ASSESSMENT BY KEY CRITERIA

Facility Options and Schedule

As noted elsewhere in this report, the schedule for WPu disposition—including both when a campaign could begin and when it could be finished—is a major component of the security criterion for comparing options. In the case

of WPu vitrification, the critical-path pacing items are likely to be (1) the design and its confirmatory testing, (2) the regulatory and political approval for the pretreatment and melter steps, (3) obtaining government financial support from Congress, and (4) construction. Although obtaining regulatory and political approvals could delay everything significantly, the major outstanding technical issues should be susceptible to resolution within a few years.

The likely schedule for vitrification cannot be assessed in more detail without simultaneously considering what facilities would be employed to carry out the mission. There are two broad options in this regard: WPu vitrification could rely on modification of vitrification facilities already in existence or planned, or new facilities could be built specifically for WPu vitrification. In the latter case, the facilities might still take advantage of existing infrastructure, for example by incorporating the melter within an existing remote-processing facility (such as the reprocessing canyons or vitrification buildings at Savannah River or Hanford). In both the modification and new construction cases, a complex approval and licensing process (including analyses under the National Environmental Policy Act) will be required that will involve delays and uncertainties of unquantifiable magnitude.

Two vitrification facilities exist in the United States today, both operated by DOE, one at Savannah River, South Carolina, and one at West Valley, New York. A third large DOE facility is planned at Hanford, Washington, but has been put on hold pending a reevaluation of previous plans. We discuss below each of these facilities, as well as foreign vitrification facilities, and consider how the various choices would affect the schedule for WPu vitrification.

Savannah River Facility

Vitrification technology is highly developed at DOE's Savannah River Site, and an advanced facility, the Defense Waste Processing Facility, has been under construction for several years there with the goal of vitrifying much of the high-level radioactive waste in the 30 million gallons currently stored there in tanks (McKibben et al. 1993). The plan includes a complex pretreatment process for extracting the bulk of the long-lived radionuclides from this waste for vitrification. The HLW logs from the DWPF will be stored onsite pending ultimate disposal at Yucca Mountain or another DOE HLW disposal site. The remaining LLW waste from the pretreatment process is to be disposed of in near-surface LLW disposal facilities onsite at SRS.

SRS has a full suite of special facilities capable (in some cases with significant modifications) of carrying out the many other steps in an overall WPu vitrification process, including facilities for storing WPu pits, carrying out the various pretreatment steps, and downstream storage and handling of the glass-product canisters.

In the DWPF process as currently planned, the pretreated intensely radioactive waste is fed to the DWPF melter as a solid-liquid mix. The melter's product is a glass-filled canister containing about 600 liters (about 1,700 kg) of borosilicate glass, and weighing just over 2 tons (about 80 percent glass, the rest being the weight of the stainless steel canister itself), standing about 3 m tall and with a 60-cm diameter.

The melter capacity is about 100 kg/h of glass, so the melter can fill one canister in about 17 hours. The plans call for HLW loading in the range of about 20 percent by weight in the glass. Approximately 6,100 glass logs are scheduled to be produced (McKibben et al. 1993). At 17 hours/canister, the entire campaign would take about 15 years if it operated nearly 24 hours a day with no problems or interruptions. However, a longer campaign over two to three decades is currently planned.

The existing waste contains small amounts of plutonium, which will amount on average to about 0.25 kg/canister, or about 0.015 percent of the glass by weight. To add 50,000 kg of WPu to one-third of the canisters in the currently planned campaign would require about 23 kg of WPu per canister, which is about 1.3 percent additional plutonium by weight added to the glass. This appears to be within existing technical limits, as described above.

The existing DWPF has suffered years of delays and substantial cost increases, and is still in its final stages of construction and commissioning. As of early 1993, the planned startup date was in 1994 (GAO 1992). At that time, an accidental flooding of the melter led to a prolonged assessment of plans and operations. The current schedule calls for full-scale production with HLW in 1996, with a campaign lasting until 2018. The project has been held up principally due to technical concerns about the upstream preprocessing of the wastes. Current practices at Savannah River, in the panel's view, indicate a lack of institutional commitment to making progress towards operation. Furthermore, the current operational design and training have indicated potentially serious weaknesses (such as the flooding of the melter just mentioned). Thus some skepticism is in order about the program's ability to meet requirements, even in the absence of WPu.

Experts from SRS have quoted a time for startup of the DWPF in a plutonium vitrification mode as about *mid-2005* (McKibben and Wicks 1993). Given that two years have passed since this estimate was made without major activities being undertaken, this estimate would presumably slip to at least 2006. This estimate was made when the planned DWPF startup was still in 1994, but the initial steps are primarily analysis, design, and regulatory ones in which the status of the DWPF would not play a major role. If DWPF was still not operating in the mid-2000s, however, a WPu disposition mission relying on that facility could be delayed.

Based on an assumed 1.3-percent plutonium loading by weight, *the plutonium part of the campaign would take eight years* (through 2013), and would

not require any additional logs to be vitrified—that is, the nominal 50 tons of WPu would go into the logs currently planned for the eight years between 2005 and 2013 without any new logs being required. Of course, if one could add as much as 4 percent WPu by weight to each canister, it would require only about 700 canisters to cope with the nominal 50 tons surplus WPu. At this higher loading, the WPu part of the campaign could be concluded in less than two years from when it began.

These estimates are those of the SRS team, and assume that vitrification would not proceed until the issues of melter and repository criticality, upstream processing, and offgas system modification had been satisfactorily resolved. The panel has not had either the time or the resources to develop independent estimates. The SRS estimates appear reasonable, assuming a high-priority decision to accomplish the mission, and assuming that the DWPF program progresses without major problems. The estimates have not been examined either for innovative ways to accomplish particular steps more quickly or to root out excessive optimism. They must therefore be considered uncertain. As with other disposition options, achieving success in a timely way would require overcoming the institutional barriers noted at the outset of the chapter, which would require high-level DOE commitment.

West Valley Facility

In West Valley, New York, the site of the only commercial reactor-fuel-reprocessing facility that ever operated in the United States, there is a vitrification facility under construction called the West Valley Demonstration Plant (WVDP). It has been developed by DOE with the goal of vitrifying over 2,000 cubic meters of HLW into glass canisters about the same size as those from Savannah River's DWPF (Jain and Barnes 1993). The waste was produced about two decades ago when the fuel-reprocessing plant operated, and is mostly liquid fission-product waste in alkaline solution. The schedule calls for a startup date slightly later than that at Savannah River, and the planned vitrification campaign will produce about 300 canisters in an 18- to 24-month period. The WVDP melter design has about half of the capacity of the DWPF melter and its technology is similar.

It would be possible to vitrify some part of the WPu at West Valley with the fission products located there. These fission products would be adequate for only a small fraction of the plutonium, however. If all of the excess WPu, or a substantial fraction of it, were to be vitrified at West Valley, fission products would have to be imported to that site—such as the encapsulated Cs-137 now at Hanford. In this case, either the upstream plutonium processing steps would have to be accomplished at other sites, with the resulting plutonium form shipped to West Valley, or the upstream plutonium-handling and plutonium-processing facilities that already exist at sites such as Hanford and SRS would

have to be duplicated at West Valley, with the attendant costs. Since this site is less prepared for this option than Savannah River, the overall schedule would presumably be somewhat longer. Moreover, the involved local community, as well as the West Valley operators, believe the site will only be used for vitrification of local waste. The site has been the subject of considerable local concern, and there has been no mention by DOE of the possibility of using the West Valley plant for WPu vitrification.

Hanford Waste Vitrification Project

DOE had planned a major vitrification facility at Hanford, Washington, known as the Hanford Waste Vitrification Project (HWVP), with a startup date in 1998, but the project has been put on hold pending a review of the plans.¹² The plans for this facility would have employed technologies very similar to those in Savannah River's DWPF, and is not discussed further here. Clearly, if this facility is ever built it could also be used for vitrifying WPu, if a decision to do so were taken with enough lead time. In fact, because of the deferral, it *might* be easier and cheaper (although probably not faster) to design a WPu capability into this facility from the start than to adapt the already built DWPF. The schedule for beginning vitrification of WPu at HWVP is indeterminate, given that it is as yet unknown when HWVP will begin operations. If HWVP could be designed from the outset for WPu, and began operations in the mid-2000s time frame envisioned for the start of WPu vitrification at Savannah River, waiting for HWVP might impose only a modest delay on initiating WPu vitrification operations.

Foreign Vitrification Facilities

Significant vitrification capability exists abroad (Odell 1992), the most impressive being that in France. For decades, the French have carried out research into various glass technologies, and at Marcoule they have operated a sophisticated facility since 1978. A second and more advanced French facility at Cap de la Hague started operations in 1989. In Belgium, a facility at Mol that is jointly supported by the German government operated from 1985–1991, and is now in the process of upgrading. In 1987, a large facility began operations with phosphate glass at Chelyabinsk in the Soviet Union (now Russia), and a British facility with technology similar to the advanced French approach started up at Windscale in 1990. A facility in Tarapur, India, has also operated for some years.

¹² USDOE (1993). The November 1993 revision contains the deferment decision for the Hanford vitrification facility and its rationale.

These are the only facilities worldwide that have significant vitrification capacity, although several research-scale operations exist in various institutions, and plans for large facilities are now developing in a few other countries, most notably in Japan and China.

These foreign processes differ in their technical approaches: for example, some use calcining in the pretreatment stage while others do not. The melter in Japan operates on porous-glass preforms saturated with fission-product solution rather than on glass frit. The waste streams feeding into the melters differ as well, some vitrifying fuel-reprocessing liquid wastes, some military-processing wastes, and some solid wastes. Much of the glass product from these foreign facilities is intended for permanent (or at least very long-term) onsite storage, usually with either forced- or natural-convection air cooling.

Taken all together, the various plants have a very large capability—several times greater than that of Savannah River's DWPF—that is diverse in both technology and operating philosophy (Odell 1992). Almost all of this foreign capability would be technically suitable or adaptable for vitrifying WPu in one form or another, if institutional barriers could be overcome. The schedule to do so would be dependent both on technical issues similar to those described above for U.S. facilities, and the institutional issues involved in shipping WPu overseas—and would also require convincing major reprocessors whose livelihoods depend on separating plutonium from fission products that the reverse operation should be performed on excess WPu.

As disposition of Russian WPu is a major issue, the Russian vitrification operation is of particular interest. A waste vitrification facility with a nominal output of 1 ton of glass per day is in operation at the Chelyabinsk-65 site in Russia and, by September 1993, was reported to have processed 150 million curies of radioactive waste, at a loading of between 150,000 and 200,000 curies per ton. The glass produced has somewhat higher loadings of radioactivity than are planned at Savannah River. Nearly 700 million curies of HLW remain in waste tanks at this site, similar to the holdings at Savannah River and somewhat more than the amount at Hanford.¹³ As noted earlier, the phosphate-glass composition employed at this facility is less appropriate for WPu disposition than borosilicate glass. Alternate melters could be used at this facility, however, to produce borosilicate glass if a decision were taken to do so.

Some of the small melters developed in the U.S. vitrification program, in particular, are relatively low in cost and transportable, and could therefore be shipped to Russia for a vitrification campaign there, if modification of existing Russian melters proved too costly. Russia has operational remote-handling facilities that could be used to operate such melters while incorporating HLW or cesium capsules in the product to create a radioactive barrier. Such small

¹³ Bradley (1993). See also Bradley (1992). For figures on wastes in the U.S. complex, see, for example, OTA (1991).

melting could be used to produce either small glass logs (which would pose a somewhat lower barrier to theft) or large glass logs like those produced in larger melters. The net cost of this approach depends on whether it is seen as an alternate way of handling the HLW vitrification campaigns already planned (in which case much of the cost might be offset by reductions in other vitrification costs) or as a separate campaign for disposing of WPu. Assuming the former case, the basic contributors to cost would be similar to those in the United States, though in Russia at present, both capital and labor costs are substantially lower than in the United States.

In general, Russian authorities have objected to WPu disposition options that would “throw away” the plutonium without generating electricity. Given the environmental legacy of past handling of plutonium and the widespread public distrust of government safety assurances, moreover, gaining public acceptance and licenses for a plan to bury plutonium in a repository in Russia might be difficult. The Russian Ministry of Atomic Energy (MINATOM) itself has recently emphasized the environmental dangers of burying long-lived actinides such as plutonium, as part of its advocacy of a closed fuel cycle in which plutonium would be reprocessed and reused. As in the case of spent fuel, however, the ease of storing and safeguarding the vitrified logs would make it possible for Russia to defer decisions on committing them to geologic disposal for a substantial period.

Specially Constructed WPu Vitrification Facilities

It is also possible to construct a new, dedicated facility for WPu vitrification, rather than using existing facilities. Given the complexities of systems capable of handling both fission products and plutonium, a decision to construct a new facility would be expected to mean a considerable stretch-out of the schedule. It is probable that even with a major commitment, the design and approval phase could take 4–6 years longer than using the DWPF. This approach would allow an optimization of the facility for this application, however, and would impose less disruption on ongoing HLW disposal programs.

Although construction of a special-purpose facility might simplify the task of vitrifying the plutonium, the total costs would be higher, because all the costs of production, handling, and disposal of this waste form (including the potentially substantial costs of providing and operating facilities capable of handling the highly radioactive materials that might be added to it) would have to be charged to the plutonium disposition mission, rather than only the net additional costs of adding plutonium to a previously planned HLW vitrification campaign.

Careful study is required, however, of how much the costs and delays of this approach would exceed those of modifying an existing or planned facility. The modifications to existing facilities needed for WPu vitrification may be substantial. In the case of the DWPF, for example, necessary modifications

would probably include a new melter; a modified offgas system; modified, critically safe systems for feeding plutonium into the melter; and installation of a complete safeguards and security system (there is currently no safeguards system for the Savannah River waste operations, since they are not handling material of proliferation concern).

Costs and delays involved in building a new facility could be reduced by making use of existing remote-handling facilities. Given the physical arrangements at DWPF, for example, it seems quite possible that a small additional melter could be built within the same building; alternatively, this could be done at the building that will house the HWVP, or in the reprocessing canyons at SRS or Hanford. Such a separate melter could vitrify plutonium with HLW, or the Cs-137 stored at Hanford could be used to provide a comparable radiation barrier. If an independent melting operation was going to be undertaken in any case, using this Cs-137 might be simpler in important respects, given the chemical complexities of the wastes at both Hanford and Savannah River.

It should be noted, however, that producing additional logs beyond those already planned would involve additional costs. All of the planned capacity in the Yucca Mountain repository will be filled by wastes already scheduled to be produced. Therefore production of additional waste products specifically for WPu disposition (rather than piggy-backing on planned HLW vitrification campaigns) would require either displacing other wastes now scheduled to go into Yucca Mountain, expanding that repository's capacity, or waiting for an indeterminate time until a second repository became available. A significant cost is associated with the disposal of each additional log that would be produced. (The same is true for spent fuel, if the reactor used for plutonium disposition would not otherwise have operated and produced this waste.)

In a new facility, the time for the full campaign of vitrifying 50 tons of WPu would depend on choices as to the capacity of the new facility. Such a facility would presumably be designed to accomplish the campaign expeditiously; even a 1- to 2-year campaign time could be fully feasible if desired.

Vitrification Without Fission Products

If WPu were to be vitrified *without* fission products—as an interim step before later “revitrification” as described above—a remote-handling facility would not be necessary, and the schedule for this initial step might be significantly compressed. The Savannah River experts estimated that small glove-box melters could begin production-scale operations with WPu glass in roughly nine years (most of that time being involved in approvals, installation of equipment, and preparing for large-scale conversion of various forms of plutonium to other forms). In particular, the small-stirred melter technology might be particularly attractive for this approach (McKibben et al. 1993). One such advanced melter is now onsite at SRS. This new design is smaller and less expensive, has a

throughput of about one-fourth to one-third of the larger unstirred DWPF melter capacity, and could be deployed relatively quickly, especially for use in vitrifying WPu alone without spiking. Also, its smaller size means that constructing additional similar melters could be feasible for accelerating the WPu vitrification throughput, either at SRS or elsewhere.

Without fission products, a remote-handling facility would not be required. This would eliminate a number of time-consuming and expensive steps in the process, such as designing the optimal scheme for the pretreatment of the mix of WPu with the other (highly radioactive) contaminants, and would simplify safety analyses.

The panel believes that for the vast majority of the WPu that is in the form of pits, there is little to be gained in terms of security from vitrification without fission products. A small fraction (but still a significant absolute amount) of the WPu in both the United States and Russia exists in other forms, however, such as partially machined pits, metal scraps, and so on. For this material, there would likely be an increase in both its safety and security if it were vitrified even without fission products, because it would all be converted to a physical form that can be easily accounted for and safely stored. Whether these advantages outweigh the financial costs, the administrative burdens, and the ES&H impacts of such an operation is both a technical and a policy question that the panel recommends should be seriously examined. Vitrifying the WPu pits themselves without fission products does not seem to the panel to be worthwhile; however, neither the security nor the safety benefits seem to be substantial enough to justify the costs and risks, except perhaps as a step in the process of vitrifying the material in a form incorporating fission products.

Late in its deliberations, the panel learned that options are being considered to vitrify actinides now stored in solution at the F-canyon at Savannah River, as part of the clean-out of that facility. Preliminary estimates suggest that the WPu stored in solution there (as well as solutions of some other actinides) could be vitrified in a glass comparable to those discussed here for only modest additional cost compared to the investment that must be made to clean out this facility in any case. This would offer the opportunity for an early production-scale demonstration of WPu vitrification. Such a demonstration would be very valuable, as the lack of such a clear technology demonstration is currently one of the weaknesses of the vitrification approach when compared to the MOX options. The panel therefore believes that the possible synergy of combining clean-out of the F-canyon solutions with an early demonstration of WPu vitrification should be seriously examined, and, if the approach is found to be technically and economically viable, seriously evaluated in the context of the overall WPu effort and, if appropriate, implemented. The panel also believes that the feasibility of incorporating fission products in this demonstration should also be explored in order to achieve a full demonstration of the entire approach described in this chapter at an early date.

Approvals and Licenses

As noted above, gaining regulatory approval for the various plutonium processes required for vitrification would be a major and uncertain component of the schedule for carrying out vitrification operations. (This component is included in the estimates quoted above, with the assumption that a high national priority was assigned to accomplishing the WPu disposition mission.) Certifying the safety of the additional processes needed to add plutonium to currently scheduled HLW vitrification campaigns would take several years. Careful attention would have to be paid to melter design to ensure against criticality and to the offgas system that must prevent release of plutonium into the environment and accumulation of plutonium within the offgas system itself. These engineering issues, while challenging, appear resolvable. Gaining public acceptance at the relevant sites may be more difficult, but if (1) the public is included in the decision-making process, (2) the association with arms reductions is made clear, and (3) a plausible case can be made that once processed, the plutonium will eventually be shipped elsewhere for burial in a geologic repository, then public approval should be achievable. Overall, licensing and approval for this approach would probably be easier than for MOX, at least in the United States. Siting approval and licensing for a vitrification facility dedicated solely to plutonium disposition would probably be more protracted than for an approach piggy-backing on already scheduled HLW vitrification campaigns.

Certification of the plutonium-bearing glass as a suitable waste form for emplacement in a geological repository, including resolution of the long-term criticality issue, would be the highest hurdle.

Safeguards, Security, and Recoverability

As noted earlier, the difficulty of extracting plutonium from the glass logs would be generally comparable to the difficulty of extracting plutonium from spent fuel, given the intensity of the radiation fields with which anyone handling the logs would have to cope.

As for the opportunities for diversion or theft of the materials, it is important that all necessary plutonium operations for the vitrification option—both pit processing and production of the plutonium-bearing glass—could be carried out at a single nuclear weapons complex site with extensive safeguards and security. Thus the number of required transportation and storage steps, and the associated opportunities for theft, would be less than in those reactor options requiring more than one site.

Fabrication of HLW logs would also be easier to safeguard than fabrication of MOX fuel bundles (Shea 1993). Monitors would have to confirm only the single step of mixing the plutonium with the HLW. Once that step had taken place, the plutonium would be in an intensely radioactive mix and very difficult to divert. There would be no capability within the vitrification facility for re-

separating the plutonium from the HLW. MOX fabrication, by contrast, requires many steps involving large-scale bulk handling of plutonium with inherent accounting uncertainties, and at each step of the process the plutonium remains in a form from which it could be readily re-separated.

For the glass operation, however, once the plutonium had been mixed with the HLW and incorporated in glass, the very high radioactivity and strong neutron absorption of the glass log would make accurate nondestructive assays of the amount of plutonium in the glass difficult. Thus, the traditional material-accounting approach of detailed measurement of the inputs and outputs of the plant might have to be modified, with safeguards relying more on confirming that the plutonium was mixed with HLW, and on containment, surveillance, and security measures to ensure that no plutonium was removed from the processing area or from the site without authorization. Although this would be an engineering challenge, adequate technologies exist to safeguard the glass production process, particularly given the relative simplicity of safeguarding the glass production process as compared with safeguarding the MOX fabrication process.

As vitrification operations do not normally include fissile materials, the types of security required for handling such materials have not to date been provided for facilities such as the DWPF. Setting up the requisite security system and procedures would be one of the significant modifications required for the DWPF if WPu were to be vitrified there. Once the logs had been produced, they could be stored and safeguarded relatively cheaply until repositories were ready to accept them, in facilities already planned, just as in the case of spent fuel.

Indirect Impact on Civilian Fuel-Cycle Risks

Treating pure weapons-grade plutonium as a waste to be disposed of would support the present U.S. administration's policy of generally discouraging the use of separated plutonium reactor fuels.

Cost

The schedule and cost for achieving large-scale WPu vitrification depends on a wide range of factors that are not yet known in detail, including such fundamental matters as the facility to be used, the number of logs to be produced, and the like. Only the roughest estimates are available at present. For those options that would incorporate WPu in HLW glass that would be produced in any case, it is important to focus on the *net additional* cost of adding the WPu, as in those cases the total cost of the vitrification operation cannot be charged to the WPu disposition mission. For all cases, it is important to separate the various preprocessing costs before vitrification begins from the costs of vitrification itself, as these preprocessing costs are similar to those that must be borne by other options as well.

Vitrification costs will depend on several issues, including: (1) whether existing facilities can be used in whole or in part, including taking advantage of existing facilities even if a new vitrification facility were to be built; (2) whether the duration of the WPu vitrification campaign can efficiently use the chosen facilities; and (3) whether the use of these facilities displaces other necessary or desirable activities. Also, the WPu vitrification cost will depend almost inversely proportionally on how much WPu loading by weight can be safely and economically added to the vitrified glass, unless the WPu campaign fits well into a previously planned campaign to vitrify HLW, such as the currently planned DWPF campaign to vitrify HLW at SRS.

The only detailed cost estimates that have been available to the panel were prepared by Westinghouse Savannah River Company, for vitrification at Savannah River Site.¹⁴ The estimated cost for vitrification with HLW in the DWPF is approximately \$600 million, plus approximately \$400 million to carry out the preliminary steps, including pit processing (which would also be required for the reactor options). The SRS team estimates the cost of vitrification without HLW at less than \$200 million (plus the same \$400 million preprocessing costs). These estimates are uncertain by at least a factor of two. As noted above, the cost of a separate plutonium vitrification campaign that incorporated radioactive materials such as Cs-137 would be higher, because the high costs of processing highly radioactive glass would then have to be borne entirely by the WPu disposition mission, rather than being shared by HLW disposal operations already planned.

These rough cost estimates are based on carrying out the vitrification at SRS. Certain economies could be realized by designing for WPu vitrification from the beginning in the currently deferred HWVP, but it is not possible now to estimate these very well.

ES&H

Extensive engineering effort has been necessary to assure that the DWPF at Savannah River, the most advanced vitrification facility that now exists in the United States, can meet all applicable environmental, safety, and health regulations. Although it will be a challenge to provide a comparable level of assurance for a facility to vitrify WPu (whether a modified DWPF or another facility), achieving adequate compliance should be within current technological capabilities. Neither cost nor schedule difficulties should be affected overwhelmingly by problems in these areas.

¹⁴ McKibben et al. (1993). This is an undiscounted estimate; discounting by 7 percent per year (see Chapter 3) would reduce the billion-dollar figure by roughly half, for comparison to other options. These estimates also include previtrification in a plutonium-only glass; eliminating this step would probably lower costs somewhat.

One benefit of the vitrification process is that it can accept either WPu or other plutonium forms, and can immobilize essentially all of the plutonium feedstock, so that the need for handling or disposing of subsequent plutonium-contaminated radioactive waste beyond the WPu-laden glass itself can be substantially reduced. (Waste from production is recycled into the melter to produce new glass.) Of course, some plutonium-contaminated waste streams, including contaminated equipment, will require subsequent LLW handling and disposal. These waste streams (besides the glass product itself) should be manageable within applicable regulations. These ES&H issues are addressed in more detail in Chapter 6.

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Comparing the Options

In this chapter we compare the reactor-related options for disposition of weapons plutonium (WPu) using the criteria developed in Chapter 3. The array of options treated comprises: currently operating light-water reactors; currently operating heavy-water reactors; currently operating liquid-metal reactors; evolutionary light-water reactors; advanced light-water reactors; advanced liquid-metal reactors; modular high-temperature gas-cooled reactors; molten-salt reactors; particle-bed reactors; accelerator-based conversion systems; and immobilization with fission-product wastes. The emphasis in these comparisons is on the options that we and the parent committee have concluded offer the greatest promise of reducing the security hazards associated with surplus WPu over the next 30 years or so—the use of currently operating reactor types or evolutionary adaptations of them to incorporate WPu into spent fuel on a once-through basis, and vitrification of WPu with defense high-level wastes (HLW).

We begin by comparing the options on criteria related to security, turning then to economics and environment, safety, and health (ES&H). Many of the reactor-option characteristics that influence these evaluations depend to some degree on details of core design, fuel composition, and refueling modes and schedules that may vary considerably within a given reactor type. For example, given a pressurized-water reactor (PWR) with a core loaded two-thirds with low-enriched uranium (LEU) and one-third with mixed-oxide (MOX) fuel, the quantity and quality of plutonium in the spent fuel will depend in detail on the initial percentages of uranium-235 (U-235) and the various plutonium isotopes in the fresh fuel, on the burnup, and on the neutron-energy spectrum, which can

vary with core and fuel design, presence of burnable poisons, and so on (see Chapter 2).

The calculations used to determine spent fuel characteristics subject to these variations are fairly complicated and are performed with large, standardized computer codes and neutronics databases. We have not performed such calculations ourselves for this study, but have relied instead on the calculations performed for similar purposes over the past few years by reactor manufacturers and national laboratories. The quantitative comparisons in this chapter are based largely on the set of such calculations summarized in Table 6-1, which includes a range of reactor types—and, within types, ranges of fuel characteristics and operating modes—sufficient to illustrate the key dependencies and variations.¹

The bulk of the comparison of options in this chapter focuses on options for disposition of U.S. WPu. This was inevitable, given the amount of relevant information available in the United States. Given the importance of disposition of plutonium in Russia as well, however, several paragraphs at the end of each section in this chapter are devoted to a preliminary discussion of how the options in Russia compare by the same criteria. The panel believes that additional study rigorously comparing plutonium disposition options in Russia against the criteria outlined in this chapter is urgently needed.

SECURITY COMPARISONS

As noted in Chapter 3, the primary motivation of the U.S. government in preparing to carry out disposition of excess WPu is to minimize the risks to national and international security posed by the existence of this material. Thus, options must meet this objective to be worthy of further consideration.

The panel was not charged with examining many of the issues related to security that are described in the report of our parent committee (NAS 1994), such as the interaction of WPu disposition with the future efforts to reduce nuclear arms and stem their spread. The panel takes note, however, of a number of important considerations outlined in the committee's 1994 report:

¹ The presentation of numbers in this table to three- or four-digit precision should not be taken either as indicative of the actual accuracy of the calculations, which is generally less, or as suggesting that small differences in these values are important, which generally they are not; the largely illusory precision in Table 6-1 is maintained simply to facilitate consistency checks and to permit associating the values in the table with particular calculations in the literature. Nor should it be assumed that the presence in the table of reactors designed by particular manufacturers constitutes a preference by the panel for these manufacturers' designs in comparison to other manufacturers' designs of the same general type; unless otherwise noted, the purpose of this specificity is simply to associate the tabulated parameters with the particular design, fuel type, and operating mode for which they were calculated.

TABLE 6-1 Plutonium Disposition Characteristics of Some Representative Reactor Types

Description	Burnup (MWd/ kgHM)	Metal Inventory (MT)	Metal Input (MT/yr)	Pu Input (kg/yr)	Pu Output (kg/yr)	Δ Pu (kg/ GWe-yr)	Pu/HM in Spent Fuel (%)
CLWR: 3,800-MWt/1,200-MWe PWR @ CF=0.75							
100% LEU (3.8%)	42.0	99.2	24.8	0	253	+281	1.0
33% MOX (4.0%)	42.0	99.2	24.8	330	387	+63	1.6
100% MOX (4.0%)	42.0	99.2	24.8	996	636	-400	2.6
ELWR: 3,817-MWt/1,256-MWe PWR @ CF=0.75							
100% MOX (6.8%)	42.2	99.2	24.8	1,672	1,215	-485	4.9
ELWR: 3,926-MWt/1,300-MWe ABWR @ CF=0.75							
100% LEU (2.6%)	27.5	155.2	39.2	0	338	+347	0.9
100% MOX (3.0%)	37.1	155.2	29.0	867	562	-313	1.9
ALWR: 3,880-MWt/1,220 MWe APWRs (2 \times 1,940-MWt/610-MWe) @ CF=0.75							
100% MOX (5.5%)	40.0	133.8	26.6	1,462	1,038	-463	3.9
100% MOX (4.0%)	50.0	98.0	21.2	850	421	-469	2.0
CHWR: 5,664-MWt/1,538-MWe CANDUs (2 \times 2,832-MWt/769-MWe) @ CF=0.80							
100% natural U	8.3	234.7	199.4	0	768	+624	0.4
100% MOX (1.2%)	9.7	232.5	170.6	2,124	1,405	-584	0.8
100% MOX (2.1%)	17.1	226.0	96.8	2,000	1,333	-542	1.4
MHTGR: 3,600-MWt/1,717-MWe MHTGRs (6 \times 600-MWt/286-MWe) @ CF=0.75							
100% PuO/PuO ₂	595.5	4.7	1.7	1,656	589	-829	36
CLMR: 1,470-MWt/560-MWe BN-600 LMFBR @ CF=0.75							
100% MOX (15.6%)	60.9		6.6	1,032	1,070	+90	16.2
ALMR: 4,200-MWt/1,440-MWe ALMR @ CF=0.75							
100% U/Pu (10.6%)	75.2	113.0	15.3	1,626	1,781	+144	11.6

ABBREVIATIONS:

- ABWR = advanced boiling-water reactor.
- ALMR = advanced liquid-metal reactor.
- ALWR = advanced light-water reactor.
- APWR = advanced pressurized-water reactor.
- CANDU = Canadian deuterium-uranium (reactor).
- CF = capacity factor.
- CHWR = current heavy-water reactor.
- CLMR = current liquid-metal reactor.
- CLWR = current light-water reactor.
- ELWR = evolutionary light-water reactor.
- GWe = gigawatt-electric.
- HM = heavy metal.
- kg = kilogram.
- LEU = low-enriched uranium (figure in parentheses is U-235 enrichment).
- LMFBR - liquid-metal fast-breeder reactor.
- MHTGR = modular high-temperature gas reactor.
- MOX = mixed-oxide fuel (figure in parentheses is Pu percentage of heavy metal).
- MT = metric ton.
- MWd = megawatt-day.
- MWe = megawatt-electric.
- MWt = megawatt-thermal.
- PWR = pressurized-water reactor.

NOTES to Table 6-1:

CLWR: These cases calculated for a nominal current PWR by Battelle Pacific Northwest Laboratories (Pritchard 1993); LEU case shows current practice for comparison.

ELWR: First case is ABB-Combustion Engineering System-80+ evolutionary PWR at 3,817 MWt (ABB-CE 1993); this case shows a plutonium loading higher than practical in current PWRs, making it possible for a single large reactor to load 50 tons of WPu in a 30-year operating lifetime. Second case is General Electric's ABWR, actually an evolutionary reactor in our terminology (GE 1993). First row parameters for use of LEU without MOX in this reactor scaled from American Physical Society study values for a nominal boiling-water reactor (APS 1978).

ALWR: Two Westinghouse PDR-600 APWRs (Westinghouse 1993). The second row illustrates a case using an advanced annular fuel design to achieve very high burnup and a 50-percent plutonium destruction fraction starting from 4-percent WPu MOX fuel; this destruction fraction is considerably higher than those attained with more typical LWR parameters.

CHWR: Two CANDU reactors of the Bruce Station type, as analyzed for conventional natural uranium fuel and "reference" and "advanced" MOX fuels by Atomic Energy of Canada, Ltd. (AECL 1994). CANDUs are given an advantage in capacity factor—an assumed 0.80 in contrast to the 0.75 assumed for the other reactor types listed—because unlike the others they do not need to shut down to refuel.

MHTGR: Six General Atomics 600-MWt gas-turbine MHTGR modules, as analyzed for 100-percent PuO/PuO₂ fuel by General Atomics (GA 1994). Plutonium weight fraction in spent fuel in this case refers to weight fraction of plutonium in plutonium plus higher actinides plus fission products.

CLMR: The CLMR is the BN-600 LMFBR operating in Russia, the largest of the world's three commercially operating LMRs. Its parameters using 100 percent MOX were calculated at Battelle Pacific Northwest Laboratories (Pritchard 1993); whether 100 percent MOX loading in this reactor is actually feasible is in doubt, however (see Chapter 4).

ALMR: Five GE 840-MWt ALMR modules, as analyzed for metallic plutonium-uranium-zirconium (Pu-U-Zr) fuel under the "spent fuel alternative" by General Electric (GE 1993). Figures given here for burnup (75.2 MWd/kgHM) and initial plutonium content in metallic fuel (10.6 percent) relate to heavy-metal content only (not including the zirconium in the ternary Pu-U-Zr fuel) and are averages for driver and blanket fuel assemblies. (Initial driver fuel plutonium content is 20.3 percent WPu in heavy metal.)

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- Current arms reductions agreements do not require the dismantlement of the nuclear weapons involved; nor do these agreements place any controls on the fissile materials these weapons contain. Dismantlement of these weapons and disposition of the resulting fissile material would significantly contribute to building confidence in the "irreversibility" of nuclear arms reductions, a goal enunciated by President Clinton and Russian President Yeltsin at their summit in January 1994. For disposition of WPu to contribute to this goal in the near term would require a clearly enunciated plan for disposition, with an expeditious beginning of implementation.
 - The foundation of international nuclear nonproliferation efforts is the Non-Proliferation Treaty. This treaty is based on a bargain between the nuclear-weapon states and the non-nuclear-weapon states, which included the nuclear-weapon states' commitment to negotiate in good faith toward nuclear disarmament. A clear commitment to disposition of excess WPu, with an early start on implementation, would contribute

to confidence that the United States and Russia were making good on this commitment.

- The current transformations in the former Soviet Union create nuclear risks of three general kinds:
 - “breakup,” meaning the emergence of multiple nuclear-armed states where previously there was only one;
 - “breakdown,” meaning erosion of government control over nuclear weapons and materials within a particular state; and
 - “breakout,” meaning repudiation of arms-reduction agreements and pledges, and reconstruction of a larger nuclear arsenal.

It would be desirable not to prolong these risks as they apply to excess WPu any longer than necessary.

The panel notes that all of these considerations point to:

- (1) timing as an absolutely critical part of minimizing security risks; and
- (2) the importance of disposition of excess WPu not only in the United States, but in Russia as well.

These points are emphasized throughout the remainder of this section, in comparing the security impacts of different reactor options. We begin our comparisons of these security impacts with some broad generalizations, related to the various issues mentioned above and the potential threats considered in Chapter 3, that have been important to our process of narrowing the range of options given serious consideration. We then focus on the timing of different options, followed by discussions of the accessibility of the excess WPu during the course of the various disposition options and the degree of difficulty of recovering the excess WPu when they are complete.

General Considerations

Some general conclusions about the security dimensions of alternative approaches to disposition of WPu follow directly from consideration of the character of the threats likely to be of greatest concern (see Table 3-1). In particular:

- (A) *Risks of Storage.* Prolonged storage of excess WPu in readily weapons-usable forms would mean a continuing risk of breakout, as well as of theft from the storage site. In addition, extended storage of large quantities of excess fissile materials, particularly in the form of weapon components, could undermine the arms reduction and nonproliferation regimes (the severity of this problem depending in part on the specific arrangements for custody of the materials in question). Thus, in judging the attractiveness of disposition options, we give heavy weight to (1) minimizing the time before processing of WPu can begin and (2) minimizing the subsequent time lag before disposition has succeeded in

reducing the accessibility of the last excess WPu (e.g., when the last WPu has been loaded into a reactor or a vitrifier). The timing for each disposition option is dependent on three factors: its technical readiness or uncertainty, the speed with which public and institutional approval (including relevant funding) could be gained, and the time required to implement it once developed and approved.

- (B) *Risks of Handling.* Nearly all disposition options other than indefinite storage as pits require processing and usually transportation of plutonium, in ways that could increase access to the material and complicate accounting for it, thus increasing the potential for diversion and theft. The biggest risks of these kinds involve the steps before the WPu has been either irradiated in a reactor or mixed with radioactive wastes. In order to ensure that the overall process reduces net security risks, an agreed and stringent standard of security and accounting must be maintained throughout the disposition process, approximating as closely as practicable the security and accounting applied to intact nuclear weapons. The parent committee called this the “stored weapons standard.” Hence, choices among long-term disposition options comparable in terms of timing should be weighted in favor of those that minimize:

- any processing steps with high accessibility and low accountability;
- the number of transport steps and the risks involved in each; and
- the number of sites at which plutonium is handled and the risks at each site.

- (C) *Risks of Recovery.* A third key security criterion for judging disposition options is the risk of recovery of the plutonium after disposition—by the state from whose weapons the WPu came (either covertly or overtly), or by potential proliferators (acquiring the material by covert theft, overt theft in the event of a loss of national authority, or overt forcible theft). Options that left the excess WPu substantially more accessible for weapons use than the global stock of plutonium in civilian spent fuel would mean that this material would continue to pose a unique safeguards problem indefinitely. Conversely, as long as the large stocks of plutonium in civilian spent fuel exist and continue to accumulate, options that made the excess WPu much less accessible than these larger stocks (for example, by eliminating it entirely or nearly so) would provide little additional security benefit, unless the same were done with the much larger stock of civilian plutonium. These considerations lead naturally to the “spent fuel standard” enunciated earlier. In any case, none of the disposition options that could plausibly be completed in less than 50 years would destroy more than

70–80 percent of the excess WPu—and whether the amount of WPu that remains in spent fuel (or vitrified waste) is 10 or 50 tons makes little difference to the overall security picture when the total stock of plutonium in spent fuel by that time will amount to over 1,500 tons.

- (D) *Indirect Impacts.* The goal of long-term disposition of WPu should be not only to ensure that the plutonium from dismantled weapons is not reused in weapons, but also to avoid substantially increasing security risks from other fissile materials. Thus, policy-makers must be attentive to possible indirect effects that the choice of disposition options might have on the proliferation risks posed by other fissile materials in the world, in addition to its direct effects on the surplus weapon material. Disposition options that entail use of MOX fuel in, and/or fuel reprocessing for, civilian power reactors could potentially encourage the expanded use of these approaches in ways that increase the vulnerability of reactor plutonium (RPu) to diversion for nuclear weaponry (including acquisition of weapons by countries that possess no WPu). Conversely, it is possible that development of MOX fuel or reprocessing approaches for WPu disposition would lead to improvements in the safeguardability of these technologies (or an increase in society's determination to safeguard them), with beneficial results in current or future civilian nuclear-energy programs of which these technologies are a part. Policy-makers examining disposition options will have to take into account these possible indirect impacts of the options on the management of civilian plutonium, and how they fit with broader national fuel-cycle policies. For either reactor or vitrification options, if the United States wishes to maintain a policy of generally discouraging fuel cycles involving the use of separated plutonium, or if it wishes to make support for such cycles contingent on stringent safeguards and security measures, it will need to make a clear statement of how its choice fits within that broader context. It is important to note, in this context, that since the WPu is already separated, choice of a reactor option would not necessarily reopen the contentious question of reprocessing in the United States.

Timing

The issue of timing, which as we have argued above is an important aspect of security, has a number of dimensions: (1) the length of time until a disposition scheme can begin receiving and processing plutonium, (2) the length of time thereafter until the total quantity of surplus plutonium has entered the process, (3) the length of time from the start of operations until all of the plutonium has reached its final dispositioned state, and (4) the lengths of time that the plu-

tonium spends in its various intermediate forms and locations during the disposition campaign (especially, of course, the most vulnerable forms and locations).

The first of these characteristic times—the length of time until disposition operations can commence—is important from the standpoint of sending an early “signal” that the commitment to remove the plutonium from the military inventory is really being carried out, as well as being a key element in the timing of the whole disposition process. The determinants of this critical length of time include: (1) the time needed to make a decision about how to proceed; (2) any time required for research, development, and design of elements of the scheme before they can be constructed at the needed scale; (3) the time needed to obtain the requisite permits and licenses, including time for any analyses required as input to the permitting and licensing processes; and (4) the time needed for construction and startup testing of any of the necessary facilities that do not already exist.

Some of these time periods can and should overlap: research and development on a variety of options can proceed in parallel with a process of decision about which option to select for deployment; construction on some elements of a disposition scheme can be underway while other elements are still under development; and licensing does not necessarily need to be complete before construction begins. (Overlap of construction with licensing can be the cause of costly mid-construction design changes, however, as experience with commercial nuclear reactors has demonstrated from time to time.) Estimation of the time requirements for the various steps that must precede commencement of disposition operations—and of the degree of overlap that can or will occur in these steps—is difficult, and the results will necessarily be approximate.

We present in Table 6-2 our estimates of the minimum plausible time requirements for the steps preceding commencement of disposition operations for a representative array of reactor-related options in the United States. The times specified in Table 6-2 for U.S. options are based on the assumption that (1) a U.S. government decision to proceed with a particular option is made early in 1996, and (2) the WPu disposition mission is given high national priority with corresponding resources—as the panel recommends be done.²

² We do not think that a decision to proceed with any of the advanced-reactor options could or should be made so quickly, but we chose to assume the same decision date for all the options in making these estimates in order to be able to compare, on an equal footing, the time requirements once a decision is made. The estimates given in Table 6-2 are in reasonably close agreement with those developed by the Fission Working Group Review Committee in the 1992–1993 Department of Energy Plutonium Disposition Study (Omberg and Walter 1993, Figure 5.1-8). We and the Fission Working Group Review Committee are more pessimistic about this timing than were the contractor studies commissioned in Phase I of DOE’s Plutonium Disposition Study, where the contracts required (unrealistically in most cases, we think) that proposals be presented for completing the plutonium disposition mission by 2018.

TABLE 6-2 Minimum Plausible Time to Commencement of Disposition Operations

Option	Steps and Time Requirements
Vitrification with defense HLW ^a	<p>Research on remaining process and repository issues: 1996–1998. Fabrication and installation at Savannah River of plutonium metal-to-oxide conversion facilities: 1998–2001. Fabrication, installation, and testing at Savannah River of suitable melters and other process equipment for adding plutonium to vitrification operations: 1999–2004.</p> <p>PLUTONIUM CONVERSION TO OXIDE AND VITRIFICATION OPERATIONS COMMENCE 2005</p>
CLWR one-third MOX, FMEF ^b	<p>Contract for completion of MOX fabrication plant, do completion work, test and license plant: 1996–2000. Choose LWRs to be used and negotiate arrangements, obtain needed permits and licenses: 1996–1999.</p> <p>PLUTONIUM CONVERSION TO OXIDE AT MOX PLANT COMMENCES 2001; MOX FUEL LOADING IN REACTORS COMMENCES 2002</p>
CLWR full-MOX, FMEF ^c	<p>Contract for completion of MOX fabrication plant, do completion work, test and license plant: 1996–2000. Choose LWRs to be used and negotiate arrangements, make any needed modifications, obtain needed permits and licenses: 1996–2001.</p> <p>PLUTONIUM CONVERSION TO OXIDE AT MOX PLANT COMMENCES 2001; MOX FUEL LOADING IN REACTORS COMMENCES 2003</p>
CLWR one-third or full-MOX, new MOX plant ^d	<p>Contract for construction of MOX fabrication plant, site, construct, test, and license plant: 1996–2002. Choose LWRs to be used and negotiate arrangements, obtain needed permits and licenses: 1996–1999/2001.</p> <p>PLUTONIUM CONVERSION TO OXIDE AT MOX PLANT COMMENCES 2003; MOX FUEL LOADING IN REACTORS COMMENCES 2004</p>
ELWR full-MOX, FMEF ^e	<p>Contract for completion of MOX fabrication plant, do completion work, test and license plant: 1996–2000. Select site and contractors for ELWR, construct, test, and license plant: 1996–2004.</p> <p>PLUTONIUM CONVERSION TO OXIDE AT MOX PLANT COMMENCES 2001; MOX FUEL LOADING IN REACTORS COMMENCES 2005</p>
ALWR full-MOX, new MOX plant ^f	<p>Contract for construction of MOX fabrication plant, site, construct, test and license plant: 1996–2002. Complete development and design work on ALWR, select site and contractors, construct, test, and license plant: 1996–2007.</p> <p>PLUTONIUM CONVERSION TO OXIDE AT MOX PLANT COMMENCES 2003; MOX FUEL LOADING IN REACTORS COMMENCES 2008</p>
MHTGR or ALMR ^g	<p>Complete development and design work on reactors and fuel cycles, select site and contractors for co-located fuel fabrication facility and reactors, construct, test, and license: 1996–2012.</p> <p>PLUTONIUM CONVERSION TO OXIDE COMMENCES 2010, MOX FUEL LOADING IN REACTORS COMMENCES 2013</p>
MSR, PBR, or ABC ^h	<p>Extensive development and design work needed for these concepts likely to add 5–10 years to time scales for MHTGR and ALMR.</p> <p>FUEL LOADING IN REACTORS COMMENCES BETWEEN 2018 AND 2023</p>

NOTES to Table 6-2: All estimates assume a U.S. government decision at the beginning of calendar 1996 to proceed with the option. Elements affecting total time are entered in boldface. Estimates are highly uncertain, for reasons discussed in the text.

^a Assumes addition of WPu at 1–2 percent by weight to glass logs incorporating 20 percent by weight defense HLW, as now scheduled to begin production (without WPu) at Savannah River in the late 1990s.

^b Current Light Water Reactor with 1/3 MOX cores; fuel produced in Fuel Materials Examination Facility (FMEF) at Hanford, Washington.

^c Same as previous case, except that it is assumed that CLWRs can be licensed to use full-MOX cores. This reduces the number of reactor sites that need to be chosen, agreed, and permitted, perhaps shortening the time required for these steps, but may increase the time needed for reactor licensing for MOX use and related analysis and testing. That time could increase from four to six years, as indicated, without changing the overall start dates from those applying in the previous case. Options involving completing existing reactors such as the WNP facilities are considered to be included in the CLWR options, as provision of the needed MOX capacity would in all probability still be the rate-limiting step.

^d Same as previous two cases except fuel produced in a new MOX plant, which adds at least two years to this pacing element.

^e Same as second case except uses newly constructed evolutionary LWRs instead of existing ones, with time estimated for final design, siting, construction, testing, and licensing at nine years. Reactor rather than MOX plant is now pacing element; use of new MOX plant rather than FMEF would delay commencement of conversion to oxide by two years but would not affect start of MOX loading in reactors.

^f Same as previous case except uses advanced rather than evolutionary LWRs, assumed to require an additional three years of design and development time. We assume that on the resulting longer overall time scale it will be decided to employ newer MOX fabrication technology in a new plant, rather than updating FMEF. Use of FMEF would permit starting conversion to oxide two years sooner but would not affect start of MOX loading in reactors.

^g Modular High-Temperature Gas-cooled Reactor or Advanced Liquid Metal Reactor. We rate these two reactor types as comparable in the amount of remaining development and design effort needed before they could be brought into operation using plutonium fuel. The indicated operation date of 2013 is based on assuming a serious national commitment to achieving this, and thus is optimistic. The associated fuel fabrication facility assumed to be completable and licensable at least three years sooner than reactor, to permit accumulation of fuel for first full core by the time reactor is otherwise ready for operation.

^h Molten-Salt Reactor, Pebble-Bed Reactor, or Accelerator-Based Converter.

We emphasize that the dates shown in Table 6-2 reflect what we think is possible, which is not necessarily the same thing as what is likely. In recent years the Department of Energy (DOE) has not had great success in carrying large projects of this kind through to completion; success in this case will require intense, high-level commitment and oversight. In the current institutional environment in the United States, moreover, delaying or halting large nuclear projects is substantially easier than carrying them out on schedule.

There is little doubt that the large-scale processing of plutonium required for virtually any disposition option will engender controversy and that any option will face opponents. At each stage of the lengthy political and regulatory process that will be required to implement any of these options, there are likely to be efforts to block or delay the process, through lobbying the relevant legislatures and regulatory bodies and legal actions in the courts. If successful, such

actions could delay implementation of disposition options for years. (Licensing issues could be particularly important in governing how rapidly options can be implemented and are summarized in the appendix to this chapter.) In short, the time estimates in Table 6-2 are optimistic: while it is extremely unlikely that the disposition options considered here could be carried out more rapidly, it is very possible, even likely, that they will ultimately be carried out more slowly.

Timing of plutonium disposition in Russia is even more uncertain than timing in the United States. The most critical limiting factor on timing in Russia is likely to be the availability of the financial resources needed to carry out the job (see discussion of economics below). As in the United States, making use of existing facilities to the extent possible would speed the process.

In the Russian context, if appropriate support and resources were available, the timing of the vitrification option might be more favorable than in the U.S. context, as Russian vitrification of HLW is already ongoing, new melter are regularly replacing older ones (so that a critically safe plutonium disposition melter could be inserted into the process without undue delay), and regulatory obstacles may be less substantial. To incorporate substantial quantities of plutonium safely, however, would probably necessitate switching from phosphate to borosilicate glasses, which could impose significant delays.

In the case of the light-water reactor (LWR) options, Russia, like the United States, does not have a production-scale operational MOX fabrication facility, but has a partially completed one; Russia's facility is farther from completion, but would have a larger capacity than the Fuel Materials Examination Facility (FMEF). Russia does have two pilot-scale MOX fabrication lines, and therefore might be in a position to carry out initial demonstrations of MOX use in LWRs more rapidly than could the United States. A new MOX facility could probably be built from scratch in Russia on a time scale comparable to that in the United States, assuming that the job had adequate resources and governmental priority—but even more than in the United States, that is a very large assumption, given the many other urgent problems the Russian government must address. (Foreign participation, as has been suggested by some European MOX manufacturers, could potentially help keep such an effort on track.)

As indicated in Chapter 4, Russian VVER-1000 reactors are similar to U.S. LWRs in their adaptability to MOX burning. Regulatory issues in Russia are difficult to predict, but could potentially impose smaller delays, because a legal process allowing multiple opportunities for opponents of a project to intervene does not yet exist to the same degree as it does in the United States. In short, the timing arguments as between the two main options discussed in this report—vitrification and use as MOX in LWRs—do not appear to be radically different in the Russian case than they are in the U.S. case.

Russia's one operating liquid-metal reactor (LMR), the 560-megawatt-electric (MWe) BN-600, would require 50 years' operation to process 50 tons of WPu on a once-through basis (see Table 6-1), even assuming that it could be

modified to use 100-percent MOX fuel. (As noted in Chapter 4, Russian Ministry of Atomic Energy (MINATOM) officials have expressed the opinion that such modification is not practical for this reactor.) The reactor has already been operating for about 15 years, moreover, which means that its remaining operating lifetime may be less than half of the 50 years needed to disposition 50 tons of WPu. As noted in Chapter 4, the record of the BN-600 and its smaller predecessor, the BN-350, raises doubts about its reliability if not its safety. In sum, its suitability for the disposition mission is questionable, and under the most optimistic of assumptions the timing with which it could perform the mission would still be unattractive.

The timing implications of choosing more advanced reactor types than current LWRs and current LMRs are similar in Russia to those in the United States: that is, the use of advanced reactors not yet ready for immediate construction would delay accomplishing the mission considerably. As noted in Chapter 4, Russian officials have argued that because Russia has tested plutonium in LMRs and has not yet explored plutonium use in LWRs, the option of completing several large LMRs and using the excess WPu in them could be accomplished more quickly than the LWR option. But LWRs have the considerable advantage of existing already, and the extensive experience of LWR MOX use in other countries would allow an LWR MOX option to be implemented rapidly in Russia, just as in the United States. The panel believes that storing the WPu until new LMRs can be financed and constructed in Russia would needlessly delay achievement of the spent fuel standard in the protection of this material, substantially extending the period of high security risks—direct and indirect—associated with storage of WPu in forms readily usable in weapons. Proposals from some other Russian analysts to save the excess WPu until other types of advanced reactors became available—such as high-temperature gas-cooled reactors (HTGRs) or lead-cooled liquid metal reactors—would create even larger delays.

In the case of use of reactors in other countries for disposition of either U.S. or Russian WPu, the timing might differ in some respects, but it is difficult to say whether the overall result would be a faster process, a slower one, or no change:

- If U.S. plutonium were to be loaded into European reactors already licensed to use one-third MOX cores, this would circumvent the time requirement for licensing U.S. reactors for such fuel; but it would add a time requirement for negotiating the terms of such an arrangement. The pacing element in the use of existing reactors for WPu disposition might still be the licensing and construction or upgrading of MOX fuel fabrication facilities rather than reactor licensing, unless this fabrication took place in existing or planned European plants. Depending on the size of the European MOX fabrication capacity in relation to civilian

MOX fabrication demand in this period, this might require the negotiation of arrangements and terms for the displacement, in European MOX fabrication plants, of the recycled civilian plutonium they would otherwise be using.³ This approach also would require overcoming political, environmental, and safeguards objections to transoceanic international shipment of U.S. WPu. If the MOX fabrication did not take place in the existing or planned European plants, then the timing would be governed by the requirements of fabricating it in the United States—hence would not differ from that shown in Table 6-2 for the one-third MOX and full-MOX current LWR options in this country—or the requirements of siting, constructing, and building new MOX plants in Europe, which even in the absence of any controversies could not occur much more rapidly than the optimistic U.S. schedule assumed in Table 6-2.⁴

- If U.S. WPu were to be loaded into CANDU (Canadian deuterium-uranium) reactors in Canada, the MOX fuel fabrication for these reactors would probably take place in the United States. Thus this pacing element of the timing would not change from the CLWR options shown in Table 6-2, and the overall timing would be the same—assuming that Canadian licensing of MOX use in CANDUs, and the negotiation of the terms under which this would be done, would require about the same amount of time as the corresponding negotiations and licensing activities in the United States.
- Concerning the possibility of loading Russian WPu into European or Canadian reactors, the timing would hinge both on the time needed to reach agreement on terms and arrangements for the transfer of Russian weapon material to other countries and on the time needed to accomplish the MOX fabrication step. The latter, if done in Europe, would be subject to the same constraints and possibilities described above in connection with making MOX from U.S. WPu in Europe; if done in the United States or Russia, it would be subject to the timing constraints on deploying new MOX capacity in those countries.

In summary, the most plausible way in which an international variant might offer timing better than that of the U.S. current light-water-reactor scenarios

³ If MOX fabrication using civilian plutonium had to be postponed, that would mean *either* that the separated civilian plutonium previously scheduled for MOX fabrication would accumulate at reprocessing plants (meaning that accelerated shrinkage of the stock of separated military plutonium would be achieved only at the cost of creating an equivalent addition to the stock of separated civilian plutonium) *or* that scheduled civilian reprocessing operations would have to be deferred (increasing the European stock of—and hence the need for storage capacity for—civilian spent fuel).

⁴ Controversies are more likely to delay construction and licensing of a new MOX plant in the United States than to delay such an effort in Europe, so the chances of actually meeting the schedule indicated in Table 6-2 might well be better in Europe.

depicted in Table 6-2 would be through the use of existing or planned European capacity for MOX fabrication, avoiding the time-consuming completion or new construction of MOX fabrication in the United States or Russia. Whether this approach would really save time would depend on whether the needed international agreements on transfer and custody of the WPu, together with any needed technical and economic arrangements for displacing the civilian plutonium scheduled to be fabricated in these MOX plants, could actually be concluded in less time than it would take to complete new MOX fabrication capacity in the United States or Russia.

Table 6-3 presents working scenarios for the timing of disposition following commencement of operations for those of the options in Table 6-2 that could begin operating before 2010, assuming a high national priority for this mission. (Later-starting options are of less interest, for reasons adduced above, and any scenarios we might devise for the phasing of their introduction would be even more speculative than those we offer for the nearer-term possibilities.) Our scenarios encompass a range of assumptions about the number and scale of disposition facilities, representing a variety of possible choices concerning the trade-offs among cost, start date, and completion date of the campaign. Table 6-4 shows, for these scenarios, the key dates and the post-2000 integrated inventories of WPu in the two most intrinsically vulnerable forms relevant to these disposition schemes: (1) metal in pits and (2) plutonium oxide as powder or MOX, before loading into reactors or HLW-bearing logs.⁵

The main conclusions we draw from these considerations of the interaction of timing with security can be summarized as follows:

- (1) Timing of all the WPu disposition options is highly uncertain, in large part because of the substantial controversies likely to attend any WPu disposition option in the United States. Estimates of the time required to accomplish major nuclear projects in the United States have historically tended to be optimistic. Thus, judgments about the absolute value of start dates and completion dates, and even about the relative timing of different options, must be considered somewhat tentative. These estimates are based on qualitative judgments, not on detailed design studies or detailed analysis of the political and regulatory processes required to implement these options.

⁵ We have assumed that the dismantling of surplus nuclear weapons containing the nominal 50 tons of plutonium has been completed by the end of the year 2000, and we begin the calculation of the integrated inventories at the beginning of the year 2001. Since none of the disposition options we are considering would begin the processing of pits before 2001, their integrated inventories of plutonium in pits will be identical up to that point and therefore irrelevant to comparisons. Conversion of pits to oxide is assumed to occur in connection with, and on the same time scale as, the MOX fuel fabrication or vitrification process.

TABLE 6-3 Timing of Disposition Following Commencement of Operation

U.S. Disposition of U.S. Weapons Plutonium

Vitrification with defense high-level waste (HLW): 1.3 weight percent plutonium, eight years. The indicated figures, suggested by preliminary analysis at Savannah River, should be considered illustrative. If a loading of 1.3 percent WPu in borosilicate glass is feasible with respect to control of criticality in glass manufacture and with respect to repository criticality, and if vitrification operations incorporating WPu commence in 2005 (Table 6-2), then addition of WPu at this loading to 2,200 logs scheduled for manufacture at the Savannah River site between 2005 and 2013 would account for the 50 tons of WPu.

CLWR, one-third MOX, FMEF: six reactors, 25 years. We assume, for the United States, use of 3,800-MWt LWRs @ CF = 0.75 and burnup of 42 MWd/kg with 4.0 percent plutonium in one-third MOX cores, implying 8.3 tons MOX and 330 kg plutonium per reactor-year (kgPu/Ry); FMEF MOX capacity is 50 tons of heavy metal per year (MTHM/yr), thus feeds six such reactors in steady state. First MOX batch requires 33 tons MOX per reactor, hence 200 tons MOX for six reactors. At 50 MTHM/yr, FMEF makes a one-third core MOX batch for one reactor every eight months. Assuming four months' lag between completion of a batch at FMEF and completion of loading of that batch at a reactor, start of FMEF operation on January 1, 2001, implies completion of loading a one-third MOX core at the first reactor on January 1, 2002. FMEF has finished fabricating 50 tons of WPu into MOX on December 31, 2025, and the last one-third MOX reload into a reactor takes place on May 1, 2026.

CLWR, full MOX, FMEF: two reactors, 25 years. Same reactor parameters as above but with full MOX enable two reactors to load the 50 tons of WPu in 25 years; first cores contain 100 tons of MOX, so first reactor can start May 1, 2003, after January 1, 2001, start of MOX plant operation and second reactor can start May 1, 2005; as in the first case, the last of the 50 tons of WPu will be transformed into MOX in 2025 and loaded into a reactor May 1, 2026.

CLWR, full MOX, FMEF: two reactors, 15 years. Same reactor parameters as above but with 6.8 percent plutonium in heavy metal. Start dates are as in previous case, but shorter campaign duration resulting from higher plutonium loading leads to last of WPu being loaded in 2016.

CLWR, new MOX plant: 12 reactors at one-third MOX or 4 reactors at full MOX, 12.5 years. Same reactor parameters but twice as many reactors as preceding two cases enable loading all of the 50 tons of WPu in the 12.5 years following MOX plant startup if a new MOX plant with capacity of 100 MTHM/yr is employed (this option trades increased cost and dispersion of disposition operations for shortened campaign duration); if new MOX plant commences operation in 2003 (Table 6-2), the last of the 50 tons of WPu will be transformed into MOX in 2015 and loaded into reactors in 2016.

ELWR, full MOX, FMEF: one reactor, 30 years. If licensing and political considerations lead to decision to build a new reactor at a U.S. government site, the fastest starting of the ELWR options examined in DOE's Plutonium Disposition Study would be an ABB-Combustion Engineering (ABB-CE) System-80+ or a General Electric (GE) ABWR, and minimizing plutonium transport would call for building it at the Hanford site (assuming FMEF provides the MOX fuel fabrication). Minimizing government investment favors building only one reactor and processing the 50 tons of plutonium in its 30-year lifetime (e.g., at 6.75 percent plutonium and 42.2 MWd/kg for an ABB-CE System-80+ with 25 MTHM/yr MOX fabrication, or 3.3 percent plutonium and 21.5 MWd/kg for a GE ABWR with 50 MTHM/yr MOX fabrication). MOX fabrication timing could be the same as for preceding FMEF cases, but reactor would not be ready for loading fuel until 2005 (Table 6-2) and last of WPu MOX would not be loaded until about 2030.

ALWR, full MOX, new MOX plant: four 610-MWe reactors, 17 years. If it were decided to use the WPu mission in order to demonstrate a more advanced LWR technology than the evolutionary reactors considered in the previous case, one might (for example) build four of the Westinghouse PDR-600 ALWRs and use them in conjunction with a 50-MTHM/yr MOX plant making 5.5 percent enriched fuel, which at a burnup of 40 MWd/kgHM would process the plutonium in about 17 years of operation. The MOX plant could commence operation in 2003 and finish in 2020, and the reactors would load MOX fuel starting in 2008, would load the last of it in 2021.

Table 6-3 continued

International Variations

Russian CLWR, one-third MOX. In Russia, use of the four 3,130-MWt VVER-1000 PWRs suitable for MOX operation at the same burnup, plutonium enrichment, and capacity factor assumed here for the U.S. LWRs (loading 272 kgPu/Ry in one-third MOX cores) would require almost 50 years to load 50 tons of WPu (probably longer than the life expectancy of these reactors); if four currently unfinished VVERs can be completed by 2000, then the total of eight could finish loading the 50 tons WPu by about 2025. If some of the older VVERs—those already operating—had to be retired at, say, age 30, the campaign's duration would increase correspondingly.

Russian CLWR, full MOX. If use of full-MOX cores with 4.0 percent WPu and burnup to 42 MWd/kgHM is feasible in VVER-1000 PWRs, then four of these could load 3.28 tons of WPu per year, and the nominal 50 tons of surplus Russian WPu could be loaded in about 15 years (given a MOX fuel fabrication capacity of 100 MTHM/yr).

Russian CLMR, full MOX. If the one operational Russian LMR, the 560-MWe BN-600, could be modified to use a full-MOX core, it would need 50 years to load the nominal 50 tons of surplus Russian WPu.

CANDU reactors, 100-percent MOX cores, FMEF. At a burnup of 9.7 MWd/kgHM, and CF = 0.80, a 2,832-MWt CANDU reactor loads about 85 MTHM/yr, containing 1,062 kgPu/yr at 1.2 percent, and two such reactors load 170 MTHM/yr containing 2,124 kgPu/yr. If the fuel is fabricated at the U.S. FMEF at a capacity of 170 MTHM/yr (which Atomic Energy of Canada, Limited, analyses indicate is practical given the simplicity of CANDU fuel), and the FMEF begins operation at this level in 2001, then the first CANDU could be loaded with a full-MOX core by the beginning of 2002 and the second by the beginning of 2003, and the last of the 50 tons of WPu would be fabricated into fuel in 2024 and loaded into the reactors in 2025.

- (2) The panel's best estimate of the minimum time requirements for accomplishing various disposition options indicates that, of all the disposition options considered in this report, the spent fuel options utilizing power reactors of currently operating types would have the earliest plausible start dates for conversion of pits to oxides and for the most important vulnerability-reducing step (loading into reactors in the case of reactor options, loading into melters for vitrification). The difference in start date between this option and the vitrification option, however, is smaller than the uncertainties in estimates of the start dates for these options.
- (3) Within the category of reactor options using reactors of currently operating types, the specific options with the earliest plausible start dates are (a) those that use existing or planned European MOX fabrication capacity (assuming rapid conclusion of any needed international agreements and arrangements for displacing civilian plutonium from these MOX plants) and (b) those that rely on completing the existing FMEF at Hanford for MOX fabrication.

TABLE 6-4 Key Dates and Integrated Inventories for Disposition Scenarios

Option	Plutonium Conversion:		Plutonium Loading: ^a		Integrated Inventory ^b (ton-yr Pu) as:	
	Start	End	Start	End	Pits	Oxides ^c
VITRF	2005	2013	2005	2013	425	13
CLWRa	2001	2025	2002	2026	625	33
CLWRb	2001	2025	2002	2026	625	38
CLWRc	2003	2015	2004	2016	413	25
CLWRd	2003	2015	2004	2016	413	30
CANDU	2001	2024	2002	2025	600	33 ^d
ELWR	2001	2030	2005	2032	625	79
ALWR	2003	2019	2008	2021	525	66

^a Loading means loading into a reactor or a melter.

^b Integrated inventories are computed from the year 2001 (see text).

^c Oxides refer here to plutonium oxides before loading into a reactor or melter.

^d Could be smaller, depending on how the continuous refueling capability of the CANDU is used.

Key to options and integrated-inventory calculations:

VITRF = Vitrification with defense HLW in the Defense Waste Processing Facility at Savannah River, with illustrative parameters given in Table 6-3. Plutonium is converted from metal to oxide and incorporated into glass at 5.6 tons/yr for 9 years. Pit integrated inventory is 50 tons \times 4 yr (2001–2004) plus $1/2 \times 50$ tons \times 9 yr (2005–2013) = 425 ton-yr. We assume an oxide stock equal to three months' input to melter will be maintained at the site, hence oxide integrated inventory is 5.6 tons / 4 \times 9 yr (2005–2013) = 13 ton-yr.

CLWRa = Six current 3,800-MWt LWRs with one-third MOX, FMEF MOX fabrication. Pit integrated inventory is $1/2 \times 50$ tons \times 25 yr (2001–2025) = 625 ton-yr of metal. Assuming that MOX fabrication plant runs 1 yr accumulating 50 MTHM before loading first 33-MTHM one-third MOX core and keeping 4-month (= 17 MTHM) minimum reserve, MOX integrated inventory as heavy metal is (1 yr \times 25 tons) + (23.67 yr \times 33 tons) + (0.33 yr \times 25 tons) + (0.33 yr \times 33 tons) = 825 MTHM-yr, \times 0.04 MTPu/MTHM = 33 MTPu-yr.

CLWRb = two current 3,800-MWt LWRs with full MOX, FMEF MOX fabrication. Pit integrated inventory as in CLWRa. Assuming that MOX fabrication plant runs 2.33 yr accumulating 117 MTHM before loading first 100-MTHM full-MOX core and keeping 4-month (= 17 MTHM) minimum reserve, MOX integrated inventory as heavy metal is (2.33 yr \times 58.5 tons) + (2 yr \times 67 tons) + (20.33 yr \times 33 tons) + (0.33 yr \times 33 tons) = 959 MTHM-yr, \times 0.04 MTPu/MTHM = 38 MTPu-yr.

CLWRc = Twelve current 3,800-MWt LWRs with one-third MOX, new MOX fabrication plant. Pit integrated inventory is 50 tons \times 2 yr (2001–2002) + $1/2 \times 50$ tons \times 12.5 years (2003–2015) = 413 ton-yrs. Assuming that MOX fabrication plant runs two-thirds of a year accumulating 66 MTHM before loading first 33-MTHM one-third MOX core and keeping 4-month (= 33 MTHM) minimum reserve, MOX integrated inventory as heavy metal is (2/3 yr \times 33 tons) + (11.67 yr \times 50 tons) + (0.33 yr \times 42 tons) + (0.33 yr \times 17 tons) = 620 MTHM-yr, \times 0.04 MTPu/MTHM = 25 MTPu-yr.

CLWRd = Four current 3,800-MWt LWRs with full MOX, new MOX fabrication plant. Pit integrated inventory is as in CLWRc. Assuming that MOX fabrication plant runs 1.33 years accumulating 133 MTHM before loading first 100-MTHM full-MOX core and keeping 4-month (= 33 MTHM) minimum reserve, MOX integrated inventory as heavy metal is (1.33 yr \times 66 tons) + (3 yr \times 83 tons) + (8 yr \times 33 tons) + (0.33 yr \times 42 tons) + (0.33 yr \times 17 tons) = 755 MTHM-yr, \times 0.04 MTPu/MTHM = 30 MTPu-yr.

CANDU = Two 769-MWe CANDU with full MOX at 1.2 percent WPu, FMEF MOX fabrication. (According to Atomic Energy of Canada, Limited, relative simplicity of CANDU fuel makes it possible for FMEF to meet the 170-MTHM/yr fabrication requirement for this scenario.

Rate of plutonium processing into fuel is 2,124 kg plutonium per year, so the fabrication operation runs for 24 years from 2001 and the integrated inventory in pits is $1/2 \times 50$ tons Pu \times 24 years = 600 ton-yr. If the average working inventory of fabricated MOX fuel before loading into the reactors is one half core's worth per reactor, hence one full core's worth altogether, the associated integrated inventory of plutonium in oxide is 116 MTHM \times 0.012 MT WPu per MTHM \times 24 yr = 33 ton-yr.

ELWR = One evolutionary 3,900-MWt LWR with full MOX (GE ABWR at 21.5 MWd/kgHM and 3.3 percent plutonium), FMEF MOX fabrication at 50 MTHM/yr. Assuming the MOX plant is operated as soon as possible (in order to begin the disposition process as soon as possible, even though the reactor will not be ready to receive the fuel as rapidly as the MOX plant can produce it), the pit integrated inventory will be 625 MTHM-yr as in the other FMEF cases. The oxide inventory ramps from zero at the beginning of 2001 to 200 MTHM at the end of 2004 just before fuel loading into the reactor begins in 2005, so the contribution to the integrated inventory from this period is $1/2 \times 200$ tons \times 4 yr = 400 MTHM-yr; then the inventory drops to about 50 MTHM as the first full core is loaded, and for the next 26 years this 50 MTHM is a base or reserve above which the MOX plant's annual output ramps an additional inventory from zero just after a reload to 50 MTHM just before a reload; thus the average inventory during this 26 years is 75 MTHM in oxide and the contribution to the integrated inventory is 1,950 MTHM-yr; at the end of one more year the last 50 tons are loaded, so this year contributes another 50 MTHM-yr, and we have altogether an integrated inventory of $400 + 1,950 + 50 = 2,400$ MTHM-yr, which at 0.033 MTPu/MTHM is 79 MTPu-yr.

ALWR = Four advanced 610-MWe LWRs with full MOX, new MOX fabrication plant at 53-MTHM/yr output. The integrated inventory in pits is 50 tons \times 2 yr (2001–2002) + $1/2 \times 50$ MTHM \times 17 yr (2003–2019) = 525 MTHM-yr. The oxide inventory accumulates at 53 MTHM per year from the beginning of 2003 to the end of 2007 just before the reactors are ready to receive fuel, contributing $1/2 \times 5$ yr \times 265 MTHM = 663 MTHM-yr to the oxide integrated inventory; from 2008 through 2019 the working inventory ramps from 0 to 53 tons each year, averaging 26.5 MTHM, and allowing for a 4-month reserve of 18 MTHM makes the integrated-inventory contribution for this period 45 MTHM \times 12 yr = 540 MTHM-yr; the total integrated inventory of heavy metal in oxide is then $663 + 540 = 1,203$ MTHM, which at 0.055 MTPu/MTHM is 66 ton-yr plutonium.

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- (4) Use of full-MOX as opposed to one-third MOX cores in the current LWR options, to the extent that it is practical, would reduce the number of reactors needed for the disposition campaign, but it would not significantly affect the start dates (unless demonstrating full-MOX capability, or attaining it through modifications, added more than the two years assumed here to the reactor timetable); and it would only shorten the duration of the campaign, once started, if MOX fuel fabrication capacity were expanded beyond what has been assumed here. Integrated inventories are not very sensitive to the choice of full-MOX versus one-third MOX cores (although of course the geographic dispersion of the inventories is likely to be sensitive to this choice).
 - (5) Construction of a new MOX fabrication plant with twice the capacity of FMEF (or expansion of FMEF to such higher capacity, which we assume would take a comparable length of time) would delay start dates for the current LWR options but could accelerate completion dates by about a decade (given the use of the doubled number of reactors that a doubled MOX capacity would permit). This choice would also reduce

the post-2000 integrated inventory of pits by some 30 percent (the faster drawdown more than compensating for the later start date).

- (6) Use of commercial CANDU reactors with full-core MOX produced at FMEF would provide the possibility of carrying out disposition of all of the nominal 50 tons of WPu in two reactors over a period of about 25 years, with start dates, completion dates, and post-2000 pit integrated inventory about the same as those of the CLWR options (assuming that the bilateral negotiations needed to arrange this option could be completed reasonably quickly).
- (7) Construction of a new evolutionary LWR for WPu disposition, as might be done if political difficulties preclude using an already operating or now partly completed LWR, would also permit loading all of the 50 tons of WPu into a single reactor during its lifetime, at the cost of modestly delaying the start date for plutonium loading, delaying by about five years the completion of the campaign, and significantly increasing the pre-load oxide integrated inventory.
- (8) Use of advanced LWRs would entail the latest start date for plutonium loading of any of the options besides other advanced reactors, and it would not offer compensating advantages over current- and evolutionary-LWR cases in completion date or integrated inventories.
- (9) The panel's best estimate of the earliest plausible start date for the vitrification option is a few years later than our estimate of the earliest plausible start date of the near-term reactor options, although as noted earlier this difference is smaller than the large uncertainty in estimating when either of these classes of options could actually begin. The vitrification option could have a completion date earlier than any of the reactor options (excluding implausibly high MOX fabrication capacities). Under the parameters postulated here, it would also have a post-2000 pit integrated inventory equal to the best of the reactor options, and a pre-load oxide integrated inventory better than any of them.
- (10) Advanced reactors other than ALWRs—including MHTGRs, ALMRs, MSRs, PBRs, and ABCs—would have such a severe disadvantage in start dates, completion dates, and post-2000 pit integrated inventories that we rate them as unacceptable in their timing for the near-term WPu-disposition mission; choosing one of these options would needlessly prolong the hazards of storing excess WPu in readily weapon-usable form.
- (11) The timing for use of analogous options for disposition of Russian plutonium is not likely to be significantly more favorable than that summarized in the preceding numbered subparagraphs and in Table 6-4, and could easily be worse.

For all reactor options, there is an important trade-off to be made between the number of reactors and sites involved and the completion time of the disposition campaign. Using WPu in a large number of reactors at several sites would substantially reduce the amount of time required to complete disposition, with the associated security benefits described at the outset of this section. In an extreme case, for example, if plutonium fuel could be produced for all the reactors in the United States, the excess WPu could be completely loaded into reactors within a few months, rather than a few decades, after the operation began. On the other hand, approaches involving many sites would also have important disadvantages: (1) there would be more plutonium transportation steps (probably the point in the disposition process when the material is most vulnerable to forcible theft, as described below), and more sites at which the plutonium would have to be secured and accounted for; (2) a larger MOX fabrication capacity would be needed to fabricate more plutonium into fuel in a shorter time (possibly requiring more time to build), and this capacity would then be idled (if it were located in the United States) as soon as the WPu disposition campaign was complete; (3) political and licensing issues would have to be overcome at a larger number of locations; and (4) symbolically, if the United States chooses to continue to generally oppose the use of fuel cycles involving separated plutonium, involving a larger fraction of the U.S. nuclear industry in the use of plutonium would tend to counteract this message.

Ultimately, choices balancing these advantages and disadvantages of increasing the number of sites will have to be based on educated judgment; the panel is not aware of any defensible means by which these advantages and disadvantages can be quantified and rigorously compared. While it is true that thousands of assembled nuclear weapons are transported each year in the United States and the former Soviet Union, and that plutonium shipments in Europe are commonplace, it is nevertheless the panel's judgment that the advantages of limiting plutonium disposition to one or two sites outweigh the timing disadvantages of doing so—particularly as, in the United States, Russia, and some other countries, sites exist with several reactors at a single location, offering the possibility of limiting the time required for plutonium disposition while simultaneously keeping the number of separate sites to a minimum. This judgment is generally reflected in the assumptions used in Table 6-3.

Other Indices, Barriers, and Threat-Barrier Interactions

Some of the security-related characteristics of plutonium in different forms are summarized in Table 6-5. These characteristics have been used, together with the definitions and criteria presented in Chapter 3 (see "Specific Security Concerns and Threat Characteristics" and "A Matrix Scheme for Characterizing Options"), to arrive at the set of characterizations presented in Table 6-6 of in-

(text continues on p. 274)

TABLE 6-5 Some Security-Related Characteristics of Plutonium in Different Forms

Form of Plutonium	Mass per Item (kg)	Max Item Dim (cm)	Pu per Item (kg)	Pu Conc (kg/kg)	Gamma Dose Rate (rem/hr):		Notes	
					At Surface	At 1 meter		
Intact pit (WPu metal)	ca 4	ca 15	ca 4	1	0.85	0.005	1	
RPu metal sphere, δ -phase	6	9	6	1	17	0.03	1	
PuO ₂ powder, WPu ^a	(powder @ 1 g/cm ³)			0.88	1	0.009	2	
PuO ₂ powder, RPu ^b	(powder @ 1 g/cm ³)			0.88	20	0.2	2	
MOX fuel pellet, WPu	0.006	1	3×10^{-4}	0.05	0.05	1×10^{-6}	3	
MOX fuel pellet, RPu	0.006	1	3×10^{-4}	0.05	1	2×10^{-5}	3	
LWR MOX fuel rod, WPu	2.5	410	0.1	0.04	0.03	1.4×10^{-4}	4	
LWR MOX fuel rod, RPu	2.5	410	0.1	0.04	0.7	3×10^{-3}	4	
LWR MOX fuel assembly, WPu	658	410	25	0.038	0.03	4×10^{-3}	5	
LWR MOX fuel assembly, RPu	658	410	25	0.038	0.7	0.08	5	
MHTGR WPu fuel block	100	80	0.8	0.008	0.5	0.02	6	
Irradiated LWR MOX fuel assembly, WPu								
0.4 MWd/kgHM,	2 yr	658	410	23	0.035	38,000	4,500	7
	10 yr	658	410	23	0.035	180	22	7
	30 yr	658	410	23	0.035	79	9	7
	100 yr	658	410	23	0.035	16	2	7
40 MWd/kgHM,	10 yr	658	410	18	0.027	18,000	2,200	7
	30 yr	658	410	18	0.027	7,900	940	7
	100 yr	658	410	18	0.027	1,600	190	7
50 MWd/kgHM,	10 yr	658	410	9	0.014	23,000	2,800	7
	30 yr	658	410	9	0.014	10,000	1,200	7
	100 yr	658	410	9	0.014	2,000	240	7
MHTGR WPu fuel block irradiated to 580 MWd/kgHM								
	2 yr	100	80	0.2	0.002	6,600	660	8
	10 yr	100	80	0.2	0.002	1,800	180	8
	30 yr	100	80	0.2	0.002	1,000	100	8
	100 yr	100	80	0.2	0.002	200	20	8
Borosilicate glass log with WPu and HLW								
small, 1.3% Pu, 20% HLW	250	50	3	0.013	not calculated		9	
large, 1.3% Pu, 20% HLW	2,200	300	22	0.013	5,200	900	9	
same, + 10 years	2,200	300	22	0.013	4,200	720	9	
same, + 30 years	2,200	300	22	0.013	2,600	450	9	
same, + 100 years	2,200	300	22	0.013	520	90	9	

ABBREVIATIONS:

Max = maximum
 Dim = dimension
 Conc = concentration

^aWPu assumed to contain 0.2 weight percent Am-241 (from initial 0.4 percent Pu-241, aged 14 years)

^bRPu, reactor plutonium, assumed to contain 4 weight percent Am-241 (from initial 9 percent Pu-241, aged 12 years).

NOTES to Table 6-5:

1. If the rate at which energy is released in gamma rays of a given energy in a sphere or cylinder (or spherical or cylindrical shell) of material is D_a (J/kg \times h), and if the radius of this material (or, in the case of a shell, the thickness), R (cm), is large compared to the mean length for gamma-energy absorption in the material, $L = 1 / [(\mu_{Pu}) \times (\text{density of Pu})]$, where μ_{Pu} (cm^2/g) is the mass energy-absorption coefficient of plutonium metal for the relevant gamma-ray energy, then it is easy to show that the rate of energy absorption in the material at its outer surface is $0.5 \times D_a$, and the contact dose rate in tissue at the surface is $D_s = 0.5 \times D_a \times (\mu_t/\mu_{Pu})$, where μ_t is the mass energy absorption coefficient of tissue. Now, for WPu containing 0.2 weight percent 430-yr Am-241, which emits a 0.06 million electron volt (MeV) gamma ray in 36 percent of its decays and which dominates the total gamma emission in this plutonium, the gamma-energy release rate is

$$\begin{aligned} D_a &= (0.002 \text{ g Am-241/gPu}) \times [(6 \times 10^{23} \text{ nuclei Am-241}) / (241 \text{ g Am-241})] \\ &\quad \times (0.693 \text{ disintegration/nucleus}) / [(430 \text{ yr}) \times (8760 \text{ h/yr})] \\ &\quad \times (0.36 \times 0.06 \text{ MeV/dis}) \times (1.6 \times 10^{-13} \text{ J/MeV}) \times (1,000 \text{ g/kg}) \\ &= 3.2 \text{ J/kg-h} . \end{aligned}$$

Now, with μ_t the mass energy-absorption coefficient of tissue at 0.06 MeV, taken to be equal to that of water at $0.032 \text{ cm}^2/\text{g}$, and with μ_{Pu} , the mass energy-absorption coefficient of plutonium at 0.06 MeV, scaled from that of uranium ($= 5.78 \text{ cm}^2/\text{g}$) by $5.78 \times (94/92)^2 = 6.03 \text{ cm}^2/\text{g}$, we have $L = 1 / [(6.03 \text{ cm}^2/\text{g}) \times (19.6 \text{ g/cm}^3)] = 0.0085 \text{ cm}$, so the assumption that this is small compared to the radius or shell thickness is clearly satisfied, and we have

$$D_s = 0.5 \times D_a \times \mu_t/\mu_{Pu} = 0.5 \times (3.2 \text{ J/kg} \times \text{h}) \times 0.032/6.03 = 8.5 \times 10^{-3} \text{ J/kg-h} .$$

In the SI units for absorbed dose, grays = joules/kg, this is 8.5×10^{-3} grays/h, which in traditional dose units is 0.85 rads/h; since sieverts = grays $\times QF$ and rem = rads $\times QF$, where QF , the quality factor, is unity for gamma rays, this is equivalent to 8.5×10^{-3} sieverts/h or 0.85 rem/h. (This is the dose rate at the surface of tissue in contact with the plutonium sphere; the average dose rate in a body of finite dimension—say, 30 cm thick—in contact with the material would be somewhat lower by virtue of attenuation both by absorption and geometry.) By spherical symmetry, and neglecting the very small attenuation in air over distances of the order of a meter, the dose at the surface of tissue 1 m from the surface of a sphere of the indicated composition would be $D_r = D_s \times (R/r)^2$, with R the radius of the surface in meters and $r = 1 + R$. Thus with $R = 0.08 \text{ m}$ we have

$$D_r = 8.5 \times 10^{-3} \text{ Sv/h} \times (0.08/1.08)^2 = 4.7 \times 10^{-3} \text{ Sv/h} = 0.005 \text{ rem/h} .$$

In the case of a sphere of reactor plutonium, the concentration of the gamma-emitting Am-241 is, for our assumed compositions, 20 times higher, so the surface dose rate is $20 \times 8.5 \times 10^{-3} \text{ Sv/h} = 0.17 \text{ Sv/h} = 17 \text{ rem/h}$, and for a sphere of radius 4.5 cm the dose rate 1 m from the sphere's surface is

$$D_r = 0.17 \text{ Sv/h} \times (0.045/1.045)^2 = 3.2 \times 10^{-4} \text{ Sv/h} = 0.032 \text{ rem/h} .$$

2. Calculated as in note 1, for spherical powder mass containing 4 kg of plutonium; PuO_2 mass energy-absorption coefficient taken to be $0.882 \times 6.03 + 0.118 \times 0.032 = 5.32 \text{ cm}^2/\text{g}$, where 0.032 is the mass energy-absorption coefficient for oxygen.

3. Calculated as in note 2, for MOX fuel pellet with diameter 0.8 cm and length 1 cm, density 10.5 gm/cm^3 , 5.5 percent plutonium in heavy metal, and 0.2 percent Am-241 in plutonium.

4. Calculated as in note 3, with active rod length having 365 pellets, and with dose at 1 m from surface, now scaling according to cylindrical rather than spherical geometry, given by $D_r = D_s \times R/r$, where R is surface radius (here 0.004 m) and $r = 1 + R$. Rod contains 2.1 kg MOX surrounded with 0.4 kg stainless steel cladding. Effect of 0.5 mm cladding layer on dose rate from 0.06-MeV gamma ray

estimated as $\exp[-(0.96 \text{ cm}^2/\text{g}) \times (8 \text{ g/cm}^3) \times (0.05 \text{ cm})] = 0.68$, where $0.96 \text{ cm}^2/\text{g}$ is mass energy-absorption coefficient for 0.06 gamma MeV in iron and 8 g/cm^3 is density of iron.

5. Calculated as in note 4, with Westinghouse PDR-600 17×17 fuel assembly containing 264 fuel rods plus control-rod channels (each assembly contains 461 kgHM, 524 kg MOX, 623 kg MOX with cladding, 658 kg assembly mass including spacers, springs, etc.). Assembly is square, 21.4 cm on a side, approximated for dose-at-distance calculation as a cylinder with $r = 4 \times 21.4 \text{ cm} / 2 \times \pi = 13.6 \text{ cm}$.

6. Based on General Atomics figure of 700 mrem/h surface dose rate on an unirradiated fuel block 36 cm across, containing 0.8 kg WPu with 0.3 percent Am-241 (GA 1993, p. 6-26), scaled to 500 mrem/h for 0.2 percent Am-241, and scaled to dose rate at 1 m via approximations referenced in note 1, above.

7. Westinghouse PDR-600 fuel assembly, as described in note 5; dose rate at 1 m after two years for 0.4 MWd/kgHM from Westinghouse (1993); surface dose rate derived from this figure via $D_r = D_s \times (R/r)$ (see note 4) with $R = 13.6 \text{ cm}$, $r = 113.6 \text{ cm}$. Dose rates at 1 m after 10, 30, and 100 years for 40-MWd/kgHM and 50-MWd/kgHM irradiations scaled from detailed calculations performed at the Idaho National Engineering Laboratory (INEL) (Schnitzler 1993), using the ORIGEN isotope generation and depletion code and the QAD point-kernel shielding code with a complete model of this fuel assembly, assuming 3.2 percent U-235 fuel irradiated to 33 MWd/kgHM. We assume difference in gamma doses between MOX and LEU spent fuel is small, since both produce about 3 Ci Cs-137 per MWd, and the 0.66-MeV gamma from 85 percent of Cs-137 decays is the dominant source of gamma rays from spent fuel between 5 and 100 years after discharge. INEL results for 33 MWd/kgHM and 3 feet from surface of fuel assembly are 20 Sv/h at 10 years, 8.4 Sv/h at 30 years, and 1.7 Sv/h at 100 years. Scaling factor for radius 1 m instead of 3 feet is $(13.6 + 91.4) / (13.6 + 100) = 0.924$. Scaling factors for higher irradiations are $40/33 = 1.21$ and $50/33 = 1.51$. Surface doses scaled from doses at 1 m using $D_r = D_s \times (R/r)$ with $R = 13.6 \text{ cm}$, $r = 113.6 \text{ cm}$. Dose rates for 0.4-MWd/kgHM irradiation at 10, 30, and 100 years scaled linearly with irradiation from 40-MWd/kgHM results. To check the approximate method of notes 1 and 4, which we use elsewhere in this table when no detailed calculations are available, we here use the approximate method to recalculate, for comparison with the indicated "exact" calculation, the dose to be expected from LEU fuel irradiated to 33 MWd/kgHM. We assume that: (a) the gamma dose from spent fuel is dominated between 5 and 100 years after discharge by the 0.66-MeV gamma ray emitted in 85 percent of decays of 30.17-year half-life Cs-137; and (b) the quantity of Cs-137 in spent fuel at discharge is 3 Ci/MWd. For the fuel assembly described in note 5, the quantity of Cs-137 per assembly after irradiation to 33 MWd/kgHM is $(3 \text{ Ci/MWd}) \times (33 \text{ MWd/kgHM}) \times (461 \text{ kgHM}) = 45.6 \times 10^3 \text{ Ci}$; after 10 years of cooling time the Cs-137 inventory is $(45.6 \times 10^3 \text{ Ci}) \times \exp(-0.693 \times 10/30.17) = 36.3 \times 10^3 \text{ Ci}$. The energy release rate in the assembly is

$$(36.3 \times 10^3 \text{ Ci}) \times (3.7 \times 10^{10} \text{ dis/Ci} \times \text{sec}) \times (3,600 \text{ sec/h}) \times (0.85 \times 0.66 \text{ MeV/dis}) \\ \times (1.6 \times 10^{-13} \text{ J/MeV}) / (658 \text{ kg}) = 659 \text{ J/kg} \times \text{h}.$$

The mass energy absorption coefficient for the whole assembly is estimated by considering the assembly to be a homogeneous mixture of 507 kg UO_2 (allowing for 3.3 percent burnup) with μ for 0.66-MeV gammas = $0.0718 \text{ cm}^2/\text{gm}$, 134 kg steel with μ for 0.66-MeV gammas = 0.0282 , 15 kg fission products (3.3 percent of the initial 461 kgHM) with μ for 0.66-MeV gammas taken as that of tin at 0.032, and 2 kg oxygen (associated with the fission products) with μ for 0.66-MeV gammas = 0.0292 ; hence

$$\mu_{\text{tot}} = (507 \times 0.0718 + 134 \times 0.0282 + 15 \times 0.032 + 2 \times 0.0292) / 658 = 0.0619 \text{ cm}^2/\text{g}.$$

The mass energy absorption coefficient for water for 0.66-MeV gammas is $0.033 \text{ cm}^2/\text{g}$, so the dose at the surface is $D_s = 0.5 \times (659 \text{ J/kg} \times \text{h}) \times (0.033/0.0619) = 176 \text{ J/kg-h}$, i.e., 176 Sv/h or 17,600 rem/h. The dose at a distance of 1 m from the fuel assembly surface is

$$D_r = D_s \times (R/r) = (176 \text{ Sv/h}) \times (13.6/113.6) = 21.1 \text{ Sv/h}.$$

This is within 15 percent of the "exact" calculation's result of 18.4 Sv/h for the same conditions, and thus confirms the usefulness of the approximate approach. Note, finally, that the diminution of plutonium over time in spent fuel because of the decay of 14.4-year Pu-241 (12–15 percent of total plutonium in high burnup LWR MOX) to 430-year Am-241 is not reflected in the plutonium column here, because Am-241 is itself weapon-usable (with critical-mass properties similar to Pu-242; see Mark 1993).

8. Doses at 1 m for times 2, 10, and 30 years are from GA (1994). Surface dose rates calculated from these by the approximation described in note 6. Doses at 100 years calculated from those at 30 years assuming a 30-year half-life of the gamma dose in this time period.

9. The large log has a glass volume of 625 liters, a glass density of 2.7 kg/liter, a glass radius of 29 cm, and a steel canister thickness of 0.9 cm. The glass content amounts to just under 1,700 kg and the steel jacket has a mass of 450 kg. Given that Savannah River wastes to be contained in the glass comprise 2.6×10^8 Ci of Cs-137 and Sr-90, and that the Cs-137 whose 0.66-MeV gamma will dominate the gamma dose constitutes about half of this (Berkhout et al. 1993, p. 184 and note 53), and given the planned campaign of 6,105 logs to contain all of the HLW at that site, the concentration of Cs-137 in the glass will be 1.3×10^8 Ci / (6,105 logs \times 625 liters/log \times 2.7 kg/liter) = 12.6 Ci/kg. The energy release rate in glass from this cesium's 0.66-MeV gamma is then

$$\begin{aligned} D_a &= 12.6 \text{ Ci/kg} \times 3.7 \times 10^{10} \text{ dis/sec} \times \text{Ci} \times (0.85 \times 0.66 \text{ MeV})/\text{dis} \\ &\quad \times 3,600 \text{ sec/h} \times 1.6 \times 10^{-13} \text{ J/MeV} \\ &= 150 \text{ J/hr-kg} . \end{aligned}$$

The mass energy-absorption coefficient for the 0.66 MeV gamma in glass with 1.3 percent plutonium is about $0.787 \times 0.0293 + 0.20 \times 0.033 + 0.013 \times 0.078 = 0.031$, where 0.0293, 0.033, and 0.078 cm^2/g are the mass energy-absorption coefficients for 0.66-MeV gamma radiation in SiO_2 , tin, and plutonium, respectively (taking tin as representative of fission products). The energy-absorption length is $(0.031 \text{ cm}^2/\text{g} \times 2.7 \text{ gm/cm}^3)^{-1} = 11.9 \text{ cm}$, which is about 40 percent of the radius of our log, so the approximate formula that gives the surface dose rate as $D_s = 0.5 \times D_a \times (\mu_a / \mu_{\text{glass}})$ is not very good but perhaps is tolerable. It gives $D_s = 0.5 \times 150 \text{ J/kg-h} \times 0.033/0.031 = 80 \text{ Grays/h} = 8,000 \text{ rem/h}$. The effect of the 0.9 cm steel container wall can be approximated by

$$\exp[-(0.028 \text{ cm}^2/\text{g}) \times (7.8 \text{ g/cm}^3) \times (0.9 \text{ cm})] = 0.82 ,$$

where $0.028 \text{ cm}^2/\text{g}$ is the mass energy-absorption coefficient for iron at 0.66 MeV, giving a surface dose rate outside the container of $0.82 \times 80 \text{ Sv/h} = 66 \text{ Sv/h} = 6,600 \text{ rem/h}$. Assuming the typical log is not produced until another 10 years after the date of the inventory figure with which we started, the dose rate will be down by another $\exp[-(10 \times 0.693/30.2)] = 0.795$, giving 5,200 rem/h. To get the dose rate at a radius of $r = 1 + R = 1.3 \text{ m}$, notice that if this radius is considered small compared to the 3-m cylinder length, the approximation for that circumstance would give $D(r = 1.3\text{m}) = 5,200 \times (30/130) = 1,200 \text{ rem/h}$. In the limit of a spherical source, we would get $D(r = 1.3\text{m}) = 5,200 \times (3/1.3)^2 = 280 \text{ rem/h}$. The first approximation is closer to right, so we estimate 900 rem/h. Doses at subsequent times figured based on the 30-year half-life of Cs-137. No figures are presented for the small log in this table because the energy-absorption length is not small compared to the radius of the log and the approximate approach used here is therefore not valid.

trinsic barriers to weapons use associated with the main plutonium forms encountered in reactor-related disposition options.

Of particular interest in Table 6-6 are the conclusions about the intrinsic barriers associated with the final forms of the plutonium for the spent fuel and vitrification options—that is, MOX spent fuel assemblies and WPu oxide in glass logs with HLW, respectively. As indicated in the table, the MOX option has a modest advantage in terms of the isotopic barrier, whereas the vitrification option has a modest advantage in terms of the mass and bulk of the objects that a thief or divertor would need to remove and transport. The radiologic barrier is higher for the spent fuel than for the vitrified log, but it is high enough for the latter that we do not give much weight to the difference. It is our judgment that the chemical barrier is roughly comparable for the two options, in terms of the complexity of and technological sophistication needed for the steps required to separate plutonium from the two final waste forms. Overall, we rate both of these final forms as meeting the “spent fuel standard.”

Table 6-7 presents characterizations of the implementation-dependent barriers and overall vulnerability to different threats for the vitrification option and for a version of the current light-water-reactor option in which two 1,300-MWe LWRs using full-MOX cores load the nominal 50 tons of WPu over a period of 25 years (option CLWRb in Table 6-4). The implementation-dependent barrier evaluations follow the prescriptions set forth in Chapter 3 under “Specific Security Concerns and Threat Characteristics.” The vulnerability evaluations depend, as indicated in Chapter 3, on the *interaction* of the intrinsic and implementation-dependent barriers with the characteristics of the different classes of threat.⁶ Key features of these interactions are:

- *Overt Diversion.* Vulnerability to overt diversion will depend mainly on the intrinsic barriers, insofar as most of the implementation-dependent barriers would be irrelevant to the United States or Russia were one of these states to contemplate overtly reversing the disarmament process to reincorporate the WPu into its arsenal. The one exception to the irrelevance of implementation-dependent barriers would be the institutional barriers in cases where the plutonium had actually been placed, at some point in the disposition process, under international control and removed from the territory of the original possessor state.

⁶ Recall from Chapter 3 that “overt diversion” refers to the situation in which a country chooses to reuse the (previously surplus) WPu in its possession for weapons purposes, without attempting to conceal this activity; “covert diversion” refers to the same situation except that the country attempts to conceal what it is doing; “theft” refers to acquisition of the material by unauthorized entities other than the initial possessor state, and might be attempted by force (“forcible theft”), by stealth (“covert theft”), or openly but without the need for the force (“overt theft”), as might occur following a breakdown of national authority. Intrinsic and implementation-dependent characteristics relevant to assessing vulnerability of an option to overt theft are an obvious combination of the characteristics germane for the cases of forcible and covert theft, so we give no further separate attention to overt theft here.

TABLE 6-6 Intrinsic-Barrier Characterization for Plutonium Forms Encountered in Reactor-Related Disposition Options

Form of Plutonium	Kg of Material per Kg of Plutonium	Qualitative Characterization of Other Intrinsic Barriers (0 = lowest, 4 = highest)				Overall Intrinsic-Barrier Characterization (0 to 4)
		Iso- topic	Chem- ical	Radio- logic	Mass/ Bulk	
Nuclear-weapon pit	1	1	0	2	1	1
WPu oxide powder	1.1	1	1	2	0	1
MOX (WPu/U) powder	20	1	2	1	0	2-
MOX fuel rod	25	1	2	1	2	2
MOX fuel assembly	25	1	2	1	3	2+
MOX spent fuel assembly ^a	35	2	4	4	3	4
WPu oxide in HLW log	50	1	4	4	4	4

NOTE: The basis for this characterization scheme is elaborated in Chapter 3 under "Specific Security Concerns and Threat Characteristics."

^a 40 MWd/kgHM.

In any other circumstances, the vulnerability to overt diversion will be essentially proportional to the physical ease of restoring the plutonium to weapon usability, or, in other words, inversely proportional to the magnitude of the intrinsic barriers: an overt divertor presumably would choose, if plutonium at several stages of disposition were available, to divert that which was closest to weapon-usable form; and a decision whether to divert the "dispositioned" plutonium at all, as opposed to obtaining additional WPu from other facilities from scratch, would likewise be related to the costs of overcoming the intrinsic barriers in comparison to the costs of from-scratch production. With respect to this class of threats and also with respect to covert diversion, the isotopic barrier may have greater importance in relation to the other intrinsic barriers than is the case with respect to theft, in that this characteristic probably is more likely to influence the thinking of the United States or Russia about whether to divert than it is to influence the thinking of countries or subnational groups contemplating theft as a means to acquire or augment a much less sophisticated arsenal.

- *Covert Diversion.* Vulnerability to covert diversion will depend not only on the factors just described that pose barriers to processing and using the material for weapons, but also on the intrinsic and implementation-dependent barriers to diverting it without detection. Thus, the

dangers of covert diversion are reduced when the material is in forms that are bulky and heavy, when it is held at one or a small number of well-guarded sites, and when it is rigorously monitored and accounted for at each handling and processing step, under international oversight.

- *Forcible Theft.* In contrast to the diversion threats, the threats of theft will tend to be on behalf of potential bomb-makers with less sophisticated requirements and less sophisticated materials-processing and weapon fabrication capabilities. Accordingly, in the case of theft the chemical and radiologic barriers will be more important, in the intrinsic-barrier category, and the isotopic barrier less important than in the case of diversion; the dilution and mass and bulk barriers will be somewhat important—less so, however, than for covert diversion and theft—and the implementation-dependent barriers (location and exposure, containment, and institutional) will be extremely important. Transport is probably the hardest operation to protect against forcible theft, so transport links involving material of low intrinsic barriers have the highest vulnerability against this class of threat.
- *Covert Theft.* The relevance of intrinsic barriers to assessing the vulnerability to covert theft is much the same as in the case of forcible theft, except that the relevance of dilution and mass and bulk is higher in the covert case. With covert theft, one is almost inevitably dealing with an “insider” component to the threat, and the greatest vulnerabilities tend to occur where material of low intrinsic barriers and high portability flows through processes where maintaining accurate material accounting is most difficult—this means, above all, the processing step in which plutonium metal is converted to oxide (which occurs in both the MOX/spent fuel and vitrification options) and the further processing steps (MOX/spent fuel option only) of blending the mixed oxides and fabricating the fuel pellets.

As is clear from a glance at Table 6-7, the vitrification option has some advantage in overall vulnerability, compared to the MOX/spent fuel option, insofar as vitrification entails fewer processing steps and probably fewer transport steps (unless the MOX fabrication capacity and MOX-burning reactors are co-located) in which vulnerabilities to covert diversion and to theft are highest.⁷ On the other hand, the vulnerability of vitrification to both overt and covert diversion threats involving the final plutonium form is higher than the corresponding vulnerability of the MOX/spent fuel option, because the isotopic barrier is modestly lower for the vitrification option. Taking into account the modest advantage of the MOX options with respect to security of the final plutonium form

⁷ This conclusion is compatible with the opinions of a number of safeguards experts consulted by the panel, who said that they considered MOX fuel fabrication plants more difficult to safeguard than a vitrification plant would be.

and the modest advantage of the vitrification option with respect to security of the operations preceding final disposition, we characterize the two classes of options as comparable overall in terms of the barriers and vulnerabilities relating to security. To put it another way, the number of different and incommensurable variables, and the uncertainties in assessing each of them, make it impossible to determine whether either of these options offers superior overall security compared to the other.

The other options from Table 6-4, including those that employ CANDU reactors and evolutionary and advanced LWRs, would have substantially the same ratings in the framework of Table 6-7 as those shown there for the current LWRs with full-MOX cores. The differences in this framework would relate mainly to cases in which multiple reactors would not be required (amounting to 1 unit on the location and exposure part of the implementation-dependent barrier characterization for the irradiation step) or in which MOX fuel fabrication was co-located with the reactor (thus omitting the line characterizing vulnerabilities of fresh fuel transport).⁸ Except for the obvious advantages of cases that minimize the number of reactors used or eliminate transport steps by co-location, then, the main security and vulnerability differences among the variants of the LWR- and CANDU-based MOX options are those associated with timing and the related question of integrated inventories of pits and oxides, as summarized in Table 6-4.

More advanced reactor options, such as the MHTGR and ALMR, would also not differ greatly, in their ratings in the framework of Table 6-7, from the LWR case shown, assuming these advanced reactor types were used on a once-through basis. The MHTGR could gain 1 unit in the isotopic barrier rating when operated at very high burnup, and the co-location of fuel fabrication facilities with the reactor(s)—as would be possible with either of these reactors but also possible, in principle, with LWRs where new reactors or new fabrication capacity were sited to achieve this—would provide some gain in security; but we regard these potential improvements as insufficient to offset the large liabilities with respect to timing suffered by all the advanced reactor types, as discussed in the preceding section. This conclusion is reinforced, of course, by the circumstance that the current-reactor options and the vitrification option already achieve, at their end-points, a standard of security comparable to that of the large quantities of spent reactor fuel that already exist, so that there would be little overall security gain from using an advanced reactor type to push the WPU to a higher-than-spent fuel standard unless the same processing were contemplated for all the civilian spent fuel as well.

⁸ There are some additional differences between CANDUs and LWRs, for example, in the size of the fuel bundles (smaller for CANDU) and the height of the radiologic barrier (lower for typical CANDU burnups than for typical LWR burnups), but these differences would not change our assessment of the overall vulnerabilities from the ratings shown for LWRs in Table 6-7.

TABLE 6-7 Barriers and Vulnerability for Selected Disposition Options

Option and Step	Duration (yr)	Overall Intrinsic Barriers (0-4)	Implementation-Dependent Barriers (0 = low, 4 = high)			Vulnerability with respect to the threat of:			
			Loc/Exp	Containment	Institutional	Overt Diversion	Covert Diversion	Forcible Theft	Covert Theft
Vitrification with Defense High-Level Wastes ^a									
Pit storage	13	1	3	3	3-4	high	med	med	med
Pit transport	9	1	0	2	3	high	high	high	med
Oxide production	9	1	2	2	2-4	high	high	med	high
Oxide storage	9	1	3	2	2-4	high	med	med	med
Log production	9	1→4	2	2	2-4	high	high	med	high
Log storage	15	4	3	2	2	low	low	low	low
Log transport	5	4	0	2	2	low	low	low	low
Logs in repository	indef	4	4	3	2	low	low	low	low
Current LWRs with Full-MOX Cores ^b									
Pit storage	25	1	3	3	3-4	high	med	med	med
Pit transport	25	1	0	2	3	high	high	high	med
Oxide production	25	1	2	2	2-4	high	high	med	high
Oxide storage	25	1	3	2	2-4	high	med	med	med
MOX production	25	1→2-	2	2	2-4	high	high	med	high
MOX storage	25	2-	3	2	2-4	high	med	med	med
Fuel assembly production	25	2→2+	2	2	2-4	med	med	med	high
Fuel assembly storage	25	2+	3	2	2-4	med	med	med	low
Fuel assembly transport	25	2+	0	1	2	med	med	high	low
Fuel storage at reactor	25	2+	2	2	2-3	med	med	med	low
Irradiation in reactor	28	2+→4	1	4	2-3	low	low	low	low
Spent fuel at reactor	33	4	3	2	2-3	low	low	low	low
Spent fuel transport	18	4	0	2	2	low	low	low	low
Spent fuel in repository	indef	4	4	3	2	low	low	low	low

NOTES: *Duration* (years) extends from the time that operations in a given step begin to the time that operations of this type cease. *Overall intrinsic barriers* (0 lowest to 4 highest) are taken from the last column of Table 6-6. *Implementation-dependent barriers* (0 lowest to 4 highest)—in the categories *location/exposure*, *containment*, and *institutional* are rated on the basis described in Chapter 3 (see section “Specific Security Concerns and Threat Characteristics”). *Vulnerability* assessments are based on the interaction of the barriers with the different types of threats, as discussed in Chapter 3.

^a*Vitrification*: It is assumed that the repository is ready in 2015 and that the process of shipment of the logs to, and their emplacement in, the repository extends over a period of five years.

^b*Current LWRs with Full-MOX Cores*: It is assumed that there is only one MOX fabrication plant. Repository becomes available in 2015. Spent fuel assumed stored at reactor for five years cooling if repository is available.

In general, the same types of threats and considerations would apply in the Russian case as in the U.S. case, but concerns about the possibility of covert theft during processing and forcible theft during transport would appear to be even more significant. Thus the premium on minimizing processing steps involving high accessibility and low accountability, and on minimizing transport steps, would be even more substantial than in the U.S. case. As noted in the U.S. case, however, this does not appear to be a strong discriminant between the various options considered in this report: while it appears that the overall in-process vulnerability of the vitrification option would be somewhat lower than the LWR MOX option, the attractiveness of vitrified material for weapons purposes would be somewhat higher, making the two options roughly comparable in overall risk.

The panel concurs with the full committee that U.S.-Russian cooperation to improve material protection, control, and accounting in Russia should be an urgent priority accorded a significantly higher level of funding than is currently the case, and notes the specific recommendations made by the parent committee in this regard (NAS 1994, pp. 133–136). Specifically for the case of disposition of excess WPu, the panel concurs with the committee's recommendation that "The United States and Russia should begin discussions with the aim of agreeing that whatever disposition options are chosen, an agreed, stringent standard of accounting, monitoring, and security will be maintained throughout the process—coming as close as practicable to meeting the standard of security and accounting applied to intact nuclear weapons" (NAS 1994, p. 227). Without assurance that such strict standards of security and accounting will be maintained, there is a very real risk that disposition of WPu, with the substantial plutonium processing and possibly transport it entails, could add to the risk of proliferation rather than reducing it.

The panel notes that unlike in the U.S. case, disposition of excess WPu in Russia will take place in a context of a nuclear economy involving substantial civilian plutonium separation and government plans for a large-scale plutonium economy, which also poses proliferation risks. It is unlikely to be possible to "delink" the WPu issue completely from the civilian plutonium issue in Russia. For example, in the U.S. case, a MOX plant that might be constructed or completed for the WPu disposition mission would presumably be closed when that mission was completed (unless U.S. fuel-cycle policy changed in the interim). In the Russian case, by contrast, such a facility would presumably continue to operate, fabricating reprocessed civilian plutonium into fuel, and thereby serving as a key element in the next stage of the plutonium economy in Russia. U.S. policy-makers, when making decisions on steps that could influence Russian plutonium disposition choices (such as provision of assistance for particular options, for example) will have to take into consideration these linkages to the civilian plutonium economy and their relation to U.S. fuel-cycle policy and nonproliferation goals.

ECONOMIC COMPARISONS

As noted in Chapter 3, economic considerations are less important than security in reaching a conclusion about the relative attractiveness of alternative options for WPu disposition. But a study of costs is nonetheless worthwhile, both to assist in ranking options that are not distinguishable on security grounds and to facilitate planning for the investments that will be required for one option or another. In what follows, we apply the economic assessment methods and assumptions introduced in Chapter 3 to illuminate the comparative costs of the main disposition options.

We consider, in turn, the costs of incorporating WPu into reactor fuel, the costs of using such fuel in currently operating reactors, the costs of using it in reactors now partly completed (which could be completed for the plutonium disposition mission), the costs of building new reactors of evolutionary and advanced types and using them to process WPu-bearing fuel, and the costs of vitrifying the plutonium with defense HLW. In these comparisons, all of the final cost estimates are stated in 1992 dollars.

The economic analyses presented here relate mainly to prospective disposition operations in the United States. One case is considered in which U.S. plutonium would be used in Canadian reactors. A subsection at the end discusses how the economic considerations would differ for the disposition of Russian WPu.

Weapons Plutonium Versus Uranium as Power Reactor Fuel

Considerable discussion and controversy has surrounded the question of whether the use of surplus WPu as fuel for power reactors would save money or cost money, compared to the costs of generating the same amounts of electricity by the means that would otherwise be used—for example, in the case of currently operating reactors, using the same reactors but with U-235 as the primary fissile material. In this section, we seek to clarify this issue by (1) identifying the component costs that bear on the comparative economics of plutonium-based and uranium-based fuels, (2) indicating and explaining the ranges of values for these component costs that have appeared in the literature, (3) offering our own estimates of the values that are most appropriate to the case at hand (i.e., to investigation of the economics of the disposition of WPu over the next 20–40 years), and (4) using these estimates to obtain a plausible range for the total costs (or savings) associated with disposition of WPu through its use as fuel in power reactors.

We take, as the starting point for our treatment, the proposition that surplus WPu would be made available to the power-generating entity at no charge, in the form of plutonium metal. That is, the costs invested in the production of the plutonium in the course of the weapons programs of the United States and the

former Soviet Union are to be treated as “sunk” costs, but any costs entailed in converting the plutonium from the metallic form (in which it emerges from surplus weapons) to oxide or other forms (required for fueling some types of reactors) are to be charged to electricity generation.⁹ Our assumption that the plutonium metal is free, of course, means that the resulting estimates of the costs of the use of WPu for electricity generation will *not* be relevant to the case of recycle of plutonium from spent reactor fuel; this would be more expensive because of the costs of reprocessing spent fuel to extract the plutonium.

In order to compare the costs of electricity generation with WPu-based versus uranium-based fuels, it is not necessary to estimate all of the generation costs; only those costs that are related to the character of the fuel need be estimated and compared, per kilowatt-hour (kWh) and, in the case of the plutonium-based fuel, per kilogram of WPu. From these figures it will be possible to calculate the incremental cost (or savings) per kilowatt-hour associated with using WPu for electricity generation, as well as the incremental cost (or savings) associated with the disposition in this way of 50 tons of WPu (or 100 tons, or any other quantity). Where these calculations require the choice of a value for the real cost of money, we shall use the figure of 7 percent per year ($r = 0.07$) specified by the U.S. Office of Management and Budget for the evaluation of government projects that produce effects in the private sector or have any interaction with it (since clearly electricity generation does this).¹⁰ Because we are concerned with the costs of plutonium disposition to society rather than the costs to any particular entity, the question of whether particular fuel-related costs are incurred by the federal government or by another entity will concern us only insofar as it affects the magnitude of these costs (as, for example, through the circumstance that a private entity may have to pay for property taxes and insurance while a government entity does not).

We first consider, for specificity, a comparison between the use of conventional LEU oxide fuel in a PWR and the use of MOX fuel of comparable reactivity in the same reactor with the same burnup—i.e., same irradiation in thermal megawatt-days (MWd) per metric ton of heavy metal (MTHM, based on the combined mass of uranium and plutonium in fresh fuel). Since a kilogram of heavy metal (kgHM) in fuel, whether LEU or MOX, then generates the same amount of electricity, it becomes both germane and convenient to compare the costs of the two fuels on a per-kgHM basis. We take up the generalization to different fuel parameters and different reactor types subsequently.

We take the “baseline” LEU fuel to have an initial U-235 enrichment of 4.4 percent and assume it is irradiated on a three-year fuel cycle, at 75-percent ca-

⁹ For purposes of comparative assessment it is necessary to account for the costs of any conversions of the WPu from metallic form, because different disposition schemes require different conversions (and some require none at all). See “Issues and Criteria in Economic Evaluation of Alternatives” in Chapter 3.

¹⁰ See Chapter 3 and OMB (1992).

capacity factor, to 40,000 MWd/MTHM—typical commercial practice in a modern PWR. We assume that the MOX is made by mixing WPu oxide with depleted uranium oxide (the uranium which contains 0.25 percent U-235) to an initial concentration of 4.8 percent WPu in heavy metal; this initial WPu concentration gives the same end-of-cycle reactivity (i.e., at 40,000 MWd/MTHM) as for the LEU fuel, arguably the most appropriate basis for comparison.¹¹ To compare the costs of these equivalently energy-generating fuels, we then need to estimate, on a comparable basis:

For the LEU fuel, the costs of

- finding, mining, and milling the uranium ore;
- converting the resulting uranium-ore concentrate (“yellowcake,” typically 60–85 percent U_3O_8) to uranium hexafluoride (UF_6);
- enriching the UF_6 to the indicated U-235 concentration,
- converting the enriched UF_6 to UO_2 ;
- fabricating the UO_2 into fuel pellets, fuel rods, and fuel assemblies;
- all the storage and transport steps associated with this chain, including delivery of the fuel assemblies to the reactor and their storage there until they are loaded into the core; and
- the costs associated with ultimate disposition of the spent fuel after it leaves the nuclear reactor.¹²

For the MOX fuel, the costs of

- conversion of WPu metal to PuO_2 ;
- acquisition of depleted uranium and its conversion, if necessary, to UO_2 ;
- mixing the oxides and fabricating the MOX into fuel pellets, fuel rods, and fuel assemblies;
- all the storage and transport steps associated with this chain, including delivery of the fuel assemblies to the reactor and their storage there until they are loaded into the core; and
- the costs associated with ultimate disposition of the spent fuel after it leaves the nuclear reactor.

¹¹ LEU and MOX fuels have somewhat different rates of change of reactivity with burnup. The closest match in nuclear performance is obtained by matching the end-of-life reactivities (see, e.g., Battelle 1993).

¹² It is appropriate to include this category explicitly both because these costs conceivably could differ, in some cases, between plutonium-based and uranium-based fuels, and also to remind ourselves, even where these costs do not differ, that a proper comparison must either include them for both fuel types or exclude them for both. (Comparisons in the literature contain some inconsistencies in this regard.) By convention, the costs of interim storage of spent fuel at the reactor, before its removal for ultimate disposition, are considered part of reactor costs rather than as part of fuel costs.

Many studies of these matters have omitted some of these cost categories, above all the costs of conversion of plutonium metal to oxide and the costs of storage and transport. Since these costs will be quite different for LEU and MOX fuels (because of the much larger radiological and security hazards of the latter), and since they also may differ significantly among different schemes that use MOX (e.g., because of different types and distances of transport), their omission can significantly distort comparisons.

We now turn to an activity-by-activity examination of the costs associated with the use of LEU and MOX fuels, beginning with LEU.

LEU: Composition of the Total Costs

The total costs per kilogram of heavy metal in fresh fuel¹³ consist of the sum of terms obtained by multiplying, for each step in the fuel production chain, the unit cost for the step (dollars per unit of activity in the step) times the number of units of activity in the step associated with one kilogram of heavy metal (kgHM) in fresh fuel. Thus we have, for LEU

$$\begin{aligned} \text{LEU fuel cost (\$/kgHM)} = & \{ \text{unit cost of uranium acquisition [\$/(kgU acquired)]} \\ & \times \text{quantity of U acquired to make 1 kgHM in fresh fuel [kgU/kgHM]} \} \\ + & \{ \text{unit cost of uranium conversion [\$/(kgU converted)]} \\ & \times \text{quantity of U converted to make 1 kgHM in fresh fuel [kgU/kgHM]} \} \\ + & \{ \text{unit cost of uranium enrichment [\$/(separative work unit; SWU)]} \\ & \times \text{quantity of U enrichment to make 1 kgHM in fresh fuel} \\ & \quad \text{[SWU/kgHM]} \} \\ + & \{ \text{unit cost of LEU fuel fabrication [\$ /kgHM]} \} \\ + & \{ \text{unit cost of ultimate disposition [\$ /kgHM]} \}. \end{aligned}$$

The cost of acquisition of uranium, as represented by the selling price of uranium-ore concentrate (typically reported as dollars per pound of U_3O_8 or as dollars per kilogram of contained uranium), includes the costs of finding, mining, and milling the uranium ore. The cost of conversion of the UF_6 to UO_2 following enrichment is, by convention, considered to be part of the cost of fuel fabrication. Costs of transport and storage of the uranium between and at the steps up to the fuel's arrival at the reactor are small compared to the other costs and are assumed to be included with them (e.g., cost of shipping to the conversion plant is included with the uranium acquisition cost, and cost of storage at the conversion plant is included with the cost of conversion). The cost of storing

¹³ It is customary to normalize all nuclear fuel-cycle costs to the quantity of heavy metal present in the fuel at the point it is loaded into the reactor. Thus, for example, the costs of spent fuel transport and waste management, per kilogram of heavy metal (kgHM), relate to the amount of fresh fuel that contained one kilogram of combined uranium and plutonium metal, not to the slightly larger amount of fuel that would contain 1 kgHM after some of the uranium and plutonium have been consumed by fission (see, e.g., OECD 1992).

fresh fuel at the reactor prior to loading it into the core is likewise small and is customarily treated as part of the reactor capital and operating charges rather than as part of fuel costs.

Determination of the activity quantities at each step requires attention to the material balance and separative work requirement for uranium enrichment, as well as to the fractional losses of uranium in other steps. The relevant enrichment relations are summarized in "Uranium and Separative Work Requirement for Enrichment" on p. 288. For our reference case values of 4.4 percent U-235 enrichment and 0.25 percent U-235 content in the enrichment tails, the uranium feed requirement is 8.83 kgU input per kgU in enriched product, and the separative work requirement is 6.59 SWU per kgU in enriched product. We follow the recent OECD fuel-cycle study (OECD 1992) in estimating other uranium losses in processing to be 0.5 percent in conversion to UF₆, 1.0 percent in fuel fabrication, and negligible at other steps. Accounting for these losses, the enrichment requirement per kilogram of heavy metal in fresh fuel is

$$\begin{aligned} SW \text{ [SWU/kgHM]} &= 6.59 \text{ [SWU/(kgU enriched product)]} \\ &\times 1.01 \text{ [(kgU in enriched product)/(kgHM in fresh fuel)]} \\ &= 6.66 \text{ SWU/kgHM,} \end{aligned}$$

and the uranium feed requirement per kilogram of heavy metal in fresh fuel is

$$\begin{aligned} F \text{ [kgU/kgHM]} &= 8.83 \text{ [(kgU into enrichment)/(kgU in enriched product)]} \\ &\times 1.005 \text{ [(kgU into conversion)/(kgU into enrichment)]} \\ &\times 1.01 \text{ [(kgU in enriched product)/(kgHM in fresh fuel)]} \\ &= 8.96 \text{ [(kgU into conversion)/kgHM] .} \end{aligned}$$

It follows that the cost of LEU fuel under our reference case conditions can be written as

$$\text{LEU cost [$/kgHM]} = 8.96 \times (\text{UCOST} + \text{CNVSN}) + 6.66 \times \text{ENRCH} + \text{FABRN} + \text{DSPSL} ,$$

where UCOST is the cost of uranium per kilogram uranium delivered to the conversion process, CNVSN is the cost of conversion per kilogram uranium delivered to the conversion process, ENRCH is the cost per separative work unit, FABRN is the cost of fabrication per kilogram of heavy metal in the fresh fuel, and DSPSL is the cost of spent fuel storage and disposition per kilogram of heavy metal in the fresh fuel.

The range of values of the first four of these unit costs (that is, excluding disposal costs) from current experience and recent studies is shown in Table 6-8. Taking the lowest values for all of the unit costs in the table would yield a total cost of LEU fuel, not including costs of spent fuel storage and disposal, of \$813/kgHM; the corresponding figure based on taking the highest values in the table is \$2,167/kgHM. This is a large difference. In what follows, we examine the individual costs to try to narrow the range that can be considered plausible

TABLE 6-8 LEU Unit Costs from Current Experience and Recent Studies

Figures have been converted to 1992 U.S. dollars and rounded to the nearest dollar.

Source	UCOST (\$/kgU)	CNVSN (\$/kgU)	ENRCH (\$/SWU)	FABRN (\$/kgHM)
Mid-1993 spot-market price ^a	17	3	66	NA
Mid-1993 long-term contract price ^a	25-43	4-6	87-107	194-291
ERI (1993) range for year 2000 ^b	35-55	6	92-114	194 ^c
USDOE (1993a) Plutonium Disposition Study ^d	65	10	125	260
OECD (1992) study, range ^c	41-92	6-11	82-133	205-358
OECD (1992) study, reference case	71.7	8	113	282
Berkhout et al. (1992) best estimate	40	7	100	200
This report	55±20	9±1	95±15	200±30

NOTE: NA indicates not applicable.

^a USDOE (1993c).

^b Energy Resources International, "Nuclear Fuel Cycle Supply and Price Report," as reported in *Nuclear Fuel*, June 9, 1993 (ERI 1993). The Office of the Assistant Secretary for Nuclear Energy (USDOE 1993c) provided this reference in response to a request from the National Academy of Sciences' Committee on International Security and Arms Control for that Office's judgment of the likely future prices of these commodities and services.

^c ERI (1993) gives a figure of \$200/kgU (1993 dollars) for PWR fuel and \$300/kgU (1993 dollars) for BWR fuel. For this report we use the former figure since the other parameters of our calculations relate to PWRs.

^d Assumptions specified by the U.S. Department of Energy for its 1992-1993 Plutonium Disposition Study (USDOE 1993a), apparently based on Delene and Hudson (1993). DOE now concedes that the indicated values for the costs of uranium and its conversion and enrichment are "high compared to today's market and sensitivity studies" (USDOE 1993c, p. 4).

^e The values presented in the Organisation for Economic Co-Operation and Development study (OECD 1992) represent levelized costs during a presumed period of operation extending from the year 2007 to 2035.

for the circumstances of interest here, namely, disposition of WPU in the period extending roughly from the year 2000 to 2030.

LEU: Uranium Acquisition

Estimates of world uranium reserves and resources versus expected selling price, when compared with plausible demands to 2015 or even 2030, give little reason to expect substantial price increases during this period above the range of current long-term contract prices.¹⁴ That was true even before the advent of

¹⁴ For recent estimates of world uranium resources see, e.g., OECD (1990). For comparisons with growth rates of demand, see ERI (1993). It is worth noting that DOE's own 1993 "Cost Estimate Guidelines for Advanced Nuclear Power Technologies" (Delene and Hudson 1993) call for assuming no real escalation of uranium-ore costs above \$25 per pound of U₃O₈ (\$65/kgU) over the 30-year operating lifetimes of plants that would begin operation between 2000 and 2010. The spot-market price of U₃O₈ in early 1995, as this report went to review, was about \$10 per pound of U₃O₈ (\$26/kgU).

agreements that, if implemented, will bring onto the civilian-power uranium market substantial quantities of LEU obtained by blending down surplus highly enriched uranium (HEU) from dismantled nuclear weapons. The Organisation for Economic Co-operation and Development study (OECD 1992) estimated the rate of increase of the real cost of uranium acquisition in the period of interest at 1.2 percent per year, with a range from -0.8 percent per year to 2.1 percent per year. We assume, perhaps a bit conservatively, a rate of growth of real uranium costs after the year 2000 of 0–2 percent per year; if we take the year 2015 as the reference point for an economic evaluation relevant to the disposition of WPu, and if we escalate the Energy Resources International (ERI 1993) low estimate for the year 2000 at 0 percent per year and their high estimate for that year at 2 percent per year, our range of values for the year 2015 becomes \$35–\$75/kgU (1992 dollars).

LEU: Conversion

The figures for conversion are neither controversial nor very important in determining the total costs of LEU fuel. We take the range of plausible values for the midpoint of our period of interest to be \$8–\$10/kgU.

LEU: Enrichment

As noted in a number of the recent studies (see, e.g., OECD 1992, USDOE 1993c), there exists a worldwide surplus of uranium enrichment capacity and no foreseeable prospect for its disappearance. The evolution of enrichment technologies in the directions already evident, moreover, will tend to lower the costs per separative work unit (OECD 1992). In these circumstances, we think it is difficult to justify an upper-limit estimate for the 2015 enrichment higher than the upper end of the range of long-term contract prices in 1993 (\$107/SWU), and that the lower end of the range must be at least at the OECD low estimate for the period 2007–2035 (\$82/SWU); we therefore take the range to be \$80–\$110/SWU (1992 dollars).

LEU: Fabrication

Much of the difference in estimates of the costs of LEU fuel fabrication per kilogram of heavy metal appears to be due to the difference between PWR fuel and boiling-water reactor (BWR) fuel. We see little reason to diverge, in our comparison based on the case of a PWR, from the ERI (1993) estimate of about \$200/kgHM for PWR fuel, (which the *Nuclear Fuel* analysis (ERI 1993) predicts to remain unchanged to the end of their forecast period of 2005). The technology of UO₂ fuel fabrication is mature and its occupational health and environmental impacts are small, so it is difficult to see what could drive up its cost

other than, conceivably, a temporary shortage of capacity that would quickly be remedied. We take the range to be \$200/kgHM \pm 15 percent, hence \$170–\$230/kgHM (1992 dollars).

LEU: Spent Fuel Storage and Waste Disposal

As noted earlier, it is customary to count costs of spent fuel storage at the reactor as part of reactor costs while including an estimate of the costs of ultimate disposal of the spent fuel in the fuel costs. Since the exact means of ultimate disposal has not been decided, this item poses some difficulty. In U.S. practice, electric utilities pay DOE a “waste-disposal fee” of \$0.001 per electrical kilowatt-hour (kWh) generated. For the parameters of our example, this is equivalent to about \$260/kgHM, evaluated as of delivery to the reactor.¹⁵ This figure, it must be emphasized, reflects not real-world experience about the ultimate cost of disposing of spent fuel as waste but, rather, DOE’s rough estimate of how much money needs to be collected at the time of generation in order to provide a waste-management fund that will be adequate to the task at the future time when disposal actually takes place. The uncertainty in such an estimate is necessarily quite large; we round the figure to \$300/kgHM and assign a judgmental 70-percent confidence interval of a multiplicative factor of two up and down, giving a range of \$150–\$600/kgHM. This range is roughly consistent with the \$200–\$1,000/kgHM range of disposal costs cited in the OECD (1992) study.

¹⁵ We assume that electric utilities pay the \$0.001/kWh waste-disposal assessment annually based on the year’s electricity generation. To translate this operating cost into an equivalent contribution to the cost of fuel per kilogram of heavy metal, one must take into account both the number of electrical kilowatt-hours generated from 1 kgHM (which for our nominal PWR and 40,000-MWd/kgHM burnup is 40 MWd/kgHM \times 0.316 MW-electric/MW-thermal \times 1,000 kW/MW \times 24 hr/d = 303,000 kWh) and the way in which electric utilities calculate carrying charges on nuclear fuel. Without accounting for carrying charges, the waste-disposal assessment would be equivalent to \$0.001/kWh \times 303,000 kWh/kgHM = \$303/kgHM. Carrying charges on fuel are customarily calculated for a period one year longer than the fuel’s residence time in the core, hence four years for a three-year residence time. Thus, in the translation between acquisition cost for the fuel per kilogram of heavy metal and the equivalent per-kilowatt-hour cost, the acquisition cost is multiplied by the annualized capital recovery factor—here $0.07 \times 1.07^4 / (1.07^4 - 1) = 0.295/\text{yr}$ for 0.07/yr real cost of money and four-year accounting lifetime. So the fuel’s acquisition cost is translated into four annual payments, each equal to 0.295 of the purchase cost, and since each of these payments can be associated with one-fourth of the fuel’s lifetime electricity output, the carrying-charge factor by which the per-kilowatt-hour cost exceeds the ratio of the purchase cost to the total electrical output is $0.295/0.250 = 1.18$. Thus, for the assumed circumstances, the purchase-cost equivalent of \$0.001/kWh is not \$303/kgHM but $\$303/\text{kgHM} / 1.18 = \$257/\text{kgHM}$.

URANIUM AND SEPARATIVE WORK REQUIREMENTS FOR ENRICHMENT

Performing a material balance for enrichment shows that the quantity of uranium "feed" (F) needed per kilogram of enriched uranium "product" is given by

$$F [(kgU \text{ input}) / (kgU \text{ in enriched output})] = (X_p - X_t) / (X_f - X_t),$$

where X_p is the U-235 fraction in the enriched product, X_t is the U-235 fraction in the waste ("tails"), and X_f is the U-235 fraction in the feed material. If the feed material is natural uranium, $X_f = 0.0072$. The separative work requirement per kilogram of uranium in the enriched product can be shown to be given by

$$SW [(separative \text{ work units}) / (kgU \text{ in enriched output})] = V(X_p) + (F - 1) \times V(X_t) - F \times V(X_f),$$

where $V(X)$, the dimensionless "separation potential," is $(2X-1)\ln[X / (1-X)]$.

The U-235 fraction left in the tails, X_t , influences both the separative work requirement (which decreases as X_t increases) and the uranium feed requirement (which increases as X_t increases). Its value is chosen to minimize the total cost of producing a kilogram of enriched uranium, and so depends on the relative costs of enrichment work and uranium feed. Under recent and current conditions, X_t has generally been chosen to be 0.0025 or 0.0030. We follow several

Although it often has been assumed that ultimate disposition costs would be about the same for MOX fuel as for LEU fuel, there is at least some basis for supposing that they might be higher. As noted in a recent study by analysts at the German nuclear research center in Karlsruhe (Kessler et al. 1992), spent MOX fuel has a higher thermal power, decaying more slowly, compared to LEU fuel irradiated to the same exposure; the German analysts estimate that this would reduce by about a factor of two the number of fuel elements that could be packaged in each of the containers envisioned for use in the German waste repository, and would increase by a similar factor the repository floor-space requirements for MOX fuel. Also, spent MOX fuel from a PWR would contain 2–3 percent residual plutonium versus about 1 percent in typical spent LEU fuel, and the associated criticality considerations may likewise dictate a lower packing density in the repository.

recent studies in assuming a value of 0.0025, although at current low prices of uranium a modest savings—about \$30/kgHM for 4.4 percent enriched fuel—would be associated with using $X_t = 0.0030$ instead. The requirements for uranium feed and separative work per kilogram of uranium in the enriched product, for tails fractions of 0.0025 and 0.0030 and various enrichment levels, are shown in the following table.

Enrichment (% U-235)	Uranium Feed (kgU input per kgU in enriched out- put)		Separative Work (SWU per kgU in enriched output)	
	$X_t = 0.0025$	$X_t = 0.0030$	$X_t = 0.0025$	$X_t = 0.0030$
3.0	5.85	6.43	3.77	3.38
3.3	6.49	7.14	4.36	3.92
3.7	7.34	8.10	5.16	4.66
4.0	7.98	8.81	5.77	5.22
4.4	8.83	9.76	6.59	5.98
5.0	10.11	11.19	7.84	7.13
5.5	11.17	12.38	8.90	8.10

NOTES: The substantial simplification of ignoring the 0.006 percent U-234 in natural uranium produces only an insignificant error in the foregoing calculations.

The separative work unit (SWU), although called "work," is a linear combination of the quantities of material flowing into and out of the enrichment process and thus, by convention, has units of kilograms. The number of SWUs needed is a nonlinear function of the product enrichment desired, the enrichment of the starting material, and the enrichment of the discarded "tails."

Because canister design and repository layout are still far from settled, and because there might be relatively inexpensive ways to accommodate the higher decay power and greater reactivity of MOX fuel compared to LEU, we cannot say much more than that the costs of ultimate disposition of MOX are likely to fall in a range extending from about the same as those for LEU to about twice as much. If the best estimate were 50 percent greater, the \$300/kgHM figure for LEU given above would imply a penalty associated with MOX of an additional \$150/kgHM. Because the uncertainties about ultimate disposition costs are so large, however, we choose not to add a figure of such questionable validity to the better-defined numbers for other elements of fuel costs. Instead we will present figures for LEU and MOX fuel costs *less* the costs of ultimate disposal, underlining here that if there is a difference between MOX and LEU in the omitted disposal costs it will be in favor of LEU.

LEU: Reactor Panel Best Estimate of Total Fuel Costs Less Disposal

The preceding considerations have led us to the following plausible ranges of unit costs relevant to the determination of total fuel costs less costs of ultimate disposal, evaluated in 1992 dollars for operations in the period 2000–2030: UCOST = \$55 ± \$20, CNVSN = \$9 ± \$1, ENRCH = \$95 ± \$15, FABRN = \$200 ± \$30. Using the uranium requirements and enrichment quantities that correspond to 4.4-percent enriched fuel and enrichment tails of 0.0025 percent U-235, with losses of 0.5 percent in conversion and 1.0 percent in fabrication, then gives a central estimate of \$1,406/kgHM. If it is assumed, for convenience, that the indicated uncertainties represent standard deviations of independent random variables, then the variance of the sum is the sum of the variances, and the standard deviation is the square root of this sum, or \$207/kgHM. Since the precision in these figures is clearly illusory, we round off the range to \$1,400 ± \$200 per kgHM.

MOX: Composition of the Total Costs

The equation for the total costs per kilogram of heavy metal in fresh MOX fuel is, in analogy to the one given above for LEU,

$$\begin{aligned}
 &\text{MOX fuel cost (\$/kgHM)} \\
 &= \{ \text{unit cost of uranium acquisition [\$/(kgU acquired)]} \\
 &\quad \times \text{quantity of U acquired to make 1 kgHM in fresh fuel [kgU/kgHM]} \} \\
 &+ \{ \text{unit cost of uranium conversion to UO}_2 \text{ [\$/(kgU converted)]} \\
 &\quad \times \text{quantity of U converted to make 1 kgHM in fresh fuel [kgU/kgHM]} \} \\
 &+ \{ \text{unit cost of plutonium conversion to PuO}_2 \text{ [\$/(gPu converted)]} \\
 &\quad \times \text{quantity of Pu converted to make 1 kgHM in fresh fuel [gPu/kgHM]} \} \\
 &+ \{ \text{unit cost of MOX fuel fabrication [\$/kgHM]} \} \\
 &+ \{ \text{unit cost of ultimate disposition [\$/kgHM]} \} ,
 \end{aligned}$$

plus any costs of plutonium storage and transport within the fuel cycle that are not already included in the preceding figures.

The concentration of WPu in our reference-case MOX fuel is 4.8 percent by weight, hence 48 g/kgHM, so the uranium requirement is 0.95 kgU/kgHM. If uranium losses in conversion and fabrication totaled 1.5 percent, as assumed for LEU fuel, the requirement would be about 0.97 kgU/kgHM. The OECD study (1992) assumed the same percentages for plutonium losses in conversion and fabrication as for uranium. This seems to us to be an overestimate, in light of the extra incentives for tight material control over plutonium that arise from its extra ES&H and security dangers; but since changing this estimate would affect the economic conclusions by an amount very small compared to other uncertainties in the calculation, we do not take the trouble to develop an alternative figure. Thus the plutonium requirement is $48 \times 1.015 = 49$ gPu/kgHM.

MOX: Uranium Acquisition and Conversion

We assume this uranium will be in the form of depleted uranium (0.25 percent U-235), which is and will remain in abundant supply in the nuclear-energy industry. The combined cost of acquiring depleted uranium and converting it to UO_2 can hardly be more than \$10/kgHM, and any error in this estimate cannot be important since it will assuredly be small compared to uncertainties in the other components of MOX fuel costs.

MOX: Conversion of Plutonium Metal to PuO_2

As noted above, it is appropriate to assign to the MOX fuel costs a contribution for conversion of plutonium metal to oxide, because some alternative schemes for disposition of the WPu—against which the MOX fuel option must be compared—would not incur this cost.

A 1992 study by DOE (USDOE 1992) estimated that conversion of 100 tons of Russian WPu to oxide over a period of 10 years, in Russia or France, would entail a facility construction cost of \$150 million and operating costs of \$10 million per year. If we assume that the costs of such an operation in the United States would be 50 percent higher, this would mean capital costs of \$225 million and operating costs of \$15 million per year (assumed to be 1992 dollars). For a conversion rate of 10 MT/yr for 10 years, and with $r = 0.07$, these figures translate on a levelized-annualized basis to a conversion cost of \$4.70/gPu. If we apply the widely used rule of thumb that the construction costs and operating costs for such facilities increase as the 0.6 power of output (meaning unit costs will decrease as the 0.4 power of output), the unit cost for converting 50 tons of plutonium to oxide in 10 years would be 32 percent higher, or \$6.20/gPu.

A 1993 study of the possibility of vitrifying the WPu in radioactive-waste-bearing glass (McKibben et al. 1993) estimated the upstream costs (previtrification costs) of a campaign treating 50 tons of WPu as \$400 million. Conversion to oxide is the most substantial of the “upstream” steps. If we assume it accounts for 75 percent of the indicated cost, or \$300 million, and that two-thirds of this is the facility cost and one-third represents 10 years of operating costs at 5 tons of plutonium per year, then with $r = 0.07$ the costs on a levelized-annualized basis would be \$7.70/gPu.

Based on the preceding two calculations, and allowing some extra leeway for the approximateness of the estimates on which the calculations are based, we shall take as our estimate of the costs of conversion of plutonium metal to oxide as $\$7 \pm \$2/\text{Pu}$ (1992 dollars, in U.S. facilities, based on a campaign treating a total of 50 tons of WPu).¹⁶

¹⁶ Two more recent examinations of the cost of plutonium conversion (which became available after the panel's economic calculations were largely complete) presented cost estimates that were in one

MOX: Fuel Fabrication

The cost of fabrication of MOX fuel is by far the largest component of the cost of MOX fuel based on cost-free plutonium metal from surplus weapons, and it is also the component for which the widest range of values can be found in the literature. The breadth of this range is attributable in part to the early stage of commercialization of MOX fabrication technology (whereby only a few plants have operated, and these are of low capacity compared to those envisioned for a mature industry) and in part to the difference in the scale of operation between the scale appropriate to a major commitment to MOX use in civilian power generation and the scale needed for a limited campaign for the disposition of 50 or 100 tons of WPu. The range of cost estimates is also due in part to differences in comprehensiveness, accounting conventions, and conservatism—as discussed in general terms in Chapter 3 (see section “Issues and Criteria in Economic Evaluation of Alternatives”)—employed in the development of the estimates.

Table 6-9 summarizes the results of our attempt to put on a somewhat more consistent basis a variety of the most recent (1992 and 1993) estimates of MOX fabrication costs found in the literature. The figures shown in the table have all been converted to 1992 dollars and reflect, insofar as has been possible given the information provided in the references, a common set of assumptions about contingency factors, real cost of money, construction time and interest during construction, treatment of dismantling and disposal costs, and other accounting conventions. (See the notes to Table 6-9 for details and “Issues and Criteria in Economic Evaluation of Alternatives” in Chapter 3 for background.) The range of unit costs shown in the table can be further reduced by adjusting for scale: if we apply the 0.6 power law mentioned above for the scaling of cost with output to determine, for each estimate in the table, the corresponding equivalent unit cost at 100 MTHM/yr output, the central values of the estimates for the 10 cases would be (in order of the cases in Table 6-9): \$1,094, \$1,438, \$1,272, \$1,486, \$1,413, \$1,340, \$1,500, \$1,574, \$1,757, and \$2,240.

The first of these figures is the GE estimate prepared for the U.S. Department of Energy’s Plutonium Disposition Study (PDS); comparison with the other estimates suggests that it is unrealistically low. The third figure, prepared

case higher, and in the other case lower, than the panel’s estimate, suggesting that the general range of the panel’s estimate is correct. A study prepared by the Lawrence Livermore National Laboratory in support of DOE’s plutonium disposition effort estimated the capital and operations costs of facilities designed to process 50 tons of plutonium in the form of pits to oxide suitable for fabrication into fuel in 20 years at \$360 million and \$23 million per year (in the cheapest case) to \$664 million and \$73 million per year (in the most expensive case) (Walter 1994). By contrast, an examination of the problem performed at Los Alamos, focusing on a somewhat different technique for carrying out the conversion, estimated that the capital and operations costs of a facility capable of processing nearly twice as many pits per year would be \$100 million and \$10 million per year (Toevs and Trapp 1994). (Both estimates appear to use 1994 dollars.)

for the PDS by Westinghouse, must similarly be regarded as low when one recalls that, unlike all the other estimates, it is said to include plutonium metal-to-oxide conversion. The last and highest figure in the list relates to a small European plant of outdated design and cannot be considered very relevant for our purposes. With the omission of these "outliers," the average of the remaining seven estimates is \$1,500/kgHM. Inspection of the contributing factors to this total (see the notes to Table 6-9) shows that the difference between paying and not paying property taxes and insurance on the fuel fabrication facility amounts to about \$150/kgHM. Hence we take, as our central estimates of fuel fabrication costs in new facilities at a scale of 100 MTHM/yr, figures of \$1,425/kgHM if property taxes and insurance are not paid and \$1,575/kgHM if these are paid. Our judgmental 70-percent confidence intervals extend \pm \$300/kgHM from these figures.

Before such estimates can be used in an economic assessment of a program for the disposition of WPu, however, they must be further adjusted to account more fully and consistently for "preoperational costs" such as research and development and safety and environmental analysis and licensing. These preoperational costs are often excluded or understated in cost estimates for a commercial operation (typically on the assumption that they are already "sunk" costs, or because it is assumed that they have been or will be covered by government); but, in assessing the costs to society of a WPu disposition program in a situation in which MOX fuel would not otherwise be being fabricated, the preoperational costs must be assigned to the disposition program.

In the contractor studies of MOX fuel-based disposition schemes performed in the U.S. Department of Energy's PDS, the estimates of preoperational costs varied quite widely. Some of this variation appears to be due to differing conventions by which various types of preoperational costs were lumped with or separated from capital costs. As best we can tell, fuel fabrication plant preoperational costs in the contractor-analyzed cases closest in scale to our 100-MTHM/yr reference level, were \$39 million (GE 1993, p. 10.5), \$45 million (ABB-CE 1993, p. VI-14), and \$68 million (Westinghouse 1993, p. 3-3); as percentages of the sum of direct plus indirect construction costs for the indicated fuel fabrication plants, these figures translate to 11.6, 10.0, and 17.5 percent, respectively. The Technical Review Committee of the PDS criticized all of the contractor studies for understating preoperational costs (USDOE 1993a, p. SC6-5). We therefore estimate the preoperational costs for a 100-MTHM/yr fuel fabrication plant as a 20-percent addition to overnight construction costs (direct + indirect + contingency), hence about \$120 \pm \$30 million for a 100-MTHM/yr operation.¹⁷ Assuming these preoperational costs are distributed over a nine-year

¹⁷ As can be seen from the notes to Table 6-9, a fuel fabrication plant that produces 100 MTHM/yr for \$1,425–\$1,575/kgHM (with and without taxes and insurance) would have overnight construction costs of about \$600 million, on which a 20-percent addition makes \$120 million.

S-curve prior to operation, the IDC (interest during construction) factor is 1.41 and the contribution to capital investment at start of operation is $1.41 \times \$120 \pm \30 million = $\$170 \pm \40 million. Using the usual fixed charge rates (FCRs) this translates to $\$140 \pm \$30/\text{kgHM}$ without property taxes and insurance and $\$170 \pm \$40/\text{kgHM}$ with property taxes and insurance.

The foregoing figures for MOX fabrication costs do not include metal-to-oxide conversion, nor do they include uranium feed. For our reference 4.8-percent plutonium fuel, and based on a conversion rate corresponding roughly to 100 MTHM/yr at 5 percent plutonium, the metal-to-oxide costs would add $\$7 \pm \$2/\text{gPu} \times 49 \text{ gPu}/\text{kgHM} = \$343 \pm \$98/\text{kgHM}$. The uranium costs are just $\$10/\text{kgHM}$, so the addition for these two items is about $\$350 \pm \$100/\text{kgHM}$. Thus we have altogether, for the costs of making MOX fuel at 4.8 percent plutonium and 100 MTHM/yr in new facilities including preoperational costs and plutonium metal-to-oxide conversion:

fabrication plant costs without preoperational-cost increment
 $\$1,425 \pm \$300/\text{kgHM}$ without paying property tax and insurance
 $\$1,575 \pm \$300/\text{kgHM}$ if property tax and insurance are paid
 preoperational-cost increment
 $\$140 \pm \$30/\text{kgHM}$ without paying property tax and insurance
 $\$170 \pm \$40/\text{kgHM}$ if property tax and insurance are paid
 cost of metal-to-oxide conversion and uranium acquisition
 $\$350 \pm \$100/\text{kgHM}$.

These figures combine to totals of $\$1,915 \pm \$318/\text{kgHM}$ without property tax and insurance and $\$2,095 \pm \$319/\text{kgHM}$ with property tax and insurance (noting that the ranges combine as the square root of the sum of the squares); we round these to $\$1,900 \pm \$300/\text{kgHM}$ and $\$2,100 \pm \$300/\text{kgHM}$, respectively.

In addition to the possibility of building MOX fuel fabrication facilities from scratch, there exists in the United States the possibility of completing the SAFLINE MOX fabrication line that now stands unfinished at the FMEF on DOE's Hanford reservation in Washington state. USDOE (1993b) has estimated the cost of completing the SAFLINE MOX fabrication line to a capacity of 50-MTHM/yr average output, meeting modern health and safety standards, at \$75 million, based on somewhat dated proprietary studies that we have not been able to examine. A more recent estimate commissioned in connection with the Isaiah Project from the individual who was responsible for the most recent work on the FMEF facility (Dahl 1993) gives a figure of "less than \$150 million," explicitly including metal-to-oxide conversion. Operating costs were given as $\$1,500/\text{kgHM}$ in the 1988 study cited by DOE (which, assuming these were 1988 dollars, would be $\$1,740/\text{kgHM}$ in 1992 dollars); this would seem very high if it did *not* include metal-to-oxide conversion, so we assume that the DOE

(text continues on p. 297)

TABLE 6-9 Partially Adjusted Estimates from Literature on MOX Fuel-Fabrication Costs

1992 dollars per kilogram of heavy metal—uranium + plutonium—in MOX fuel, excluding, except where noted, costs of conversion of Pu metal to oxide and “preoperational costs” [e.g., research and development, safety and environmental assessment and licensing, and plant startup and testing]. Extra significant figures are to assist in checking calculations; the precision is illusory.

Cost and Source of Estimate	Basis
\$845 to \$907 (GE 1993) ^a	174-MTHM/yr average output U.S. plant operating for 30 years, without metal-to-oxide conversion.
\$1,026 to \$1,153 (USDOE 1992) ^b	200-MTHM/yr average output commercial plant in France, operating for 30 years, without metal-to-oxide conversion.
\$1,041 to \$1,122 (Westinghouse 1993) ^c	200-MTHM/yr capacity U.S. plant, 150-MTHM/yr average output, 30-year operation, with metal-to-oxide conversion.
\$1,126 ± \$307 (OECD 1992) ^d	200- to 400-MTHM/yr output commercial plant in Europe, 30-year operation, without metal-to-oxide conversion.
\$1,341 to \$1,484 (ABB-CE 1993) ^c	100-MTHM/yr average output U.S. plant, 30-year operation, without metal-to-oxide conversion.
\$1,427 to \$1,576 (USDOE 1993c) ^f	100-MTHM/yr capacity U.S. plant, 30-year operation, without metal-to-oxide conversion.
\$1,500 ± \$300 (USDOE 1993c) ^g	Range of reported estimates for costs of MOX fabrication in European plants in the 100-MTHM/yr size class.
\$1,650 ± \$170 (<i>Nuclear Fuel</i> 1992) ^h	Estimated range for commercial-scale European plants (taken to be 100–125 MTHM/yr).
\$1,890 (Kessler et al. 1992) ⁱ	Estimate for German plant operating at 120 MTHM/yr.
\$3,900 (<i>Nuclear Fuel</i> 1993) ^j	MOX fabrication fees reportedly being charged by Siemens to German utilities in 1990–1991, at output below 20 MTHM/yr plant.

NOTES: The figures shown are based on a starting point of PuO₂ powder delivered to the fuel fabrication plant; costs of conversion of plutonium metal to PuO₂ powder and costs of transport of the PuO₂ to the fabrication plant (if any is required) are not included in these figures, nor is any charge for the WPu itself, or for the uranium feed and its conversion to UO₂ and transport to the fabrication plant, or for cost of ultimate disposal of the spent fuel (or of HLW derived from it). These costs are considered in adjacent subsections of the report. Estimates of construction costs and operating costs for MOX fuel fabrication plants from the indicated references have been put on a consistent basis by the Reactor Panel using the following conventions: contingency factor = 25 percent of direct plus indirect construction costs (i.e., construction-cost estimates that did not include a contingency were multiplied by 1.25, and those that included a different contingency fraction, *c*, were multiplied by 1.25 / [1+*c*]); annual real cost of money *r* = 0.07; interest during construction (IDC) computed at *r* = 0.07 on the basis of S-curves for cumulative construction investment over a construction period of six years unless otherwise specified, giving IDC of 0.27; dismantling and disposal (D&D) costs for facilities assumed to require availability at shutdown of a fund equal to 10 percent of the initial capital investment (including interest during construction) (USDOE 1988b, p. 43) accounted for an annuity taken from operating costs and invested at real interest rate *i* = 3 percent per year (see “Issues and Criteria in Economic Evaluation of Alternatives: in Chapter 3), with magnitude equal to this sum times 0.03 / [(1.03)^{*n*} - 1], where *n* is the operating

lifetime; levelized annual capital charges figured as direct plus indirect construction charges, multiplied as indicated by appropriate contingency and IDC multipliers, then multiplied by fixed charge rate $FCR = r(1+r)^n / [(1+r)^n - 1]$, where n is the total operating lifetime or the duration of the plutonium disposition campaign (if facilities not expected to have any subsequent use). If a private entity were building the facility, the r value might be higher and the FCR would have to be increased by a further 0.02/yr or so to cover property taxes and insurance.

^a GE (1993, p. 10.10) estimates the direct construction cost of a BWR fuel fabrication plant with an average output of 174 MTHM/yr (enough for six 1,300-MWe reactors using 29 MTHM/yr each at 37,100-MWd/MTHM average burnup) at \$240.2 million, and application of General Electric's (GE's) indirect cost factor of 0.4 gives direct plus indirect costs of \$336.2 million. With contingency, overnight costs become $1.25 \times \$336.2 = \420.4 million, and with interest during construction the total capital cost is $1.27 \times \$420.4$ million = \$533.9 million. The levelized-annualized capital costs ($r = 0.07$, 30 years) are \$43.0–\$53.7 million/yr or \$247–\$309/kgHM. GE (1993, p. 10.13) gives the operation and maintenance (O&M) costs for this plant as \$103 million/yr, and with D&D annuity equal to $\$53.4 \text{ million} \times 0.03 / [(1.03)^{30} - 1] = \1.1 million/yr, the total O&M costs are \$104.1 million/yr or \$598/kgHM and the total fuel fabrication costs are \$845–\$907/kgHM. These figures do *not* include plutonium metal-to-oxide conversion (GE 1993, p. 3.16).

^b Indicated cost estimate based on construction of a new MOX fabrication plant in France at a cost of \$1,000 million for a plant with average output 200 MTHM/yr, which would run for 10 years at 5 percent WPU to process 100 MT WPU, then run another 20 years in commercial operation using reactor plutonium. We assume the \$1,000 million includes contingency but not IDC and D&D. Then IDC at $r = 0.07$ and 6-year construction is 1.27 giving initial capital cost = \$1,270 million. FCR = capital recovery factor (CRF) is 0.0806 (no increment for property taxes or insurance is considered), so levelized-annualized cost is \$102.4 million and capital contribution to MOX fuel cost is \$512/kg. Operating cost is quoted as \$100 million/yr, to which D&D charge adds $\$127 \text{ million} \times 0.04 / [(1.07)^{30} - 1] = \2.7 million, so total operating cost is \$102.7 million/yr / 200,000 kgHM/yr = \$514/kgHM, and the fuel cost is \$512/kgHM + \$514/kgHM = \$1,026/kgHM. Addition of 0.02/yr contribution to FCR, corresponding to property taxes and insurance that would be typical for a private operator in the U.S. context, would increase the total by \$127/kgHM to \$1,153/kgHM.

^c The Westinghouse contractor study for the 1992–1993 Department of Energy Plutonium Disposition Study (PDS) gave an estimate of direct plus indirect construction costs for a plant with nameplate capacity of 200 MTHM/yr and average output of 150 MTHM/yr as \$385.4 million (Westinghouse 1993, p. 3-12), a figure said to include the capacity to convert plutonium metal to oxide (Westinghouse 1993, p. 1.4-1). With 25 percent contingency and 27 percent IDC (six years, $r = 0.07$), the initial capital cost becomes \$611.8 million; for a 30-year plant life, FCR = 0.0806/yr without property taxes and insurance and 0.1006 if these are included, so the annualized capital charges are \$49.3–\$61.5 million, or at 150 MTHM/yr \$329–\$410/kgHM. Westinghouse (1993, p. 3-22) gives operating costs for this plant as \$105.5 million/yr, and the D&D annuity for a 30-year operating period would add $\$61.2 \text{ million} \times 0.03 / [(1.03)^{30} - 1] = \1.3 million/yr, so the total is \$106.8 million/yr or \$712/kgHM, and the sum of capital and operating charges is \$1,041–\$1,122/kgHM.

^d The central figure is the OECD study's "reference case" value of \$1,100/kgHM in 1991 dollars, explained in the report's Annex 6 and applicable during OECD's assumed operating period of 2007–2035, and here converted to 1992 dollars as $\$1,100 \times 1/0.977 = \$1,126$. The figure is based on the assumption that MOX fuel fabrication will cost four times the report's \$275/kgHM figure for LEU fuel fabrication. The sensitivity studies cited in the report yielded a range of \$800–\$1,400/kgHM (1991 dollars) surrounding the "reference" value, hence $\pm \$300 \times 1/0.977 = \pm \307 .

^e The ABB-Combustion Engineering estimate of direct plus indirect construction costs for a 100-MTHM/yr average output fuel fabrication plant in connection with DOE's PDS is \$450 million (ABB-CE 1993, p. VI-14, interpreted as the sum of direct plus indirect charges but without contingency in USDOE [1993a, pp. SC6-10 and A-13]). With contingency, the overnight cost is $1.25 \times \$450 \text{ million} = \562.5 million, and the total capital costs with IDC are $\$562.5 \text{ million} \times 1.27 = \714.4 million. At $r = 0.07$ and with a 30-year operating life, the FCR is 0.0806 without property taxes and insurance and 0.1006 with these, so the levelized annual capital charges are

\$57.6–\$71.9 million/yr, and at 100 MTHM/yr these translate to \$576–\$719/kgHM. ABB-CE estimates the operating costs of this plant to be \$75 million/yr (ABB-CE 1993, p. VI-16) and the D&D annuity adds $\$71.4 \text{ million} \times 0.03 / [(1.03)^{30} - 1] = \1.5 million/yr , so total operating costs are \$76.5 million/yr, or \$765/kgHM. The total MOX fabrication costs are then \$1,341–\$1,484/kgHM; in the ABB-CE analysis these do *not* include plutonium metal-to-oxide conversion (ABB-CE 1993, p. III-82).

^f USDOE (1993c) quotes an overnight cost of \$440 million (1992 dollars) for a 100-MTHM/yr capacity plant and annual operating costs of \$62 million (from the Nuclear Energy Cost Data Base maintained at the Oak Ridge National Laboratory) including D&D annuity. With IDC ($r = 0.07$, six years) the capital cost is \$558.8 million, and the levelized annual capital costs ($r = 0.07$, 30 years) are \$45.0–\$56.2 million/yr (with and without property taxes and insurance). We assume this plant operates at 75-percent capacity factor, yielding an average output of 75,000 kgHM/yr, hence capital charges costs of \$600–\$749/kgHM and operating costs of \$827/kgHM, for total costs of \$1,427–\$1,576/kgHM. We assume that, following conventional practice in LWR fuel-cycle analysis, the input to this fuel fabrication plant was assumed to be plutonium dioxide, so that metal-to-oxide conversion costs are not included.

^g USDOE 1993c quotes a range of estimates that have been made for European plants in the 100 MTHM/yr class as extending from \$1,200–\$1,800/kgHM, hence $\$1,500 \pm \300 .

^h Quoted in Berkhout et al. (1993). The range given was \$1,300–\$1,600/kg of MOX, which translates to \$1,480–\$1,820/kgHM.

ⁱ Based on Kessler et al. (1992, p. 25) where MOX fabrication costs “are estimated at about DM 3,000/kg at present in a fabrication plant with a throughput of 120 t/yr and adequate plant utilization.” At \$0.63/1 DM, 3,000 DM equals \$1,890 in U.S. dollars.

^j The figure is based on a successful Siemens lawsuit against the state of Hesse for damages in the form of lost revenues in a state-ordered shutdown of that firm’s 25-MTHM/yr MOX fabrication plant in Hanau. Lost revenues were estimated at DM 550,000 per day for an output of 103 kg/d of MOX measured as oxide, hence 6,070 DM/kgHM, which at \$0.63/1 DM is about \$3,800/kgHM in 1991 U.S. dollars or \$3,900/kgHM in 1992 U.S. dollars.

capital-cost as well as operating-cost estimates for the FMEF MOX fabrication line do include this conversion. The Dahl (1993) study did not estimate operating costs.

In constructing our own estimate of the cost of MOX fuel from the FMEF facility, we use a range of \$100–\$150 million in completion costs, which we take to be overnight costs; with an assumed four-year construction time, hence IDC multiplier of 1.19, the initial capital investment would be \$119–\$179 million. Since this facility is already on a federal site, we use only the lower fixed charge rate corresponding to no property tax or insurance charges, giving levelized-annualized capital charges of \$9.6–\$14.4 million/yr for a 30-year operating life, or \$192–\$288/kgHM at 50 MTHM/yr.

Concerning operating costs at FMEF, scaling from a central estimate of \$800/kgHM for the operating cost of a new plant producing 100 MTHM/yr (see notes to Table 6-9) would give \$1,060/kgHM for a 50-MTHM/yr plant, and adding \$226/kgHM for the operating-cost component of metal-to-oxide conversion¹⁸ gives roughly \$1,290/kgHM. Because per-unit-output operating costs at

¹⁸ Based on \$7/gPu total at a scale corresponding to 100 MTHM/yr and 49 gPu/kgHM, increased by $2^{0.4} = 1.32$, as derived earlier in this section, assumed split 50/50 between capital charges and operating costs.

FMEF are expected to be higher than at a new plant, we take this to be the lower limit of a range with the DOE estimate of \$1,740/kgHM at the upper end. Assuming dismantling and disposal (D&D) costs require \$40 million at the end of plant life, the 30-year annuity payment required to produce this sum if invested at 3 percent per year real interest rate is \$0.8 million/yr or \$17/kgHM, so our operation and maintenance (O&M) costs are \$1,310–\$1,760/kgHM. The capital plus operating costs for the FMEF MOX fabrication line, not including preoperational costs, would then be \$1,500–\$2,050/kgHM.

We suppose that part of the preoperational costs already have been spent in the case of the FMEF MOX fabrication line, and we consider these sunk. Some additional preoperational costs probably are embedded in the DOE and Dahl estimates of the completion costs for the facility. We assume the incremental preoperational costs would be \$20–\$30 million (20 percent of \$100–\$150 million); if these are incurred over a six-year period prior to operation (greater than the assumed four-year construction period), they will add $1.27 \times \$20\text{--}\30 million = \$25–\$38 million, which translates to \$40–\$60/kgHM. The total costs of MOX fuel fabrication at FMEF, at a scale of 50 MTHM/yr, including metal-to-oxide conversion, are thus \$1,530–\$2,110/kgHM, which we round to $\$1,800 \pm \$300/\text{kgHM}$.

MOX: Incremental Costs of Plutonium Storage and Transport

In the case of LEU, costs of storage and transport at and between the various steps of fuel preparation are included in the cost figures that were given. In the case of MOX fuel the storage and transport costs can be considerably higher (because of the extra ES&H and security hazards posed by the plutonium), and not all of them have been included in the estimates presented above. (It may be presumed that storage at the plutonium conversion plant and MOX fuel fabrication plant is included in the estimates of the costs of these activities, but transport costs and any extra costs associated with the storage of MOX fuel after it leaves the fabrication plant but before it is loaded into the reactor core are not included.) The OECD study (1992) cites plutonium storage costs of \$1–\$2/gPu per year and plutonium transport costs of \$500–\$900/kgPu for transport within the European community (i.e., excluding long-distance transport by ship). We assume that the incremental storage costs for a MOX fuel operation in the United States will be between \$0 and \$2/gPu and that the incremental transport costs will be between \$0.5 and \$1.5/gPu, so that the total incremental costs for storage and transport are $\$2 \pm \$1/\text{gPu}$, hence about $\$100 \pm \$50/\text{kgHM}$ for 49 gPu/kgHM.

MOX: Cost of Ultimate Disposal

As indicated in the corresponding discussion under LEU, ultimate disposition costs for MOX fuels may range from the same as to perhaps twice as high as those for LEU fuels, but because of the large uncertainties associated with these costs for either fuel we present our main comparisons without them.

MOX: Estimated Total Fuel Costs, Less Disposal

A campaign to process 50 tons of WPu into MOX fuel with our reference-case plutonium content of 4.8 percent would entail fabricating about 1,000 MTHM of MOX fuel, which could be done in 20 years at an average fabrication rate of 50 MTHM/yr or in 10 years at an average rate of 100 MTHM/yr. If the intent were to use this fuel in one or two reactors with full-MOX cores (as might be desirable to minimize transport), the 50-MTHM/yr option would be adequate. If the aim were to complete the campaign as quickly as possible (at the cost of somewhat higher capital investment in fabrication capacity and the engagement of a larger number of reactors), the 100-MTHM/yr option would be appropriate. This second option would of course also be adequate to handle, over a period of 20 years, a case in which the campaign doubled in size from 50–100 tons of WPu because of either deeper arsenal reductions or acquisition of Russian plutonium for disposition in the United States. A fuel fabrication rate as low as 25 MTHM/yr, applied over a period of 30 years, would suffice to process 50 tons of WPu if plutonium loading were increased to 6.7 percent by weight.

The range of scales of MOX operations likely to be considered for a WPu disposition campaign therefore extends from about 25 MTHM/yr to about 100 MTHM/yr. For our reference-case MOX fuel, designed to deliver 40,000 MWd/MTHM with end-of-life reactivity equal to that of 4.4-percent enriched LEU after the same irradiation, we estimate the fuel-cycle costs at these scales of operation in the period from 2000 to 2030 to be as shown in Table 6-10, excluding repository fees. (Scaling from 100-MTHM/yr operations in a new plant and from 50-MTHM/yr operations at FMEF has been carried out assuming total costs in a plant of a given type scale with the 0.6 power of output, so that unit costs decline with the 0.4 power of output.)

Net Fuel-Cycle Costs of Using WPu in PWRs

For the reference case we have been considering—substitution of WPu in MOX for 4.4-percent U-235 (LEU) PWR fuel achieving an irradiation of 40,000 MWd/MTHM—we found above that the relevant levelized-annualized fuel cost (less the cost of ultimate waste disposal) for LEU fuel in the period 2000–2030 is \$1,400 ± \$200/kgHM. In Table 6-11, this figure is combined with the MOX

TABLE 6-10 Summary of MOX Fuel-Cycle Costs (in 1992 U.S. dollars)

	New Fuel Fabrication Plant		Fuel Fabrication at FMEF
	No Tax and Insurance	With Tax and Insurance	
MOX Production Costs			
100 MTHM/yr	1,900±300	2,100±300	NA ^a
50 MTHM/yr	2,500±400	2,800±400	1,800±300
25 MTHM/yr	3,300±500	3,700±500	2,400±400
Incremental Plutonium Storage and Transport	100±50	100±50	100±50
Totals			
100 MTHM/yr	2,000±300	2,200±300	NA ^a
50 MTHM/yr	2,600±400	2,900±400	1,900±300
25 MTHM/yr	3,400±500	3,800±500	2,500±400

NOTES: MOX production costs include fabrication with preoperational costs, plutonium metal-to-oxide conversion, and depleted uranium acquisition and conversion. Costs are in 1992 dollars/kgHM and apply to the small-scale MOX operations that would be associated with a program restricted to disposition of WPu. They do *not* apply to a large-scale commercial MOX program, in which economies of scale could be gained from higher outputs and preoperational costs would be distributed over much higher production. Uncertainty ranges are judgmental 70-percent confidence intervals (corresponding approximately to one standard deviation of a normally distributed random variable).

^a NA indicates not applicable. To our knowledge, expanding the FMEF MOX fabrication capability to 100 MTHM/yr has not been studied.

fuel-cycle-cost estimates from Table 6-10 to give a set of values for the expected excess of MOX fuel-cycle costs over those for LEU, expressed in dollars per kilogram of heavy metal, per electrical kilowatt-hour, and per gram of WPu.

The differential costs in Table 6-11 can be converted readily into annual costs and into net discounted present values of such cost streams for a full plutonium disposition campaign. For example, at a MOX fabrication rate of 50 MTHM/yr and the indicated plutonium loading of 4.8 percent in heavy metal, a 50-ton plutonium campaign would require 21 years of MOX fuel fabrication and corresponding MOX-based electricity generation, at an excess cost of \$91.4 ± \$27.4 million/yr (including fuel carrying charges) if the MOX is fabricated in a new plant that pays property taxes and insurance, \$73.1 ± \$27.4 million/yr if it is fabricated in a new plant that does not pay taxes and insurance, and \$29.5 ± \$20.6 million/yr if it is fabricated at FMEF. The discounted net present values of these cost streams at the start of operation, with real cost of money at 7 percent per year, are \$991 ± \$297 million, \$792 ± \$297 million, and \$329 ± \$231 million, respectively. (A campaign with this timing and plutonium loading could be accomplished using two 1,250-MWe PWRs with full-MOX cores or six such

TABLE 6-11 Fuel-Cycle Cost Differentials for MOX Versus LEU Fuel (in 1992 U.S. dollars)

	MOX from New Fabrication Plant		
	No Tax and Insurance	With Tax and Insurance	MOX Fabrication at FMEF
Cost excess of MOX over LEU fuel in \$/kgHM if MOX is fabricated at a scale of			
100 MTHM/yr	600±350	800±350	NA ^a
50 MTHM/yr	1,200±450	1,500±450	500±350
25 MTHM/yr	2,000±550	2,400±550	1,100±450
Equivalent differential in electricity generation cost in \$/MWh ^b			
100 MTHM/yr	2.3±1.3	3.0±1.3	NA ^a
50 MTHM/yr	4.6±1.7	5.7±1.7	1.9±1.3
25 MTHM/yr	7.6±2.1	9.1±2.1	4.2±1.7
Equivalent cost per gram of WPU ^c			
100 MTHM/yr	15±9	20±9	NA ^a
50 MTHM/yr	30±11	37±11	12±9
25 MTHM/yr	50±14	60±14	27±11

NOTES: Figures apply to fuel irradiated to an average of 40,000 MWd/MTHM, with initial loadings of 4.8 percent WPU in heavy metal (MOX case) and 4.4 percent U-235 in heavy metal (LEU case).

^a NA indicates not applicable. Expanding the FMEF MOX fabrication capability to 100 MTHM/yr has not been studied.

^b Thermal-to-electric conversion efficiency = 0.33, fuel accounting life five years with real cost of money $r = 0.07$.

^c Equal to cost per kilogram of heavy metal times carrying-charge factor on five-year fuel cycle, divided by 49 gPu/kgHM. Rounded to nearest dollar.

PWRs with one-third MOX cores, and could perhaps begin, if the fuel were fabricated at FMEF, shortly after the year 2000; see Table 6-2.)

Variations in Fissile Content and Burnup

The estimates derived from calculations along the foregoing lines depend on the assumptions chosen about the fissile content and burnup of fuel. At lower fissile content, and burnup, the economic comparison becomes even less favorable to MOX fuel; at higher fissile content and burnup, it becomes more favorable to MOX. The reason for this is that two of the main components of LEU fuel cost per kilogram of heavy metal—uranium-acquisition and enrichment costs—increase sharply as the fissile content rises, while the cost of MOX fuel fabrication per kilogram of heavy metal, which is by far the largest component

Table 6-12 MOX-LEU Fuel-Cycle Cost Comparison Versus Enrichment and Burnup (in 1992 U.S. dollars)

	30,000 MWd/MTHM		40,000 MWd/MTHM		50,000 MWd/MTHM	
	3.3% U-235 in LEU	3.6% WPu in MOX	4.4% U-235 in LEU	4.8% WPu in MOX	5.5% U-235 in LEU	6.0% WPu in MOX
Fuel cost, \$/kgHM	1,040	2,120	1,400	2,200	1,780	2,280
Fuel cost, \$/MWh	5.2	10.5	5.2	8.2	5.3	6.8
MOX penalty, \$/kgHM	— 1,080 —		— 800 —		— 500 —	
MOX penalty, \$/MWh	— 5.3 —		— 3.0 —		— 1.5 —	

NOTE: Costs relate to PWRs fed by new MOX fuel fabrication plants paying property taxes and insurance and operating at 100 MTHM/yr.

of MOX fuel cost in a situation where the plutonium metal is obtained free of cost, goes up only a little or not at all with plutonium loading.¹⁹

The sensitivity of the cost comparison to fissile content is illustrated quantitatively in Table 6-12, where it has been assumed that the only component of MOX fuel cost, from a new fuel-fabrication plant, that varies with plutonium content is the cost of conversion of plutonium metal to oxide (taken to be \$7/gPu at 100 MTHM and 5 percent plutonium by weight in heavy metal). It is apparent that, if the MOX made from WPu were replacing 5.5-percent enriched LEU, the central-estimate cost penalty would be cut almost in half compared to the 4.4-percent enriched LEU substitution, and the penalty per kilowatt-hour of electricity generated would fall by more than twofold. It cannot be assumed, however, that further reductions in the WPu/MOX cost penalty would be obtainable by going to still higher plutonium loadings in PWR MOX, because burnups above 50,000 MWd/MTHM are not available in LWRs with current fuel designs. (This limitation might change in the future, but then fuel fabrication costs probably also would change, invalidating the present calculations.)

Costs of Reactor Modifications and Licensing

The preceding estimates do not take account of the costs of any modifications to existing LWRs in order to enable them to utilize 100-percent MOX cores, nor do they include costs that would be associated with safety analysis

¹⁹ There may be some component of the fuel fabrication cost that depends on plutonium content rather than just on total amount of heavy metal in fabricated fuel, but we have not been able to find an estimate of such an effect either in the literature or in conversations with individuals experienced in the design and operation of MOX fuel fabrication plants.

and licensing of MOX use in existing LWRs whether at one-third or full-MOX operation.

If it were decided to use only one-third MOX cores, or to use full-MOX cores in one or more of the three operating U.S. LWRs that were designed for full-MOX operation (the three ABB-Combustion Engineering System-80 reactors at the Palo Verde station in Arizona), no costs would be incurred for reactor modification; but there would still be costs for MOX-related safety analyses and licensing. A study of completing, for the purpose of WPu disposition, the WNP-1 1,250-MWe PWR in Washington state estimated the “safety, licensing, and permitting” costs of doing this at \$94 million (SAIC 1993a); what fraction of this total is related to MOX use per se, and hence applicable to the case of licensing an already operating reactor for this purpose, is not clear. In the Plutonium Disposition Study of the Department of Energy (USDOE 1993a), vendor estimates of “preoperational costs” (a category that includes research and development and plant testing as well as safety analysis and licensing) for advanced LWRs ranged from \$93 to \$244 million; and DOE’s Technical Review Committee complained that the vendors had underestimated the safety analysis and licensing costs. We assume here that the safety and licensing costs for MOX operation in existing U.S. LWRs would fall in the range of $\$100 \pm \50 million.

We have seen no detailed estimates for the cost of modifying an operating U.S. LWR to be able to use a full-MOX core, if that were required.²⁰ Our own very rough estimate is that, in a case where substantial modifications to the control systems were in fact necessary, such modifications might cost $\$100 \pm \50 million in labor and equipment, while requiring a shutdown of the reactor for perhaps a year. The lost electricity revenues from a year’s shutdown of a 1,200-MWe reactor that had been operating at 75-percent capacity factor would amount to about \$390 million if the busbar value of the electricity is \$0.05/kWh (see Chapter 3), or $\$390 \pm \130 million allowing for a ± 33 -percent uncertainty range on the product of shutdown time and electricity value.

Net Economic Costs of Using WPu in Currently Operating PWRs

Consider a reference case in which two currently operating PWRs in the 1,200-MWe class are to load the nominal 50 tons of surplus U.S. WPu over a period of 21 years, using 100-percent MOX cores with 48 kgPu/MTHM and average burnup of 40,000 MWd/MTHM. If the reactors in question need no modification to use 100 percent MOX and if safety analysis and licensing costs

²⁰ Analyses performed by vendors in the second phase of DOE’s Plutonium Disposition Study, which became available late in the panel’s deliberations, suggest—contrary to previous assumptions—that several existing LWR types besides the System-80 could in fact use 100-percent MOX cores safely without undergoing significant modification. Further analysis and review will be required before this conclusion can be considered firm. See also “Environment, Safety, and Health” below.

TABLE 6-13 Net Economic Impact of WPu Disposition in Currently Operating LWRs (in millions of 1992 U.S. dollars)

Fuel From:	No Reactor Modifications Required			Reactor Modifications Required		
	FMEF	New Plant no Tax	New Plant with Tax	FMEF	New Plant no Tax	New Plant with Tax
MOX fuel costs	329±231	792±297	991±297	329±231	792±297	991±297
Costs of reactor						
Licensing	127±64	127±64	127±64	127±64	127±64	127±64
Modification	0	0	0	1,000±300	1,000±300	1,000±300
Rounded Totals	456±240	919±304	1,118±304	1,456±384	1,919±427	2,118±427

NOTE: Figures are discounted present value, as of start of operation (shortly after the year 2000), of the net incremental costs of using WPu-MOX instead of LEU in two 1,200-MWe PWRs to disposition 50 tons of WPu in 21 years of reactor operation (assuming 48 gPu/kgHM as loaded—49 gPu/kgHM manufactured—and average burnup of 40,000 MWd/kgHM).

of \$100 ± \$50 million in 1992 dollars are assumed to accumulate in the usual S-curve during a period of six years prior to the start of MOX operation at the reactors, the contribution to the discounted net present value of incremental MOX costs at the time of startup, for real cost of money at 7 percent per year, would be $1.27 \times (\$100 \pm \$50 \text{ million}) = \$127 \pm \64 million .

If both reactors need substantial modifications to be able to use 100-percent MOX cores, the estimated additional costs are $2 \times (\$100 \pm \$50 \text{ million})$ in labor and equipment plus $2 \times (\$390 \pm \$130 \text{ million})$ in lost electricity revenues during shutdown, or about \$1,000 ± \$300 million in total modification costs. Table 6-13 combines these estimates with the above-derived net discounted present value of the extra fuel-cycle costs as of start of operations, giving a range of central estimates of the net economic impact of using currently operating PWRs for WPu disposition extending from \$450 to \$2,100 million net discounted present value at start of operations, depending on whether FMEF or a new MOX fuel fabrication plant is used and on whether substantial modifications to permit 100-percent MOX use are required or not.

Variations in Reactor Type

The economics of using WPu in reactors would be more attractive for reactors that require high levels of enrichment and derive high burnups from it. In a liquid-metal fast-breeder reactor (LMFBR), for example, fuel to be used in the core would typically be enriched to 20 percent fissile plutonium or 30 percent U-235²¹ and would achieve a burnup of 100,000–150,000 MWd/MTHM. The uranium requirement for 30-percent enriched fuel is 63.3 kgU/kgHM and the

²¹ Plutonium is substantially more reactive in a fast-neutron spectrum than is U-235.

separative work requirement is 64.2 SWU/kgHM (assuming 0.25 percent U-235 in the tails). The Nuclear Energy Cost Data Base (USDOE 1988b) reference value for the fabrication cost of plutonium-based LMFBR core fuel is \$2,700/kgHM (in 1992 dollars), compared to \$500/kgHM for fabricating enriched-uranium LMFBR fuel. Thus, with the usual assumption of 0.5 percent losses in conversion and 1 percent losses in fabrication, and with uranium acquisition and conversion costs of (\$55 + \$9)/kgU and enrichment costs of \$95/SWU, the cost of enriched uranium fuel for an LMFBR core (without ultimate disposal costs) would be

$$1.015 \times 63.3 \text{ kgU/kgHM} \times \$64/\text{kgU} + 1.01 \times 64.2 \text{ SWU/kgHM} \\ \times \$95/\text{SWU} + \$500/\text{kgHM} = \$10,772/\text{kgHM} .$$

On the MOX side, allowing \$9/gPu for conversion to oxide and incremental storage and transport costs, as before, and \$7/kgHM for acquisition and conversion costs for 0.7 of depleted uranium per kilogram of heavy metal, leads to a corresponding MOX fuel cost of

$$\$7/\text{kgHM} + 1.015 \times 200 \text{ gPu/kgHM} \times \$9/\text{gPu} + \$2,700/\text{kgHM} \\ = \$4,534/\text{kgHM} .$$

Thus a cost advantage of about \$6,000/kgHM is predicted for the use of plutonium in a MOX-fueled LMFBR, given cost-free WPu as the raw material (translating to \$0.0074/kWh on a levelized basis, assuming $r = 0.07$, a four-year fuel cycle, irradiation of 100,000 MWd/MTHM, and thermal-to-electric conversion efficiency of 0.40). A similar result could be expected from a comparison of MHTGRs using 100-percent plutonium fuel against the use of 94-percent enriched uranium fuel in these reactors.

It should not be assumed, however, that the fuel-cost benefits of using free WPu in these advanced reactor types, compared to using enriched uranium in them, mean that there is a large economic benefit to be derived from choosing such reactors as the disposition option. The overall economic consequences of such a choice would depend also on the construction costs of such reactors, on their total fuel costs compared to those of LWRs, and on the time delay (and consequent charges for plutonium storage) before they could be deployed. These matters are discussed further in the section "Building New Reactors for Plutonium Disposition" below.

A last alternative reactor case for which the analysis of the economic effect of the use of WPu is especially interesting is that of the CANDU reactor, of which a substantial number are in commercial operation. These reactors normally use unenriched uranium-oxide fuel. A study by the manufacturer of the use of CANDU reactors for WPu disposition (AECL 1994) indicates that two such reactors at Canada's Bruce station could irradiate 50 tons of WPu in 24 years using a current-technology MOX fuel containing 1.2 weight percent plutonium (burnup 9.7 MWd/kgHM) and that four reactors could irradiate 100 tons

of WPu in 25 years using an advanced MOX fuel containing 2.1 weight percent plutonium (burnup 17.1 MWd/kgHM). Based on a rather detailed study of FMEF at Hanford, Washington, analysts at Atomic Energy of Canada, Limited, (AECL) conclude that this facility could be modified to fabricate the needed quantities of MOX CANDU fuel for an overnight cost of \$118 million, and that the operating costs would be about \$64 million per year for the reference MOX fuel and about 20 percent more for the advanced fuel (which, however, would fuel four reactors rather than two and, thus generate twice as much electricity), assuming plutonium metal-to-oxide conversion has been performed (and paid for) elsewhere.²²

For AECL's assumed four-year construction time for modifications to the FMEF, the IDC factor would be 1.18 and the initial investment at startup would be $1.19 \times \$118 \text{ million} = \140 million . The corresponding levelized constant-dollar capital charges at $r = 0.07$ and a 24-year operating lifetime would be \$12 million/yr (no insurance and taxes), which when added to the O&M costs of \$64 million/yr gives fuel costs of \$76 million/yr. (Divided by the 170.6-ton annual output of fuel, this gives \$446/kgHM. Adding metal-to-oxide conversion costs of \$7/gPu and incremental plutonium storage and transport costs of \$2/gPu would add another $\$9/\text{gPu} \times 2.12 \text{ MgPu/yr} = \19 million/yr or $\$9/\text{gPu} \times 12 \text{ gPu/kgHM} = \$108/\text{kgHM}$.) Fueling costs for the standard natural uranium feed for these reactors are estimated by AECL at \$16.3 million/yr for 199.4 MTHM/yr (burnup 8.3 MWd/kgHM), hence \$82/kgHM. The costs of reactor-facility modification and reactor licensing to permit use of plutonium fuel are estimated by AECL at \$37 million, which under the same assumptions as used above would translate to a levelized constant-dollar cost of \$3.8 million/yr. The incremental cost of plutonium disposition under this option can be calculated, therefore, as $\$12 \text{ million/yr} + \$64 \text{ million/yr} + \$19 \text{ million/yr} + \$4 \text{ million/yr} - \$16 \text{ million/yr} = \83 million/yr for 24 years. The equivalent discounted present value at start of plutonium disposition operations at the reactor is \$952 million.

Completing Existing LWRs

Within the option of using U.S. LWRs to process WPu into spent fuel, one of the potentially attractive variants is the possibility of completing, for this purpose, one or more existing partially completed reactors that exist in the United States, such as those of the Washington Public Power Supply System (WPPSS) or the Tennessee Valley Authority (TVA). Two of the WPPSS Nuclear Project reactors—WNP-1 and WNP-3—are of particular interest. The WNP-1 facility is

²² The assumption in the AECL study, on which its estimates of FMEF completion costs and operating costs are based, that plutonium metal-to-oxide conversion is performed elsewhere, differs from the assumption of FMEF studies (cited above) that this conversion would be performed at the FMEF. The AECL analysts agreed that FMEF could be adapted to perform the conversion step, but they did not estimate the costs of this.

a 1,250-MWe Babcock and Wilcox PWR located on DOE's Hanford reservation and is 63 percent complete. The WNP-3 facility is a 1,240-MWe Combustion Engineering System-80 PWR located at Satsop (west of Olympia) and is 75 percent complete. WPPSS has been keeping these reactors in mothballs for a number of years, but has recently decided to stop maintaining them and to sell the parts; thus to keep this option open would require government action in the near term.

The attractions of this variant include, above all, the possibility of processing the entire 50 tons of U.S. surplus WPu at a single reactor site (since either reactor could be completed with the capability to utilize a full-MOX core and could be expected to be operable for at least 30 years after startup) and the minimization of shipping of MOX fuel (since WNP-1 is on the Hanford reservation where a MOX fuel fabrication capability is already partly in place, and WNP-3 is less than 200 miles from that site). In addition, the circumstances of these two reactors lend themselves to government acquisition of the facilities, if it is deemed preferable to perform the plutonium disposition mission at a government site; and because the reactors are not now operating, equipping them with the capacity to use full-MOX cores can be accomplished without the loss of revenue that would be associated with shutting down one or more existing LWRs for modifications.

If only one of the two reactors is to be used for the plutonium disposition mission, the choice between them will involve a trade-off between (1) ease of completion and licensing for use with a full-MOX core, in which WNP-3 has an advantage because it is a Combustion Engineering System-80 reactor (designed from the outset to be able to use a full-MOX core), and (2) reducing MOX transport, in which WNP-1 has an advantage because it is located at the Hanford site where the MOX fuel is likely to be fabricated. The advantage of using *both* reactors for the plutonium disposition mission would be the capacity to load the quantity of MOX containing 50 tons of WPu in half the time that would be required if only one reactor were used. (Another two-reactor variant would be to use WNP-1 together with the WNP-2, which is an operating 1,100-MWe BWR located at the Hanford site. This might or might not entail shutting down WNP-2 temporarily for modifications to permit it to operate with a full-MOX core—see Chapter 4.)

The possibility of using the WNP-1 and WNP-3 reactors for this purpose has been intensively investigated and promoted by a consortium of companies calling this effort the Isaiah Project.²³ In connection with this joint venture, the costs of completing the two reactors and operating them in a plutonium disposition mode have been explored. Under the Isaiah Project proposal, the consor-

²³ The consortium comprises Science Applications International Corporation (SAIC), Newport News Industrial Corporation, and Battelle Memorial Institute. The Isaiah Project is described in SAIC (1993a, 1993b).

tium would complete the WNP-1 and WNP-3 reactors and operate them as government-owned, contractor-operated facilities that would sell steam to WPPSS, which would retain ownership of the turbine generator facilities at the two reactors and would sell the electricity to the Bonneville Power Administration (BPA).

The U.S. government would pay all fuel-cycle and other operating costs (including, e.g., the costs of conversion of WPu to metal and of MOX fuel fabrication, and the costs of spent fuel disposal). A part of the revenue stream from the operation would provide the return to the private investors in the consortium; additional money borrowed against the rest of the anticipated revenue stream would be used to reduce by \$2 billion the debt incurred by BPA in connection with the initial construction of these reactors, and to provide an additional \$2 billion to DOE for use in other antiproliferation programs, perhaps including WPu disposition in the former Soviet Union.

We do not address here the merits of the particular institutional and financing arrangements proposed by the Isaiah Project, confining ourselves instead to summarizing and commenting on the Project's analysis of the costs and revenues associated with WPu disposition using the WNP-1 and WNP-3 plants.

The Project estimates that the completion of the two reactors would cost \$3.3 billion (1993 dollars)—\$1.7 billion for WNP-1 and \$1.6 billion for WNP-3—including costs of safety analysis, licensing, and permitting for full-MOX operation, but not including the turbine generator portion of the plants, which under the Isaiah proposal would be financed separately. The indicated figures are overnight costs, including both direct and indirect cost components plus a contingency of 10 percent. They are based on extensive recent analyses by the WPPSS and the reactor vendors and architect engineers originally responsible for these plants (WPPSS 1992a, 1992b). In the same analyses, the costs of completing the turbine generator part of the plants were estimated as \$240 million for WNP-1 and \$100 million for WNP-3. The proponents argue that the credibility of these estimates is reinforced by their status as "capped-cost" figures, meaning that the contractors have offered to complete the plants at these costs, absorbing any overruns themselves.

To put these estimates into our own framework for calculating initial capital costs as the basis of a levelized-annualized capital charges calculation, we need to convert the figures to 1992 dollars and add a correction for IDC. The conversion to 1992 dollars consists of a multiplicative factor of $1/1.03 = 0.97$ (see Table 3-6). Interest during construction, given the Isaiah Project's 1993 estimate of reactor startup in 1999 if work had been initiated at the beginning of 1994, would contribute a multiplicative factor of about 1.25 at a real cost of money of

7 percent per year (see Table 3-7).²⁴ Thus the 1992 dollar completion costs of the two plants as of a 1999 startup date would be

$$\text{WNP-1: } (\$1.7 \text{ billion} + \$0.24 \text{ billion}) \times 0.97 \times 1.25 = \$2.35 \text{ billion}$$

$$\text{WNP-3: } (\$1.6 \text{ billion} + \$0.10 \text{ billion}) \times 0.97 \times 1.25 = \$2.06 \text{ billion .}$$

These are not the total construction investments as of startup but the incremental investments needed to finish partly completed plants. Thus it is not appropriate, strictly speaking, to apply the rule of thumb from the Chapter 3 section "Principles and Pitfalls in Cost Comparisons" that 10 percent of the initial investment will be needed at the end of a facility's life to cover its dismantling and disposal (D&D) costs. If, however, we use instead the figure recommended in DOE's Nuclear Energy Cost Data Base (USDOE 1988b) of \$145 million (1987 dollars) for an 1,100-MWe plant, converting to 1992 dollars and scaling up for a 1,250-MWe plant would give an end-of-life D&D fund of \$200 million, which is about what the 10-percent rule of thumb would give if applied to the completion costs. For a nominal 30-year reactor lifetime and real cost of money of 7 percent per year, the effective increment on initial capital investment is \$200 million divided by $(1.07)^{30}$, or \$26 million. With this addition, the effective initial capital investments to complete the two plants would become, respectively, \$2.61 and \$2.32 billion.

Let us now estimate the costs and revenues of using one of these plants to process the entire nominal 50 tons of WPu into spent fuel in an operating lifetime of 30 years; this corresponds (see Table 6-1) to plutonium loading in fresh fuel of 6.8 percent of total heavy metal by weight, capacity factor of 75 percent, average burnup of about 42 MWd (thermal)/kgHM, and fuel fabrication of 25 MTHM/yr. If MOX fuel production costs are $\$2400 \pm \$400/\text{kgHM}$ (Table 6-10, for 25-MTHM/yr throughput) and if the incremental costs of plutonium storage and transport are a relatively low $\$1/\text{gPu}$ (in light of minimal transport between fuel fabrication and reactor), hence $\$70/\text{kgHM}$, and with costs of $\$10/\text{kgHM}$ for uranium acquisition and conversion, the fuel-cycle costs less disposal would be about $\$2,500 \pm \$400/\text{kgHM}$. If carrying charges on the fuel are based on a five-year fuel accounting lifetime at a real cost of money of 7 percent per year, then at 42.2 MWd/kgHM and a thermal-to-electric conversion efficiency of 0.33, these fuel-cycle costs amount to $\$0.0076\text{--}\$0.0106/\text{kWh}$, and the addition of the statutory $\$0.001/\text{kWh}$ for waste-disposal costs makes this range $\$0.0086\text{--}\$0.0116/\text{kWh}$. The nonfuel operation and maintenance costs for a large PWR are in the range of $\$0.012\text{--}\$0.016/\text{kWh}$ according to DOE's Nuclear Energy Cost Data Base (USDOE 1988b, with conversion of the reference's 1987 dollars to 1992 dollars).

²⁴ Shifting the projected time from 1994 to 1996, and projected completion to 2001, would have only a modest impact on the projected cost.

For a nominal plant lifetime of 30 years and a real cost of money of 7 percent per year, the levelized-annualized capital charges on the completion costs of \$2.3–\$2.6 billion would be

- \$2.3–\$2.6 billion \times \$0.0806/yr = \$185–\$210 million/yr for an entity that did not pay property taxes and insurance, and
- \$2.3–\$2.6 billion \times \$0.1006/yr = \$231–\$261 million/yr for an entity that paid property taxes and insurance amounting to 0.02/yr of the initial capital investment.

The corresponding range of capital charges per kilowatt-hour at 75-percent capacity factor would be \$0.023–\$0.032/kWh.

These figures do not include the cost of acquiring one of the WNP reactors from its current owners. The Isaiah Project allowed a cost of \$1 billion per reactor for this purpose. (WPPSS had invested some \$5 billion in the two reactors, but, inasmuch as they are now slated to be sold for scrap, the owners might be quite content with \$1 billion per reactor, possibly less; see Lange and Hanson 1993.) Financing this additional \$1 billion at 7 percent real cost of money over 30 years would add costs of \$80.6–\$100.6 million/yr (depending on whether a charge for property taxes and insurance is assessed against this sum), hence \$0.0098–\$0.0123/kWh.

In summary, our estimate of the cost of using one of the WNP reactors for disposition of 50 tons of WPu over a 30-year period, if expressed as a cost per kilowatt-hour of electricity generated in the process is:

MOX fuel-cycle costs, including waste disposal	\$0.0101 \pm \$0.0015/kWh
nonfuel operation and maintenance	\$0.0140 \pm \$0.0020/kWh
capital charges on completion costs	\$0.0275 \pm \$0.0045/kWh
capital charges on \$1 billion acquisition cost	\$0.0111 \pm \$0.0012/kWh

Total (range is square root of sum of squares)	\$0.063 \pm \$0.006/kWh

The range of \$0.063 \pm \$0.006/kWh is to be compared with our estimate from Chapter 3 of \$0.050 \pm \$0.015/kWh for the avoided costs associated with baseload electricity generation in connection with plutonium disposition in new plants. Combining the ranges based on the square root of the sum of the squares gives an expected net cost of \$0.013 \pm \$0.016/kWh, or, stated another way, our 70-percent judgmental confidence interval for the net economic effect of the use of WNP-1 or WNP-3 for WPu disposition under the indicated assumptions, on a levelized-annualized basis, extends from a profit of \$0.003/kWh to a loss of \$0.029/kWh. Over 30 years of operation at 75-percent capacity factor the range extends from a profit of \$0.7 billion to a loss of \$7.1 billion. (These are sums of constant-dollar annual figures; the corresponding net present values at start of reactor operation, in 1992 dollars, extend from a profit of \$0.3 billion to a loss of \$2.9 billion.)

If it were taken into account that gas-fired combined-cycle generation costs in the Northwest (SAIC 1993a) appear to be at the low end of the range cited in Chapter 3, however, the result would tilt even farther toward the net cost side. If the 70-percent confidence range on the avoided cost were $\$0.04 \pm \$0.01/\text{kWh}$, for example, the 70-percent confidence range on the net cost per kilowatt-hour would be $\$0.023 \pm \$0.011/\text{kWh}$, and the range of net present values as of reactor startup would extend from a loss of \$1.2 billion on the low end to a loss of \$3.5 billion on the high end. Similarly, an increase of acquisition cost from \$1 to \$1.2 billion would pull the optimistic end of the 70-percent confidence range down from a modest profit to bare break-even (for the $\$0.050 \pm \$0.015/\text{kWh}$ avoided-cost range), while the acquisition of the reactor for nothing would make the 70-percent confidence range for the net cost of the operation approximately symmetric around the break-even point (net present value at reactor startup from minus \$1.6 billion to plus \$1.6 billion for the $\$0.050 \pm \$0.015/\text{kWh}$ avoided-cost range).

If the high residual plutonium content in spent fuel associated with a 6.8-percent initial plutonium loading were considered problematic, so that both reactors needed to be used in order to complete the disposition campaign in 30 years or less, the up-front costs of the option would roughly double, but the per-kilowatt-hour costs would fall because of economies in scale in MOX fabrication. At 4-percent initial plutonium loading and 42-MWd/kgHM average burnup, it would take about 50 reactor-years to process 50 tons of plutonium, hence 25 years each for the two WNP reactors. If the reactors were financed over the 25-year period, the range of expected costs of the operation would be $\$0.0615 \pm \$0.0039/\text{kWh}$ and the differential over the assumed avoided cost of baseload electricity generation would be $\$0.0115 \pm \$0.0155/\text{kWh}$, or $\$189 \pm \255 million/yr for the two-reactor system. The discounted present value at start of reactor operation of a 25-year stream of costs or revenues of this magnitude would range from a \$0.8-billion profit to a \$5.2-billion loss. If the two reactors were assumed to operate for the remaining 5 years of a nominal 30-year lifetime using LEU fuel, the levelized net income from the operation of the reactors in this period would be $\$0.030 \pm \$0.015/\text{kWh}$; the effect of the resulting 5-year profit stream at the end of the 30-year period on the discounted present value of the enterprise at the start of reactor operation would improve the profit by about \$550 million (to some \$1.3 billion) if the electricity value is at the high end of the range and would reduce the loss by about \$185 million (to about \$5.0 billion) if the electricity value is at the low end of the range.

We note, finally, that the somewhat complex institutional and financial arrangements put forward in the Isaiah proposal appear at first glance to yield a more favorable financial picture than portrayed above for one crucial reason: it is assumed in that proposal that the federal government pays all of the fuel-cycle costs and all of the nonfuel operation and maintenance costs for the reactors, exclusive of the O&M costs of the turbine generator. Taking the turbine genera-

tor O&M costs to be about 15 percent of the fuel plus nonfuel O&M costs for the plant (SAIC 1993b), this means—in the case of our example—that the government would be paying $\$0.020 \pm \$0.002/\text{kWh}$ of the levelized-annualized costs of the operation while the various investors would be paying $\$0.043 \pm \$0.003/\text{kWh}$. The discounted present value in 1992 dollars, at the start of reactor operation, of the government's 30-year cost stream in this one-reactor case would be, under the central estimate, about \$2.0 billion—which is just the amount that the Isaiah Project proposes to provide the government.

Building New Reactors for Plutonium Disposition

In Phase I of the U.S. Department of Energy's Plutonium Disposition Study, carried out in 1992 and 1993 (USDOE 1993b), the only options considered were those involving construction of new reactors of evolutionary and advanced types. These would be dedicated to the task of WPu disposition, although they would also generate electricity—the sale of which would defray the costs of the activity. Five reactor types that could be used for this purpose were examined in detail in the Phase I PDS, based on studies commissioned from the companies that would provide the reactors if one of them were selected for the plutonium-disposition mission: the ABB-Combustion Engineering System-80+, an evolutionary pressurized-water reactor with net electrical output of 1,256 MWe per reactor (ABB-CE 1993); the General Electric Advanced Boiling-Water Reactor (ABWR), an evolutionary BWR with net electrical output of 1,300 MWe per reactor (GE 1993); the Westinghouse PDR-600, an advanced PWR with net electrical output of 610 MWe per reactor (Westinghouse 1993); the General Atomics Modular High-Temperature Gas-Cooled Reactor (MHTGR), with net electrical output of 169 MWe per reactor (GA 1993); and the General Electric/Argonne National Laboratory Advanced Liquid-Metal Reactor (ALMR), with net electrical output of 303 MWe per reactor (GE-ANL 1993).

These contractor studies included detailed economic assessments, which were supposed to follow a set of ground rules provided in advance by DOE. The Technical Review Committee (TRC) set up by the DOE to evaluate and compare the contractor studies found that the ground rules were not always consistently followed. Problems mentioned by the TRC included (USDOE 1993a, pp. SC6-1/30):

- understatement (by all of the contractors) of site-support/infrastructure costs;
- understatement (by all of the contractors) of preoperational costs, which include research and development and conceptual design, safety and environmental impact analyses, operational readiness review, plant startup and testing, and more;
- “overly aggressive” schedule assumptions (by all of the contractors);

- “overly aggressive” assumptions about cost reductions from learning in connection with construction of multiple units (by some contractors);
- widespread understatement of cost and schedule risks associated with licensing of MOX fuel fabrication facilities and of reactors for MOX use;
- nonuniform treatment of indirect costs;
- contingency factors lower in some cases than specified in the ground rules;
- incomplete and nonuniform estimates of the costs of treatment of the waste streams from fuel fabrication, and of costs of interim spent fuel storage;
- nonuniform assumptions about whether WPU feed would be provided as metal or as oxide;
- nonuniform treatment of seismic considerations; and
- nonuniform treatment of dismantling and disposal (D&D) costs.

Almost without exception, the effects of these deficiencies were to understate the costs of the options in question.

The TRC produced a set of modified cost estimates and comparisons in which some—but far from all—of the above-mentioned deficiencies were corrected (mainly, the variations from guidelines in contingency factors and indirect costs, and some but not all of the overoptimism about schedule and learning). As noted in Chapter 3, however, the TRC’s modified economic assessment still suffered from use of an inappropriately low interest rate, failure to account for interest during construction, and frequent lack of clarity about which deficiencies in the contractor studies had been remedied and how.

These shortcomings of the TRC report were criticized in the Peer Review Report (USDOE 1993b) commissioned by DOE as part of the PDS. The Peer Review Report also criticized the TRC report for inadequate analysis of cost uncertainties, insufficient attention to the implications of schedule differentials between more advanced and less advanced reactor types, failure to make comparisons with the economic costs of using existing reactors for the plutonium disposition mission, and failure to compare the costs of electricity generation using plutonium versus uranium fuels. While the Peer Review Report was criticizing a version of the TRC report different from the one publicly released, the panel believes that many of these criticisms apply to the final report as well. Phase II of the PDS—in which, it is to be hoped, some of these shortcomings will be remedied—is reported to be essentially complete, but was not available at the time the panel finished its work.

In the meantime, we have prepared our own set of economic comparisons for the new-reactor options, basing these in part on the fuel-cycle analyses and direct and indirect cost estimates of the Phase I PDS contractor studies as modified by the TRC report. In so doing, we have attempted to improve on the PDS

contractor and TRC reports in a number of respects, including development and consistent treatment of uncertainty ranges, use of a more appropriate real cost of money, consistent treatment of preoperational costs, inclusion of interest on construction and on other preoperational costs, and, in general, adherence to the generally accepted practices described in Chapter 3 and DOE guidelines (USDOE 1988b, Delene and Hudson 1993) for economic evaluation of projects in terms of levelized-annualized costs and net discounted present values.²⁵

Our cost calculations for the five types of reactors considered in the contractor studies and the TRC report are based on reactor specifications and operating modes summarized in Table 6-14. The contractor studies and TRC report examined a variety of operating modes for each reactor—corresponding to “spiking,” “spent fuel,” and “destruction” variants, all subject to the very severe constraint that 100 tons of WPu be processed within 25 years of a program start date taken to be January 1, 1994. We do not consider this constraint a useful basis for a comparative evaluation; it is virtually a prescription for unrealistic deployment schedules for advanced-reactor options, which under realistic schedules would not meet the constraint. Instead, we evaluated, for each of the five reactor types, the economics of a reactor array with sufficient capacity to process 50 tons of WPu within approximately one nominal reactor lifetime of 30–40 years from the start of reactor operation, using the contractor-analyzed fuel-cycle variant that we considered most representative of that reactor type’s near-term plutonium disposition capabilities. (We chose the highest burnup variants that used the reactor type’s “standard” fuel, which were the “spent fuel” variants for the PDR-600, System-80+, and ALMR, and the “destruction” variants for the ABWR and MHTGR.)

Our cost calculations are based, moreover, on reactor operational lifetimes coinciding exactly with the period of operation required to irradiate 50 tons of WPu to the average burnups indicated. (These time figures were derived using the simplifying assumption that fuel shuffling is used in the beginning and ending phases of reactor operation in order to achieve the indicated average burnup for all of the plutonium-based fuel, without introducing any uranium-based fuel.) As can be seen in Table 6-14, the operating times actually required—under the stated assumptions about plutonium percentages in spent fuel, average burnups, numbers of reactors of a given type, and lifetime-average capacity factors—range from 28 to 36 years.

In contrast, then, to the approach of DOE’s TRC, which assumed that the plants would complete an assumed 40-year operating lifetime (running on uranium fuel for the part of that period after WPu disposition has been completed,

²⁵ We have omitted, however, a substantial amount of analytical complexity in the form of aspects of electric-utility accounting practices that depend on debt-to-equity ratios, accelerated depreciation schemes, tax policy, and allocation of electricity value between capacity credit and energy credit, since these context-specific complications add much speculation and calculation, but little value in a general comparison of the sort undertaken here.

and earning additional revenues from electricity generation during this period of uranium-fueled operation), we have assumed in the calculations for the table that the plants do not operate after the plutonium disposition mission has been completed. Accordingly, our fixed charge rates for determining levelized-annualized capital charges are based on just the operating lifetimes needed to complete that mission.

There are several reasons for choosing this approach for our baseline economic comparison of the five options: it removes the unfortunate characteristic of the TRC approach that economic benefits are attributed to plutonium disposition in proportion to the amount of time that reactors operate with nonplutonium fuel; it reduces the sensitivity of the results to highly uncertain assumptions about distant-future electricity prices; it avoids uncertainty about whether these reactor types would in fact be able to achieve an operating life as long as 40 years; and it reflects the possibility that society might choose, for economic or other reasons, not to use the plutonium disposition reactors for electricity generation once the plutonium mission has been completed. At the end of this section, however, we do show how the results would change under the TRC approach of assuming continuing uranium-based electricity generation to a total operating life of 40 years.

The cornerstones of a calculation of the monetary cost of any reactor option are the estimates of direct plus indirect construction costs, of fuel-cycle costs, and of nonfuel O&M costs. Rather than accepting the point-value estimates for these quantities provided in the PDS—and accepting along with these estimates the inevitable doubts about whether the different contractors derived these estimates in comparable ways, with comparable degrees of optimism or conservatism—we have used the PDS estimates together with cost estimates for these fuel cycles and reactor types from other sources to develop ranges of values for use in our own economic calculations. Central estimates and ranges for MOX fuel-cycle costs were obtained in this way in “Weapons Plutonium Versus Uranium as Power Reactor Fuel” above. Here we do so for the construction and nonfuel O&M costs and for the fuel-cycle costs in the non-MOX cases.

Because our choices about the time frame for plutonium disposition, and hence the scale of the facilities needed to accomplish it, are different than those employed in DOE’s PDS, we have had to adjust the cost estimates provided by the PDS contractors and the TRC, in some instances, to apply them to different numbers of reactors. In contrast to the extremely optimistic “learning” assumptions applied by some of the contractors to multiple-reactor systems, we assume (consistent with the TRC) that the overnight construction costs and nonfuel O&M costs of such systems increase with the 0.9 power of the number of reactors (costs per reactor decline with the -0.1 power of the number of reactors). For the non-MOX fuel fabrication facilities (for the MHTGR and ALMR) analyzed here, we assume as before that total costs in a facility of a given type scale

TABLE 6-14 New-Reactor Specifications and Reference Operating Modes

Characteristics	Evolutionary BWR (GE ABWR) ^a	Evolutionary PWR (ABB-CE System- 80+) ^b	Advanced PWR (Westing- house PDR-600) ^c	Advanced Gas Reactor (GA MHTGR) ^d	Advanced Metal Reactor (BE/ANL ALMR) ^e
System Performance					
Thermal power, MWt/reactor	3,926	3,817	1,940	450	840
Net electric power, MWe/reactor	1,300	1,256	610	169	303
Conversion efficiency	0.331	0.329	0.314	0.376	0.361
Assumed capacity factor	0.75	0.75	0.75	0.79	0.75
Reactor output, 10 ⁹ kWh/yr	8.55	8.26	4.01	1.17	1.99
Operating Mode					
Pu as percent of heavy metal	3.0	6.8	5.5	100.0	10.5
Average burnup, MWd/kgHM	37.1	42.2	40.0	580.0	69.1
Mean fuel life, yr	5.3	4.0	5.0	2.0	6.0
Fuel loaded, MTHM/reactor-yr	29.0	24.8	13.3	0.2	3.3
Pu loaded, kgPu/reactor-yr	870	1,685	731	224	350
Array Characteristics					
Reactors × yrs for 50 MT Pu	2 × 29	1 × 30	2 × 34	8 × 28	4 × 36
Array capacity, MWe	2,600	1,256	1,220	1,352	1,212
Array output, 10 ⁹ kWh/yr	17.09	8.26	8.02	9.36	7.97
Fuel fabrication requirement MTHM/yr	58.0	24.8	26.6	1.8	13.3
Pu loaded, kgPu/yr	1,739	1,685	1,461	1,791	1,399

^a GE ABWR: General Electric Advanced Boiling-Water Reactor. In destruction option, 26.6 percent of core amounting to 41.1 MTHM is discharged every 17 months at an average irradiation of 37,090 MWd/MTHM. Fuel life is $(17/12) / 0.266 = 5.3$ years and annual plutonium loading is $0.030 \text{ kgPu/kgHM} \times (12/17) \times 41,100 \text{ kgHM/reactor-yr} = 870 \text{ kgPu/reactor-yr}$.

^b ABB-CE System-80+: ABB-Combustion Engineering System-80+. In spent fuel option, full core amounting to 99.2 MTHM is discharged every 48 months at an average irradiation of 42,200 MWd/MTHM (achieved by fuel shuffling annually). Fuel lifetime is 4.0 years and annual plutonium loading is $0.068 \text{ kgPu/kgHM} \times 99,200/4 \text{ kgHM/reactor-yr} = 1,686 \text{ kgPu/reactor-yr}$.

^c Westinghouse PDR-600: In spent fuel option, one-third of core amounting to 22.15 MTHM is discharged every 20 months at an average irradiation of 40,000 MWd/MTHM. Fuel life is thus $(20/12)/0.333 = 5.0$ years, annual fuel fabrication requirement is $22.15 \text{ MTHM/cycle} / (1.667 \text{ yr/cycle}) = 13.29 \text{ MTHM/reactor-yr}$, and annual plutonium loading at $0.055 \text{ kgPu/kgHM} \times 13,290 \text{ kgHM} = 731 \text{ kgPu/reactor-yr}$.

^d GA MHTGR: General Atomics Modular High-Temperature Gas-Cooled Reactor. In destruction option, 50 percent of core amounting to 224 kgHM irradiated to an average of 580,000 MWd/kgHM is discharged annually. Fuel lifetime is 2 years and plutonium loading is $224 \text{ kgHM/reactor-yr}$ since heavy metal in this fuel is 100 percent plutonium. The indicated 79-percent capacity factor for this reactor is the value assumed by GA in its analyses; the panel chose not to redo the analysis for the 75-percent capacity factor assumed for the other reactor types. After most of our economic analyses were completed, GA proposed a modified high-temperature gas reactor known as the modular helium reactor (MHR) for the plutonium disposition mission. This modification included the use of a direct (Brayton) cycle in which the high-temperature helium is sent directly to the turbine, eliminating the need for a steam generator. This cycle is estimated by the vendor to lead to a 25-percent increase in thermal efficiency, as well as reducing the capital cost of the reactor. The vendor also proposes to increase the size of the reactor to 600 MWt, further reducing the cost per megawatt. We have not used, for our analysis, the revised cost estimates provided by GA for this modification. The direct-cycle HTGR may prove to meet the vendor's expectations, but we think the information now available is too preliminary to support our using the revised

figures. Licensing review, in particular, will be more difficult for the new design, as the vendor's own analysis concedes (GA 1994, p. 13-1).

^c GE/ANL ALMR: General Electric/Argonne National Laboratory Advanced Liquid-Metal Reactor. In spent fuel option, one-third of core fuel assemblies amounting to 3,480 kgHM are discharged every two years at an average irradiation of 106,300 MWd/MTHM and one-fourth of blanket fuel assemblies amounting to 3,185 kgHM are discharged every two years at an average irradiation of 28,300 MWd/MTHM. Fuel life for core fuel is $2/0.333 = 6$ years. Plutonium is loaded only into core fuel at a rate of $0.202 \text{ kgPu}/(\text{kgHM in core fuel}) \times 1,740 \text{ kgHM core fuel}/\text{reactor-yr} = 351 \text{ kgPu}/\text{reactor-yr}$. Plutonium loading averaged over core and blanket fuel is $351 \text{ kgPu}/\text{yr} / (3,480 \text{ kgHM} + 3,185 \text{ kgHM}) / (2 \text{ yr}) = 0.105 \text{ kgPu}/\text{kgHM}$. Average irradiation at discharge for all fuel is $(840 \text{ MW} \times 365.25 \text{ days}/\text{yr} \times 0.75) / (3,480 \text{ kgHM} + 3,185 \text{ kgHM}) / (2 \text{ yr}) = 69.05 \text{ MWd}/\text{kgHM}$.

with the 0.6 power of output, so that unit costs decline with the -0.4 power of output.

The resulting central estimates and ranges for direct plus indirect costs of power-plant construction, as well as nonfuel O&M costs, are presented and compared to the PDS point estimates in Table 6-15. In using estimates from the literature to develop our construction-cost ranges, we have applied where appropriate an increment of 20 percent on direct plus indirect costs for first-of-a-kind costs, since the new reactors under consideration would indeed be first-of-a-kind if built for the immediate plutonium disposition need; the 20-percent figure is based on arguments and examples given in DOE costing-guideline documents (USDOE 1988b, Delene and Hudson 1993). The ranges in the table have been chosen to extend a uniform ± 20 percent on construction costs for the evolutionary reactors, ± 25 percent on construction costs for the advanced PWR, ± 30 percent on construction costs for the MHTGR and ALMR, and ± 15 percent on operating costs for all reactor types; these are our judgmental 70-percent confidence intervals. Further details on the derivations of the estimates are given in the notes to the table.

The estimates of direct plus indirect power-plant construction costs and nonfuel O&M costs in Table 6-15, plus the plutonium-disposition-array characteristics from Table 6-14 and the estimates of MOX-fuel-associated costs developed in the preceding section and summarized in Table 6-10, provided the starting point for comparative economic calculations we carried out in spreadsheets developed for the purpose and which are summarized in Table 6-16. The subparagraphs that follow here provide further information on assumptions and parameter choices used in those calculations. (A more detailed discussion of the concepts and methods relevant to economic comparisons of this sort was provided in "Issues and Criteria in Economic Evaluation of Alternatives" in Chapter 3. Further details specific to the individual reactor types are provided in notes to the table.)

(text continues on p. 320)

TABLE 6-15 Panel Estimates of Construction and O&M Costs for New Reactors (in 1992 U.S. dollars)

New Reactors	Direct + Indirect Construction Costs, \$/kWe		Nonfuel O&M Costs, Million \$/GWe Capacity per Year	
	TRC Estimate	Panel Estimate with Range	TRC Estimate	Panel Estimate with Range
Evolutionary BWR, ^a 2 × 1,300 MWe	1,805	1,800±360	43	75±11
Evolutionary PWR, ^b 1 × 1256 MWe	2,246	1,900±380	74	80±12
Advanced PWR, ^c 2 × 610 MWe	1,824	2,100±525	88	90±14
Advanced Gas Reactor, ^d 8 × 169 MWe	2,063	2,400±720	78	85±13
Advanced Metal Reactor, ^c 4 × 303 MWe	2,311	2,500±750	92	90±14

NOTES: TRC estimates have been adjusted for scale as described in the footnotes. Two- to four-digit precision is illusory but has been preserved to facilitate checking results.

^a Evolutionary boiling-water reactor: PDS TRC estimates direct plus indirect capital costs at \$2,515.0 million for one 1300-MWe power plant (USDOE 1993a, p. A-22). Assuming costs scale with the 0.9 power of the number of units, this corresponds to $\$2,515 \times 2^{0.9} = \$4,693$ million for two units, or $\$4,693 \text{ million} / 2.6 \text{ GWe} = \$1,805/\text{kWe}$. In the absence of other estimates for an ABWR per se, we refer to an estimate of the Fission Working Group Review Committee of the PDS (Omberg and Walter 1993, p. 23) of \$1,600/kWe direct plus indirect capital costs, excluding first-of-a-kind costs, for a generic evolutionary LWR, and to a Nuclear Energy Cost Data Base (NECDB) (USDOE 1988b, p. 26) figure of \$1,147/kWe in 1987 dollars, hence \$1,394/kWe in 1992 dollars, for an 1,100-MWe evolutionary PWR. Based on examples given in DOE costing-guideline documents (USDOE 1988b, Delene and Hudson 1993), we take first-of-a-kind costs, as would apply to the WPU disposition case we are considering, to be an increment of 20 percent on the indicated direct plus indirect costs; applying this factor and the two-unit, unit-cost learning factor of $1/2^{0.1}$ makes the two indicated estimates \$1,560/kWe and \$1,790/kWe for our 2 × 1,300 MWe case. We take as our central estimate \$1,800/kWe, with a judgmental 70-percent confidence interval of ±20 percent, hence ±\$360. The TRC estimates nonfuel O&M costs less D&D annuity at \$60.4 million/reactor-yr for one reactor, which becomes $\$60.4 \text{ million} \times 2^{0.9} = \112.7 million/yr for a two-reactor plant, or \$43 million/GWe/yr. The NECDB (USDOE 1988b, p. 32) gives nonfuel O&M costs for an 1,100-MWe conventional LWR as \$62 million/yr fixed and \$0.0005/kWh variable in 1987 dollars, and says this figure is assumed applicable to evolutionary and advanced plants as well. Converting to 1992 dollars and assuming the fixed part scales with capacity, the relation for a 1,300-MWe plant would be \$88 million/yr + \$0.0006/kWh, or at 75-percent capacity factor \$93 million/yr; with a 25-percent increment as appropriate on nonfuel O&M costs for first-of-a-kind (USDOE 1988b, Delene and Hudson 1993), this would be \$116 million/yr, or for two plants $\$116 \text{ million/yr} \times 2^{0.9} = \216 million/yr, hence \$83 million/GWe/yr. Even allowing for reduced labor requirements in an evolutionary plant, the GE/TRC estimate seems very low. Our central estimate is \$75 million/GWe/yr, with a judgmental 70-percent confidence interval of ±15 percent or ±\$11 million/yr.

^b Evolutionary pressurized-water reactor: The PDS TRC estimates direct plus indirect capital costs at \$2,820.9 million for one 1,256-MWe power plant (USDOE 1993a, p. A-10), or \$2,246/kWe. The contractor estimate of indirect + indirect costs submitted to the PDS for this case (ABB-CE 1993, p. VI-14) was \$2,348 however, or \$1,869/kWe, and the basis on which the TRC increased the estimate is not clearly explained in the TRC report. The estimates in the NECDB and the report of the Fission Working Group Review Committee, mentioned above, would become

\$1,920/kWe and \$1,673/kWe, respectively, for this case after the addition of 20 percent first-of-a-kind costs. One expects a PWR to be a bit costlier than a BWR, all else being equal. We take as our central estimate \$1,900/kWe, with a judgmental 70-percent confidence interval of ± 20 percent, hence $\pm \$380$. The PDS contractor estimate for nonfuel O&M costs in the spent fuel mode are \$92.8 million/reactor-yr without the D&D annuity (ABB-CE 1993, p. VI-24), hence \$74 million/GWe/yr. The NECDB estimate would be the same as for the evolutionary BWR, hence, with 25 percent first-of-a-kind increment, \$116 million/reactor-yr or \$92 million/GWe/yr. Our central estimate is \$80 million, with judgmental 70-percent confidence interval of ± 15 percent or $\pm \$12$ million/yr.

^c Advanced pressurized-water reactor: TRC estimates direct plus indirect costs at \$1192.6 million for one 610-MWe power plant (USDOE 1993a, p. A-1). Assuming costs scale with the 0.9 power of the number of units, this corresponds to $\$1,192.6 \text{ million} \times 2^{0.9}$ for two units, or \$2,225.5 million / 1.22 GWe = \$1,824/kWe. The NECDB (USDOE 1988b, p. 26) estimates direct plus indirect costs for a 550-MWe APWR at \$743 million, hence \$1,351/kWe in 1987 dollars or \$1,630/kWe in 1992 dollars; with first-of-a-kind costs of 20 percent and unit costs lower by $2^{0.1}$ for two units, this estimate would be \$1,825/kWe. Despite the close coincidence of this number and that of the PDS, we think it rather unlikely that cost-saving advances between evolutionary and advanced PWRs will offset the diseconomies of scale suffered by 600-MWe units in comparison to 1,200-MWe ones. Thus we take as our central estimate \$2,100/kWe, with a judgmental 70-percent confidence interval of ± 25 percent or $\pm \$525$. The contractor estimates nonfuel O&M costs less D&D annuity as \$57.3 million/yr for one reactor (Westinghouse 1993, p. 3-19), hence \$57.3 million $\times 2^{0.9}$ = \$106.9 million/yr for a two-unit plant, or \$88 million/GWe/yr. The NECDB estimates nonfuel O&M costs for a 2×550 -MWe advanced LWR at \$71 million/yr fixed and \$0.0005/kWh variable in 1987 dollars; adjusting for 2×610 MWe and 1992 dollars makes this \$95 million + 0.0006/kWh, which at 75-percent capacity factor is \$100 million/yr or \$82 million/GWe/year; with a 25-percent first-of-a-kind increment, this becomes \$102.5 million/GWe/yr. Our central estimate is \$90/GWe/yr, with judgmental 70-percent confidence interval ± 15 percent or $\pm \$14$.

^d Advanced gas reactor: The PDS TRC estimates direct plus indirect costs at \$2,789.4 million for an eight-module, 1,352-MWe power plant (USDOE 1993a, p. A-37), or \$2,063/kWe. The Fission Working Group Review Committee estimates direct plus indirect costs for the MHTGR at \$2,200/kWe, which a first-of-a-kind increment of 20 percent would increase to \$2,640/kWe. There is no MHTGR estimate in the 1988 NECDB, but a late 1980s DOE-sponsored comparison of advanced fission and fusion reactors (Holdren et al. 1989), which used NECDB methods to generate its cost estimates, produced a direct plus indirect cost estimate for an eight module MHTGR that translates to \$1,786/kWe in 1992 dollars, which with a 20-percent first-of-a-kind increment would be \$2,143/kWe. We take as our central estimate \$2,400/kWe, with a judgmental 70-percent confidence interval of ± 30 percent or $\pm \$720$. The contractor estimate for nonfuel O&M costs for the eight-module case (GA 1993, Table A-7) is \$105.9 million/yr or \$78.3 million/GWe/yr. The NECDB contains no estimate for MHTGR operating costs. The fusion/fission study estimated nonfuel O&M costs for an eight-module MHTGR at \$0.0083/kWh in 1986 dollars, hence \$0.0103/kWh in 1992 dollars, which translates at the assumed capacity factor of 0.79 to \$96.4 million/yr or \$71.3 million/GWe/yr; with 25-percent first-of-a-kind increment this is \$89.1 million/GWe/yr. Our central estimate is \$85 million/GWe/yr, with a judgmental 70-percent confidence interval of ± 15 percent or $\pm \$13$ million.

^e Advanced metal reactor: PDS estimates direct plus indirect costs at \$1,501 million for a two-reactor module, 606-MWe power plant (USDOE 1993a, p. A-28). Assuming costs scale with the 0.9 power of the number of units, this corresponds to $\$1,501 \text{ million} \times 2^{0.9}$ = \$2,801 million for two such plants, or \$2,801 million / 1.212 GWe = \$2,311/kWe. The Fission Working Group Review Committee estimates direct plus indirect costs for the ALMR at \$2,100/kWe, which a first-of-a-kind increment of 20 percent would increase to \$2,520. The NECDB direct plus indirect cost estimate for an 1,100-MWe LMR is \$1,988/kWe in 1987 dollars, or \$2,398/kWe in 1992 dollars, which with a 20-percent first-of-a-kind increment would be \$2,878. (This LMR is not of the modular design considered here.) The fission/fusion study estimated modular and conventional LMRs of 1,200- to 1300- MWe total capacity to have direct plus indirect costs nearly identical at \$1,700/kWe (after translation to 1992 dollars), or \$2,040/kWe after addition of a 20-percent first-of-a-kind increment. We take as our central estimate \$2,500/kWe, with a judgmental 70-percent confidence interval of ± 30 percent or $\pm \$750$. The TRC estimate of nonfuel O&M costs less D&D annuity is \$60.0 million/yr for a two-reactor power block, hence $\$60.0 \text{ million/yr} \times 2^{0.9}$ = \$112

million/yr for a four-reactor plant, or \$92.4 million/GWe/yr. The NECDB estimate of LMR operating costs is \$67 million/yr fixed plus \$0.0006/kWh variable for an 1,100-MWe (nonmodular) unit, in 1987 dollars, hence \$86 million/yr in 1992 dollars at 75-percent capacity factor, or \$71 million/GWe/yr; with 25-percent first-of-a-kind increment this is \$89 million/GWe/year. The fission/fusion study's estimated nonfuel O&M costs for a modular LMR similar in size to our case were \$0.0079/kWh in 1986 dollars, hence \$0.0098/kWh in 1992 dollars, translating to \$78 million/yr or \$64 million/GWe/yr; with 25-percent first-of-a-kind increment this is \$80 million/GWe/yr. Our central estimate is \$90 million/GWe/yr, with a judgmental 70-percent confidence interval of ± 15 percent of $\pm \$14$.

Constant Dollars

All figures are reported in January 1, 1992 constant dollars.

Contingency

The TRC for the DOE's Plutonium Disposition Study applied a contingency of 0.20 of direct plus indirect costs of a power plant's nuclear equipment and 0.15 of direct plus indirect costs of the ECA (Energy Conversion Area) for the evolutionary systems; 0.25 of nuclear reactor direct plus indirect costs and 0.20 of direct plus indirect costs of the ECA for the advanced systems; and 0.25 of direct plus indirect costs for fuel fabrication plants for both evolutionary and advanced systems. Since many sources of power-plant cost estimates do not disaggregate nuclear equipment from the ECA, we assume a typical 70–30 percent division so that the combined contingency, using the same values as the TRC, is 0.185 for evolutionary systems and 0.235 for advanced systems.

Preoperational Costs

To replace the drastically divergent treatment of preoperational costs of different options in the contractor studies and the TRC report, we have used a simple rule of thumb that approximates these costs as the following fractions of overnight construction costs: 10 percent for evolutionary reactor systems, 20 percent for the fuel fabrication plants for LWRs (whether current, evolutionary, or advanced), and 25 percent for advanced reactor systems²⁶ and for the fuel fabrication plants for the MHTGR and ALMR.

²⁶ In the interest of simplicity, no distinction has been made in reactor-system contingency or preoperational costs, expressed as a percentage of direct plus indirect construction costs, between the advanced LWR and the other advanced reactor types for which detailed estimates were made, the MHTGR and the ALMR. It can be argued, however, that the advanced LWR is based on more proven technology and that this would be likely to be reflected in smaller contingency and preoperational costs. Incorporating such a distinction would not significantly affect any of the conclusions drawn.

Interest During Construction

We have assumed an S-curve distribution of construction and preoperational costs over a period of nine years preceding full operation in the case of reactors and advanced fuel fabrication plants. At the OMB-recommended real annual interest rate of $r = 0.07$, this procedure gives IDC equal to 41 percent of overnight costs (see section in Chapter 3 “Issues and Criteria in Economic Evaluation of Alternatives”).

Fixed Charge Rates

These are computed according to the prescriptions described in Chapter 3, with $r = 0.07$ and depreciation periods equal to the operating lifetimes shown in Table 6-14 for irradiation of 50,000 kg of plutonium, both with and without an allowance of 0.02/yr of total beginning-of-life capital costs for property taxes and insurance.

Dismantling and Disposal Costs

For D&D costs of reactors we use contractors' estimates where available and otherwise use the DOE guideline (see “Principles and Pitfalls in Cost Comparisons” in Chapter 3) whereby end-of-life D&D cost = $B + 0.02(P-1,200)$ million 1992 dollars where P is the unit thermal power and B is 145 for PWRs, 185 for BWRs, and 165 for other reactor types. For fuel fabrication plants we use the DOE guideline suggested in the PDS (USDOE 1993a, p. SC6-7) whereby the end-of-life D&D cost is taken as 10 percent of the beginning of life capital cost (direct plus indirect plus contingency), not including IDC. In each case the annual payment needed during operation in order to produce the indicated sum at the end of operating life is computed (assuming the payments are invested at 3 percent per year real) and added to the operating costs.

Fuel Costs

Estimated costs of MOX fuel in \$/kgHM for the evolutionary and advanced LWRs are taken from Table 6-10 for the three cases presented there—fabrication at FMEF, paying no property tax or insurance, and fabrication at new plants both with and without property taxes and insurance—at the scales most closely corresponding to the plutonium disposition arrays described in Table 6-14, and corrected for plutonium metal-to-oxide conversion and incremental transport and storage costs where the plutonium loading differs from the reference loading of 4.8 percent used in deriving the Table 6-10 figures. For the MHTGR and ALMR, fuel costs per kilogram of heavy metal are derived starting

TABLE 6-16 Costs of Plutonium Disposition with Evolutionary and Advanced Reactors (in 1992 U.S. dollars)

	Evolutionary BWR (GE ABWR)	Evolutionary PWR (ABB-CE System 80+)	Advanced PWR (Westing- house PDR 600)	Advanced Gas Reactor (GA MHTGR)	Advanced Metal Reactor (GE/ANL ALMR)
Array Characteristics					
Reactors \times net MWe	2 \times 1,300	1 \times 1,256	2 \times 610	8 \times 169	4 \times 303
Array annual TWh	17.09	8.26	8.02	9.36	7.97
MTHM/yr	58.0	24.8	26.6	1.8	13.3
kgPu/yr	1,739	1,685	1,461	1,791	1,399
yr for 50 MTPu	29	30	34	28	36
Power-Plant Construction Costs					
Array dir+indir, \$/kWe	1,800 \pm 360	1,900 \pm 380	2,100 \pm 525	2,400 \pm 720	2,500 \pm 750
Array overnight, \$/kWe	2,133	2,252	2,594	2,964	3,088
Array w IDC, \$/kWe	3,008	3,175	3,657	4,179	4,353
Array w IDC, M\$	7,820	3,987	4,461	5,650	5,276
Array pre-op, M\$	782	399	1,115	1,413	1,319
Array total, M\$	8,602 \pm 1720	4,386 \pm 877	5,577 \pm 1,394	7,063 \pm 2,119	6,595 \pm 1,979
Power-Plant Array Capital Charges					
FCR, no tax, yr ⁻¹	0.0814	0.0806	0.0778	0.0824	0.0767
FCR, w tax, yr ⁻¹	0.1014	0.1006	0.0978	0.1024	0.0967
LACC no tax, M\$/yr	701	353	434	582	506
LACC w tax,	873	441	545	723	638
c/kWh no tax	4.10 \pm 0.82	4.28 \pm 0.86	5.41 \pm 1.35	6.22 \pm 1.86	6.35 \pm 1.90
c/kWh w tax	5.10 \pm 1.02	5.34 \pm 1.07	6.80 \pm 1.70	7.72 \pm 2.32	8.01 \pm 2.40
Power-Plant Array Nonfuel O&M Costs					
O&M w/o D&D, M\$/GWe-yr	75 \pm 11	80 \pm 12	90 \pm 14	85 \pm 13	90 \pm 14
O&M w/o D&D, M\$/yr	208	94	110	115	109
Array D&D, M\$	447	197	298	330	326
D&D annuity, M\$/yr	9.9	4.1	5.2	7.7	5.1
O&M w D&D, M\$/yr	218 \pm 31	98 \pm 14	115 \pm 17	123 \pm 17	114 \pm 16
O&M w D&D, c/kWh	1.27 \pm 0.18	1.19 \pm 0.17	1.43 \pm 0.21	1.31 \pm 0.18	1.43 \pm 0.21
MOX Fuel Costs,^a \$/kgHM					
FMEF	1,900 \pm 300	2,500 \pm 400	2,500 \pm 400	NA	NA
New plant, no tax	2,600 \pm 400	3,400 \pm 500	3,400 \pm 500	38,000 \pm 7,600	3,800 \pm 760
New plant, w tax	2,900 \pm 400	3,800 \pm 500	3,800 \pm 500	44,000 \pm 8,800	4,100 \pm 820
Pu conversion correction	-202 \pm 66	284 \pm 90	99 \pm 31	0	0
Adj FMEF ^b	1,799 \pm 302	2,642 \pm 403	2,550 \pm 400	NA	NA
Adj new plant no tax	2,398 \pm 405	3,684 \pm 508	3,499 \pm 501	38,000 \pm 7,600	3,800 \pm 760
Adj new plant w tax	2,698 \pm 405	4,084 \pm 508	3,899 \pm 501	44,000 \pm 8,800	4,100 \pm 820
Fuel Cost Contribution to Electricity Cost, Including Repository Fee					
Carrying-charge factor	1.271	1.219	1.259	1.143	1.219
FMEF, c/kWh	0.88 \pm 0.12	1.07 \pm 0.15	1.16 \pm 0.17	NA	NA
New plant no tax, c/kWh	1.13 \pm 0.17	1.45 \pm 0.19	1.56 \pm 0.21	0.93 \pm 0.17	0.93 \pm 0.17
New plant w tax, c/kWh	1.26 \pm 0.17	1.59 \pm 0.19	1.73 \pm 0.21	1.06 \pm 0.19	0.99 \pm 0.18
Total Levelized Generation Cost, c/kWh					
Reactor no tax, FMEF ^c	6.25 \pm 0.85	6.54 \pm 0.89	8.01 \pm 1.38	8.46 \pm 1.88	8.71 \pm 1.92
Reactor, new plant both tax	7.64 \pm 1.05	8.13 \pm 1.10	9.96 \pm 1.72	10.10 \pm 2.33	10.43 \pm 2.42
Levelized Costs Net of Electricity Revenues @ 5.0\pm1.5 c/kWh, M\$/yr					
Reactor no tax, FMEF ^c	213 \pm 295	127 \pm 144	241 \pm 163	324 \pm 225	296 \pm 194
Reactor, new plant both tax	452 \pm 313	258 \pm 154	398 \pm 183	477 \pm 260	433 \pm 227

	Evolutionary BWR (GE ABWR)	Evolutionary PWR (ABB-CE System 80+)	Advanced PWR (Westing- house PDR 600)	Advanced Gas Reactor (GA MHTGR)	Advanced Metal Reactor (GE/ANL ALMR)
Net Cost of Campaign Discounted to Start of Reactor Operation, billion \$					
Reactor no tax, FMEF ^c	2.6±3.6	1.6±1.8	3.1±2.1	3.9±2.7	3.9±2.5
Reactor, new plant both tax	5.5±3.8	3.2±1.9	5.1±2.4	5.8±3.2	5.6±3.0
Cost Excess Over Same Reactor Using LEU Fuel, c/kWh					
LEU fuel-cycle cost	0.51±0.09	0.61±0.07	0.68±0.08	NC ^d	NC ^d
FMEF	0.37±0.16	0.45±0.16	0.48±0.19	NC ^d	NC ^d
New plant w tax	0.76±0.19	0.98±0.20	1.04±0.22	NC ^d	NC ^d
Cost Excess Over Same Reactor Using LEU Fuel, \$M/yr					
FMEF	63±27	38±14	39±15	NC ^d	NC ^d
New plant w tax	130±33	81±17	84±18	NC ^d	NC ^d
Alternative Net Cost for Campaign, Discounted to Start of Reactor Operation, M\$					
FMEF	777±328	466±169	494±192	NC ^d	NC ^d
New plant w tax	1,590±409	1,006±205	1,075±232	NC ^d	NC ^d

ABBREVIATIONS:

- c/kWh: cents per kilowatt-hour.
- D&D: dismantling and disposal.
- FCR: fixed charge rate.
- LACC: levelized annual capital charge.
- M\$: million dollars.
- O&M: operation and maintenance.
- TWh: terawatt-hour.

NOTES: Based on estimates from Tables 6-10 and 6-15, with reactor characteristics and operating modes as indicated in Table 6-14. See also additional assumptions and disclaimers in the text and notes. Two- to four-digit precision is illusory but has been preserved to facilitate checking results. Uncertainty ranges are the Reactor Panel's judgmental 70-percent confidence intervals; these are not shown for many intermediate quantities to avoid clutter.

^a Includes capital charges, O&M, D&D, plutonium metal-to-oxide conversion, and incremental plutonium storage and transport).

^b Only half of the correction is applied to the FMEF (for operating costs of the additional plutonium conversion but not the capital costs), since the FMEF facility's capital cost includes allowance for a metal-to-oxide conversion operation.

^c FMEF not assumed applicable to MHTGR and ALMR. Entries in this row for those reactors are based on a new fuel fabrication plant that does not pay property tax or insurance.

^d Not calculated because of lack of information on fuel costs with uranium fuel.

from construction-cost and operating-cost estimates provided by the PDS contractors, and employing contingency, preoperational costs, IDC, FCRs, and D&D costs as indicated above. The \$/kgHM fuel costs are then capitalized over an accounting life of n (= core residence time plus one) years, in order to compute cost per kilowatt-hour:

$$\$/kWh = \frac{\$/kgHM \times n \times r(1+r)^n}{\{[(1+r)^n - 1] \times 1000 \times MWd/kgHM \times 24 \times \eta\}}$$

where η is the thermal-to-electric conversion efficiency. With $r = 0.07$ and $n = 4$ years, for example, this becomes,

$$$/kWh = 1.18 \times \$/kgHM / (24,000 \times MWd/kgHM \times \eta) .$$

The statutory repository fee of \$0.001/kWh is then added to this cost per kilowatt-hour.

Total and Net Costs

Total costs per kilowatt-hour, levelized in constant 1992 dollars, are then obtained by summing the power plant capital charges and nonfuel O&M costs (including the annuity to cover power-plant D&D costs) and the fuel costs. The uncertainty bounds are calculated as the square root of the sum of the squares of the bounds on the components of the sum, preserving the notion of a consistent judgmental confidence interval. These are shown just for the high and low bounding cases—on the low side, LWRs paying no property tax or insurance and supplied with MOX from the FMEF, which also pays no property tax or insurance, and, on the high side, reactors and new fuel fabrication plants all paying property tax and insurance.

Net costs on a levelized-annualized basis are obtained by subtracting from the total costs the busbar value of the electricity generated in the course of plutonium disposition, based on a figure of $\$0.05 \pm \$0.015/kWh$ (see “Issues and Criteria in Economic Evaluation of Alternatives” in Chapter 3 and “Completing Existing LWRs” above). The net cost of the campaign figured as a net present value as of the start of reactor operation is then computed by discounting the annualized cost stream at a real interest rate of $r = 0.07/yr$.

Alternate net costs on a levelized-annualized basis and as a net present value at the start of reactor operation reflect the difference between the calculated cost of plutonium-based operation and operation for the same period using LEU fuels (for the MOX-based, evolutionary systems), based on LEU fuel-cycle costs developed in the section “Weapons Plutonium Versus Uranium as Power Reactor Fuel” above.

Total capital investments as of the start of reactor operation are just the sums of the investments in the power plants and the fuel fabrication plants.

Table 6-16 shows that the central estimates for the net costs of the campaign, measured as 1992-dollar net present values as of the start of reactor operation (which of course would be at a more distant time in the future for the advanced reactor types than for the evolutionary ones), range from \$1.6 billion in the most favorable case to about \$5.5 billion in each of three least favorable cases. The 70-percent confidence interval extends into the “profit” range only in two cases, both being associated with the use of FMEF to fabricate MOX fuel for LWRs that do not pay property taxes or insurance. A very substantial advan-

tage is associated with the use of FMEF to do the MOX fuel fabrication for the LWRs; this benefit arises by virtue of the substantial construction costs already invested in this facility, which are treated as sunk. For MHTGRs and ALMRs that do not pay property taxes and insurance, fed by fuel fabrication plants that also do not pay these items, the 70-percent confidence ranges of the net discounted campaign costs as of the start of reactor operation are \$1.2–\$6.6 billion and \$1.4–\$6.3 billion, respectively.

For reactors that do pay property taxes and insurance, fed by new fuel fabrication plants that also pay these items, the 70-percent confidence ranges of the net discounted campaign costs for the five cases are, respectively, \$1.7–\$9.3 billion for the ABWR, \$1.3–\$5.1 billion for the evolutionary PWR, \$2.7–\$7.5 billion for the advanced PWR, \$2.6–\$9.0 billion for the MHTGR, and \$2.6–\$8.6 billion for the ALMR. It is difficult to ascribe much significance to the differences among these cases. Within the considerable limitations of available basic cost estimates and the procedures we have used to try to treat uncertainties in a consistent manner, however, a rather robust overall conclusion is that the net campaign costs in 1992 dollars, discounted to the start of reactor operations, for all new facilities paying property taxes and insurance, seem likely to be in the range of \$2–\$9 billion.

The three clusters of entries at the end of Table 6-16 show, in addition, the alternative measure of net costs obtained by comparing the gross costs of the plutonium disposition campaigns with the costs of operating the same reactors for the same periods using uranium-based fuels. (These calculations were done only for the LWRs; as discussed in “Completing Existing LWRs” above, the MHTGR and ALMR would be expected to show a net economic benefit from using free WPu in place of uranium, but we did not think that the estimates of uranium-based fuel-cycle costs available to us for these reactors were good enough to warrant a numerical calculation.) Discounted to the start of reactor operation, the central estimates of these LWR net costs relative to LEU operation are seen in the table to range from \$500–\$800 million in the case of reactors that do not pay property taxes and insurance, fed by MOX from FMEF, up to \$1,000–\$1,500 million for reactors that do pay these items, fed by new fuel fabrication plants that also pay them.

It must be emphasized that, although we believe the procedures by which these figures were calculated represent some improvement in consistency and transparency over those used in Phase I of DOE’s PDS, it would nonetheless be a mistake to attribute great accuracy to any of the results shown. Some potentially important shortcomings of the contractor studies—such as inconsistent and sometimes inadequate treatment of site costs, interim spent fuel storage, low-level waste management, and seismic requirements—were not corrected here; costs and delays in safety and environmental analysis, licensing, and the like could be greater than implied in the prescriptions we have used for preoperational costs and IDC; and it is hard to be sure whether our procedures for

expanding the uncertainty ranges on contractor estimates of direct plus indirect costs of power-plant and fuel-fabrication-plant construction, and of O&M costs for these facilities, have compensated adequately for noncomparable degrees of comprehensiveness and conservatism in the original estimates.

It should also be emphasized that the per-kilowatt-hour figures in Table 6-16 are not to be confused with estimates of the costs of electricity to be expected from reactors of these types if they were placed in commercial operation on a substantial scale. After all, the estimates given here are based on prorating, over only a few units, substantial preoperational costs associated both with new reactor types and with implementation of plutonium fuel cycles; per-unit and per-kilowatt-hour charges associated with these costs would be smaller if the new reactor types and fuel cycles saw large-scale commercial service. In the other direction, private companies contemplating the use of new nuclear power plants for commercial electricity generation might base their financial calculations on a considerably higher real cost of money than the 0.07/yr figure used here.

We now address briefly the sensitivity of our economic estimates to our assumption that the plants operate only for the duration of the plutonium disposition campaign, rather than continuing to generate electricity using uranium fuel for an assumed 40-year plant lifetime. This question can be addressed (see "Issues and Criteria in Economic Evaluation of Alternatives" in Chapter 3) by considering that the power plant has a residual value at the end of the plutonium disposition campaign equal to the discounted present value at that time of the stream of electricity revenues from its future operation less the discounted present value of the stream of its future operating costs. (There is also a small correction for the postponement of the D&D costs of the power plant, but we neglect it here as small compared to the other uncertainties inherent in the calculation.)

We apply this prescription, as an example, to the case of the evolutionary PWR, using just central estimates. We take its nonfuel O&M costs to be unchanged from those shown in Table 6-16, at about as \$0.012/kWh, its fuel-cycle costs on LEU to be \$0.006/kWh (including repository charge), and the revenues to be \$0.050/kWh. The annual output is 8.26×10^9 kWh, so the net revenue stream is 8.26×10^9 kWh/yr. \times ($\$0.050 - 0.018$)/kWh = \$264 million/yr during the 10 years of plant life beyond the 30 years of operation in the plutonium campaign, which at $r = 0.07$ /yr has a discounted present value, at the end of the plutonium campaign, of \$1,854 million. Discounting this sum to the start of reactor operation at the beginning of the plutonium campaign (30 years earlier), for comparison with the net cost of campaign at reactor startup shown in Table 6-16, gives \$243 million. This approach, then, would reduce the central estimates of the net cost of campaign for the evolutionary LWR from \$1.6 to \$1.4 billion (reactor paying no property tax and insurance, fed by FMEF) or from \$3.2 to \$3.0 billion (reactor and new fabrication plant both pay property taxes

and insurance). Thus, while we believe for the reasons mentioned earlier that the most informative economic comparisons are those based only on the period of plutonium disposition, it is apparent that our conclusions would not be changed much if we instead adopted the TRC approach.

Economics of Vitrification

Much less information and analysis are available on the costs of vitrifying surplus WPu along with defense high-level wastes, compared to the amounts of information and analysis we were able to draw on in making our cost estimates for the reactor options. The one economic analysis we have seen (McKibben et al. 1993) is for the incorporation of 50 tons of U.S. surplus WPu into a fraction of the glass logs already scheduled to be produced at the Savannah River site as a means of immobilizing defense high-level radioactive wastes (see Chapter 5 and Table 6-3). The panel believes that variations of the vitrification option that made less use of already planned facilities and operations would cost more, assuming that the full costs of constructing new facilities, and of producing and disposing of thousands of glass logs that would not otherwise be produced, were charged to the WPu disposition mission.

The McKibben et al. analysis of the cost of adding 50 tons of WPu to the Savannah River log-making operation estimated that the preliminary steps, including conversion of plutonium metal to oxide, would cost about \$400 million and the other increases in the costs of the log-making campaign would amount to about \$600 million. We assume that the preliminary costs are spread over a nine-year period prior to melter operation, which for the usual S-curve pattern and real interest rate of 0.07/yr gives (see Table 3-7) a discounted present value at start of melter operation of $\$400 \text{ million} \times 1.41 = \564 million . If the remaining \$600 million in the McKibben et al. estimate represents additional costs incurred at the rate of \$75 million per year during the eight years of log-making that follow the initiation of plutonium additions to the melter, the contribution of this cost stream to the net discounted present value at start of melter operation would be about \$448 million. We consider that the resulting \$1 billion total for the net discounted present value of the preoperational and operational cost streams as of the start of melter operation (which in our reference scenario would be in 2005) should be considered uncertain by ± 50 percent, given the lack of actual operational experience with an activity of this kind. Thus our estimate for this reference vitrification case is \$0.5–\$1.5 billion.

Economics of Russian Disposition Options

No comparably detailed assessment of the economics of Russian WPu disposition options is possible with the information available to the panel. Costs of large capital projects in Russia, while substantially lower than in the United States, are highly uncertain and, with current substantial inflation, changing

rapidly and unpredictably. Labor costs in Russia are lower than those in the United States by large factors at present (the average salary corresponding, as of mid-1994, to approximately \$100 per month), but labor rates too are changing quickly and unpredictably. Costs in different sectors can be dramatically different. Electricity prices are still at extremely low subsidized levels, and there is a massive nonpayments problem. This has led to a severe economic crisis throughout the nuclear industry, with plants having too little income to pay workers, purchase fuel, and the like. Nuclear workers have staged well-publicized strikes, and have asserted that labor conditions are now further imperiling nuclear safety. A nuclear regulatory agency and an antinuclear environmental movement have only come into being in recent years, and their full impact on how nuclear operations will be conducted in Russia is not yet established. Given these conditions, calculating likely costs of nuclear projects that would take place years in the future is not practical.

Nevertheless, a few remarks are in order. First, Russia has abundant supplies of cheap uranium and uranium enrichment services; indeed, in today's fuel market, the states of the former Soviet Union are the cheapest available source of these commodities. (With plants already built, the primary marginal cost of enrichment is electricity, which, as just noted, is currently quite cheap in Russia.) Second, like the United States, Russia does not have an operating commercial-scale MOX fabrication facility, and therefore very large capital investments would be required before plutonium could be used as a fuel on a substantial scale. It seems clear, therefore, that the conclusion that a kilogram of LEU fuel is cheaper to provide than an equivalent kilogram of MOX fuel—even if the plutonium in the MOX is itself available “free” as a result of weapons dismantlement—is also true in the Russian case.

Because of the low material and labor costs in Russia, however, it may be that a kilogram of MOX fuel could be produced in Russia for less than the cost of an equivalent kilogram of LEU fuel in the West. This could create opportunities for largely private financing of plutonium disposition in Russia—minimizing or eliminating the need for explicit government subsidies—that do not exist in the case of U.S. WPU disposition. For example, if one assumes (conservatively, the panel believes) that the capital cost of building a MOX plant in Russia would be roughly one-half what it would be in the West, and that *all* of the cost of MOX is accounted for by capital cost rather than labor cost (where the cost would be reduced by an even larger factor), then the cost of producing MOX in Russia would be one-half the cost we have calculated for the United States using a new facility and paying no property taxes, or some \$800/kg. (If the partly completed facility could be completed for a lower cost and meet acceptable safeguards and safety standards, the cost might be lower still.) This compares to the \$1,400/kg cost of equivalent LEU projected earlier in this chapter for 2015, the estimated midpoint of plutonium disposition operations.

Thus, if the large current price differentials between Russian and Western prices of labor and materials were maintained over a long period, one could imagine two alternatives by which MOX production in Russia might be privately financed:

- (1) MOX produced in Russia could be sold on the commercial market in Western Europe in competition with LEU. If Western countries or utilities agreed to firm contracts to purchase such MOX, these contracts could serve as the basis for borrowing, on international capital markets, the funds necessary to build and operate the plant.
- (2) If there were a desire to keep MOX operations limited to a small number of sites in the country of origin, Western countries might agree to permit additional LEU sales in their markets (to which Russia's access is limited by trade agreements) for each kilogram of MOX Russia produced and used in its own reactors. In other words, rather than producing MOX for sale abroad and LEU for its own reactors, Russia would produce an equal amount of MOX for use in its own reactors and ship that amount of LEU abroad instead. The price available in the West (and therefore the possibility of borrowing on international capital markets) would be similar to that in the previous case.

Needless to say, major nuclear capital projects of this kind in Russia today would involve substantial uncertainties and risks. The major banks would probably have higher confidence in the success of the project (and therefore charge a lower rate on capital) if a Western firm with experience in MOX production (such as Siemens, COGEMA, British Nuclear Fuels, Limited, or Belgonucleaire) were involved. Some of these firms have already been discussing possible cooperation in MOX fabrication with Russia. (Both government and corporate policy-makers, however, will have to take into account the possibility of stiff commercial competition from a Russian MOX producer, once the plant was built and operational.)

The vitrification option would presumably also be cheaper to implement in Russia than in the West, due to lower capital and labor costs, and possibly a less stringent regulatory environment. In the United States, as described above, the likely net subsidies required for the LWR MOX and vitrification options are likely to be in the same general range. In Russia, however, the possibility described above for largely or completely private financing of the operation presents itself only in the case of the MOX option. Since vitrification produces no saleable product, its full cost would have to be provided by the government, as in the West.

ENVIRONMENT, SAFETY, AND HEALTH

We argued in Chapter 3—dealing with criteria relating to environment, safety, and health (ES&H)—that suitable options for the disposition of WPu in the United States should:

- comply with existing U.S. regulations governing radioactivity and radiation from civilian nuclear-energy activities;
- comply with existing international agreements and standards on the disposition of radioactive materials in the environment; and
- not add significantly to the ES&H burdens that would result, in the absence of programs for WPu disposition, from appropriate management of civilian nuclear-energy generation and of the environmental legacy of past nuclear weapons production.

We argued, further, that disposition activities to take place in other countries should meet the same criteria, with the replacement of U.S. regulations by equivalent regulations of the countries in question. In the present section, we summarize our understanding of the specific issues that will have to be addressed in order to establish that the leading-candidate approaches described in Chapters 4 and 5—namely, the MOX fuel option in power reactors of existing types, and the vitrification-with-high-level-waste option—can indeed meet these criteria.

The main ES&H issues posed by the MOX fuel and vitrification options were identified in “The Main ES&H Issues in Weapons Plutonium Disposition” in Chapter 3. In addressing those issues here, we begin with a synopsis of the characteristics of plutonium that bear on ES&H hazards and then discuss, in turn, the hazards of interim storage, transport, and processing of plutonium; the influence of the use of plutonium-based fuel on reactor safety; and the ways in which the MOX fuel and vitrification options may influence the ES&H characteristics of radioactive wastes. Our treatment of these ES&H issues relates primarily to disposition of WPu in the United States. The arguments would not be substantially different for disposition in Russia or other countries, although the background of existing ES&H risks from civilian and military nuclear-energy activities and the regulatory structures responsible for minimizing these risks would differ.

Relevant Characteristics of Plutonium

Table 6-17 compares some of the hazard-relevant characteristics of principal isotopes of plutonium with the same characteristics of the uranium isotopes that make up natural uranium. Notice that while the half-lives of most of the plutonium isotopes are long enough to make these a long-lived hazard when measured against the scale of a human lifetime, the plutonium half-lives are nonetheless very much shorter than those of the uranium isotopes. As a result,

TABLE 6-17 Hazard-Relevant Properties of Principal Plutonium and Uranium Isotopes

Isotope	Half-Life (yr)	Curies per Gram	Main Emissions ^a (MeV)	Worker Ingestion ALI (μCi)	Worker Inhalation ALI (μCi)	Public DCLW (Ci/m ³)	Public DCLA (Ci/m ³)	Dilution Volume in Air (m ³ /g)
Pu-238	8.8×10 ¹	1.7×10 ¹	5.5 a	9×10 ⁻¹	7×10 ⁻³	2×10 ⁻⁸	2×10 ⁻¹⁴	8.5×10 ¹⁴
Pu-239	2.4×10 ⁴	6.2×10 ⁻²	5.2 a	8×10 ⁻¹	6×10 ⁻³	2×10 ⁻⁸	2×10 ⁻¹⁴	3.1×10 ¹²
Pu-240	6.6×10 ³	2.3×10 ⁻¹	5.2 a	8×10 ⁻¹	6×10 ⁻³	2×10 ⁻⁸	2×10 ⁻¹⁴	1.2×10 ¹³
Pu-241	1.4×10 ¹	1.0×10 ²	0.02 b	4×10 ⁻¹	3×10 ⁻¹	1×10 ⁻⁶	8×10 ⁻¹³	1.3×10 ¹⁴
Pu-242	3.7×10 ⁵	4.0×10 ⁻³	4.9 a	8×10 ⁻¹	7×10 ⁻³	2×10 ⁻⁸	2×10 ⁻¹⁴	2.0×10 ¹¹
Am-241	4.3×10 ²	3.5×10 ⁰	5.5 a, 0.06 g	8×10 ⁻¹	6×10 ⁻³	2×10 ⁻⁸	2×10 ⁻¹⁴	1.8×10 ¹⁴
U-234	2.5×10 ⁵	6.2×10 ⁻³	4.8 a	1×10 ¹	4×10 ⁻²	3×10 ⁻⁷	5×10 ⁻¹⁴	1.2×10 ¹¹
U-235	7.1×10 ⁸	2.1×10 ⁻⁶	4.4 a	1×10 ¹	4×10 ⁻²	3×10 ⁻⁷	6×10 ⁻¹⁴	3.5×10 ⁷
U-238	4.5×10 ⁹	3.3×10 ⁻⁷	4.2 a	1×10 ¹	4×10 ⁻²	3×10 ⁻⁷	6×10 ⁻¹⁴	5.5×10 ⁶

ABBREVIATIONS:

- ALI: annual limit of intake.
 Ci/m³: curie per cubic meter.
 DCLA: derived concentration limits for air.
 DCLW: derived concentration limits for water.
 MeV: million electron volts.
 μCi: microcurie

NOTES: Half-lives and emissions are from IAEA (1986a). Allowable intakes and concentrations are from OFR (1992, Appendix B to Sections 20.1001-20.2401) and apply after January 1, 1994. Americium-241 is included here because of its rapid buildup in plutonium from the decay of 14.4-year half-life Pu-241.

^a a = alpha particle, b = beta particle, g = gamma ray.

the intensity of the radioactivity of plutonium—measured as “specific activity” in curies per gram—is much larger than that of uranium. For example, the specific activity of the most common plutonium isotope, Pu-239, is about 200,000 times greater than that of the most common uranium isotope, U-238. Since the alpha particles emitted in the radioactive decay of Pu-239 are about 25 percent more energetic than those emitted in the decay of U-238, the former is altogether about 250,000 times more damaging per gram, radiologically, than the latter.

The preceding comparison does not take into account differences in the pathways and residence times of plutonium and uranium in the body or in the environment. The differences with respect to pathways and residence times in the body have been accounted for, however, in the Annual Limit of Intake (ALI) by ingestion and inhalation for workers in nuclear industries and the Derived Concentration Limits for air and water (DCLA and DCLW) for public exposure, as promulgated by the Nuclear Regulatory Commission (last 4 columns of Table

TABLE 6-18 Hazard Indices for Various Mixtures of Heavy-Metal Isotopes

	Weight Percent of Isotopes in Mix						Dilution Volume (m ³ /g)		Surface Gamma Dose (rem/hr)
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	In Air	In Water	
	WPu	0.01	93.8	5.8	0.13	0.02	0.22	4.2×10 ¹²	4.2×10 ⁶
RPu	1.3	60.3	24.3	5.6	5.0	3.5	2.9×10 ¹³	2.8×10 ⁷	14.6
	U-234 U-235 U-238								
Natural U	6×10 ⁻³	0.72	99.2				1.3×10 ⁷	2.3×10 ⁰	1.2×10 ⁻⁵
Reactor U	0.03	3.5	96.5				3.8×10 ⁷	6.5×10 ⁰	5.7×10 ⁻⁵
Weapons U	0.12	94.0	5.9				1.8×10 ⁸	3.2×10 ¹	1.5×10 ⁻³

NOTES: Dilution volumes based on specific activities and DCLs from Table 6-17. Surface gamma doses calculated for uranium and plutonium metal at highest naturally occurring densities.

6-17). The DCLs and specific activities can be combined to calculate, isotope by isotope and for any mixture of these, the volumes of air or water that would be required to dilute a gram of any radioactive material to the concentration limits specified by these standards.²⁷ This "dilution volume" index is widely used in preliminary environmental assessment of nuclear technologies, to convey a rough idea of radiologic hazard potential.

Table 6-18 shows the dilution volumes for isotopic mixtures of plutonium corresponding to weapons-grade and reactor-grade material, with comparisons to natural uranium, LEU (at the 3.5 percent enrichment corresponding to typical fuel for a PWR), and HEU (at the 94 percent enrichment corresponding to nuclear weapons application). One sees there that, per gram, weapons-grade plutonium represents a potential inhalation hazard about 23,000 times greater and a potential ingestion hazard about 130,000 times greater than the corresponding potential hazards of weapons-grade uranium; the potential hazards of natural uranium in these respects are about 14 times smaller than those of weapons-grade uranium, and those of reactor-grade plutonium are about 7 times greater than those of weapons-grade plutonium.

²⁷ The DCLs for public exposure correspond to the concentrations that would produce a steady-state whole-body dose commitment of 0.5 millisieverts (mSv) (50 millirem; mrem) per year to an individual who continuously breathed air or continuously drank water contaminated to the indicated levels. If i different isotopes are present simultaneously in air at concentrations C_i , the requirement is that $\sum_i (C_i/DCLA_i) < 1$, and, similarly, if j different isotopes are present simultaneously in water at concentrations C_j , the requirement is that $\sum_j (C_j/DCLW_j) < 1$. Note that dose rates of 0.5 mSv each from air and water, to which the DCLA and DCLW correspond, relate to the total dose rate of 1 mSv (100 mrem) per year permitted to an individual member of the public, under Environmental Protection Agency regulations, from all nuclear facilities combined; the dose permitted from any single nuclear facility, however, is 10 times smaller (see "Some Relevant Standards Limiting Doses and Emissions" in Chapter 3).

The DCLs and dilution volumes for the plutonium and uranium isotopes are dominated by the dose potentials presented by these isotopes as internal (inside the body) emitters of alpha particles. As can be seen by comparison of the occupational annual intake limits for plutonium via ingestion and inhalation (Table 6-17, columns 5 and 6), a given number of curies is much more dangerous if inhaled than if ingested. This is because most ingested plutonium passes through the body without being absorbed, whereas plutonium inhaled in the form of particles of small diameter tends to become lodged in the lungs; there the plutonium can do great damage directly, as well as being gradually absorbed into the bloodstream and thus gaining access to bone, kidneys, and other vulnerable parts of the body. Recent reviews of the available information on the carcinogenicity of inhaled plutonium oxide suggest that the damage-to-exposure ratio is in the range of 3 to 12 excess cancer deaths per milligram of weapons-grade plutonium inhaled in oxide form by an exposed population (Fetter and von Hippel 1990, National Research Council 1988).²⁸

Notwithstanding the high carcinogenic potential of alpha particles when emitted inside the body, they cause no harm when emitted outside it because they cannot penetrate the dead layer of the skin. Gamma rays can penetrate the body from outside it, however, and while plutonium and uranium isotopes and most of their daughter products emit very little in the way of gamma rays, the americium-241 (Am-241) daughter of Pu-241 emits a 60-kiloelectron volt (keV) gamma ray that can be a significant source of external radiation dose. The amount of Am-241 in a given quantity of plutonium depends both on the initial Pu-241 concentration (which is relatively high in reactor-grade plutonium and low in weapons-grade plutonium) and on the time elapsed since the plutonium was separated (since the buildup of Am-241 is governed by the 14-year decay half-life of the Pu-241). The gamma dose rates at the surface of metallic spheres of plutonium and uranium of different isotopic compositions are shown in the last column of Table 6-18. It can be seen there that even the modest percentage of Am-241 in weapons-grade plutonium (about 0.2 percent for the nominal composition considered here) is enough to yield a gamma dose rate about 600 times higher than that from weapons-grade uranium, and the gamma dose rate from reactor-grade plutonium containing a nominal 3.5 percent Am-241 is another 15 times higher still.

The ratios of dilution volumes and of surface gamma doses are instructive in clarifying why the handling and processing of plutonium require far more stringent precautions for the protection of workers and the public against radia-

²⁸ Under the usual linear hypothesis (see Chapter 3, Appendix D), this would mean that 3–12 excess cancer deaths would be expected in a population of 1,000 people inhaling 0.001 milligrams each of weapons-grade plutonium in oxide form, that the same number of excess cancer deaths would be expected in a population of 100 people inhaling 0.01 milligrams each, and, at the extreme, that one individual inhaling 80–300 micrograms could expect to suffer a fatal cancer as a result.

tion than do the handling and processing of uranium. Some additional factors, relating to the pathways by which plutonium can become available for intake by humans, are also relevant. One such factor is that plutonium metal shavings or filings can be pyrophoric (that is, they can ignite spontaneously in air); this phenomenon has been responsible for a number of serious fires in the U.S. nuclear-weapons-production complex (IPPNW/IEER 1992, National Research Council 1989). Plutonium fires, of course, produce plutonium-oxide smokes representing an inhalation hazard for workers and, if the fire is uncontained, for the public.

Another set of pathways by which plutonium could reach humans is propagation through food chains following dispersal in the environment by, for example, a fire or other accident during transport. Plutonium appears to bind strongly to soil particles and is not readily taken up by terrestrial plants, with concentration factors (parts per million [ppm] plutonium in dry plant material divided by ppm plutonium in dry soil) reported to be typically less than 0.01 (Eisenbud 1973). This means that resuspension of respirable plutonium particles from contaminated soil is likely to be a more important pathway to human exposure than ingestion of plutonium taken up from soil by food crops. In marine ecosystems, by contrast, plutonium is strongly concentrated by plankton, seaweed, and shellfish: marine concentration factors for plutonium used by the International Atomic Energy Agency in connection with the regulation of ocean dumping of radioactivity include factors of 100,000 (ppm plutonium in wet biomass divided by ppm plutonium in seawater) in plankton, 2,000 in seaweed, 3,000 in molluscs, and 40 in fish; the corresponding concentration factors for uranium are 20, 100, 30, and 10, respectively (IAEA 1986b). These figures suggest that marine food chains could be important pathways to human exposure if plutonium found its way in significant quantities into the oceans.

In addition to the characteristics of plutonium described in the preceding paragraphs, all of which bear on the radiological hazards presented by the handling and possible dispersal of this element, the possibility of an accidental chain reaction involving plutonium must also be mentioned. The possibility of plutonium's "going critical" as a result of mishandling or other mishap in processing, transport, or storage represents a danger of direct irradiation of humans in the vicinity by the intense flux of neutrons and gamma rays produced by criticality (see section on "Physics and Technology of Nuclear Fission" in Chapter 2) as well as a secondary radiologic hazard from the fission products produced by the chain reaction; and it could represent, in some circumstances, a source of sufficient energy release to disperse the plutonium itself and any accompanying radioactive material more rapidly or widely than would be likely in the absence of a chain reaction.

It must be emphasized that the kinds of criticality accidents to which we are referring here—that is, excluding the accidental detonation of a nuclear weapon (which is outside the scope of our responsibilities in this report) and criticality

accidents in nuclear reactors (which are discussed separately below)—are not expected to entail actual nuclear explosions or even the smaller but still very impressive energy releases associated with severe nuclear reactor accidents.²⁹ The few criticality accidents known to have occurred over the years in the U.S. nuclear-weapons complex, for example, have involved nuclear-energy releases ranging from a few megajoules to a few hundred megajoules, in contrast to about 85 million megajoules in a 20-kiloton nuclear weapon and perhaps 1 million megajoules in the criticality accident that set off the Chernobyl reactor disaster (National Research Council 1989, IPPNW/IEER 1992). The nuclear energy releases in the nonreactor criticality accidents were enough, however, to produce potentially lethal radiation doses at a distance of several meters; and it is certainly possible to imagine circumstances—e.g., with the reactants confined by soil or rock in a shallow burial site or deep geologic repository—where the energy release could continue for long enough to become a significant driver of dispersal of radioactivity.

Hazards in Interim Storage of Plutonium

All options for the disposition of WPu will entail some period of interim storage of this material subsequent to the dismantling of the warheads. The 1994 CISAC study, for which our own work was conducted, concluded that the preferred approach for this interim storage is to store the plutonium in the form of the nuclear-weapon cores or “pits” in which it is removed from the weapons during dismantling, perhaps with the addition of some deformation or other modifications to reduce the immediate reusability of the pits in new nuclear weapons. Destroying the pits and converting their plutonium metal to alloys with other metals, or to plutonium oxide or plutonium nitrate or other compounds, was not recommended for this interim-storage purpose because (1) such processing produces security liabilities that at least partly offset the gains;³⁰ (2) the processing would entail costs, which it is more efficient to defer until it has become clear what form of processing is needed for whatever subsequent disposition option is chosen; and (3) the processing would entail additional ES&H hazards, beyond those associated with storing the pits, with no clear reason to want to face them in the absence of substantial security gains from the processing and in view of the possibility that a different form of processing might turn out later to be what is required.

²⁹ Achieving a true nuclear explosion requires bringing the fissile material from a subcritical to a highly supercritical configuration in a very small fraction of a second. The considerable difficulty of accomplishing this intentionally in a nuclear weapon, in which a substantial quantity of chemical high explosive and carefully configured fissile components are employed for the purpose, suggests that its accidental occurrence in the absence of these ingredients is extremely improbable.

³⁰ Processing itself provides opportunities for diversion or theft, and the plutonium pits are easier to count and to guard than, for example, plutonium-oxide powders.

The problem of pit storage, being common to all disposition options, is not strictly speaking a part of the Reactor Panel's charge, but we nonetheless will offer some observations here on the ES&H aspects of this operation, in the interest of completeness. The main ES&H issues associated with pit storage would appear to be:

- (a) avoidance of criticality accidents resulting from excessive proximity and inadequate shielding of two or more pits in combination, as could occur in the course of bringing the pits into the facility, or as a result of insufficient care in designing the array in which they are to be stored, or as a result of subsequent unintended rearrangement of this array, for example, by flood, earthquake, or aircraft impact;
- (b) avoidance of accidental plutonium dispersal, particularly by fires in the storage facility (which could mobilize the plutonium metal in the pits as plutonium-oxide smoke); and
- (c) avoidance of excessive doses to the workers handling the pits at the time of their emplacement in storage or in subsequent monitoring of them (such avoidance being mainly a matter of a combination of shielding and restricted exposure time, with particular reference to the gamma emissions from the Am-241 daughter of Pu-241).

Careful attention will have to be paid to all of these issues, and assurances about how the first two in particular will be managed will have to be provided to the communities in the vicinity of the storage facilities as well as to the responsible regulatory authorities.

At the same time, we see no reason that these risks cannot be restricted—given appropriate diligence and the provision of adequate funds—to very low levels, well within our criteria that existing U.S. regulations about public and worker exposures to radioactivity would be met and that the total ES&H burdens of this activity would be small compared to those associated with appropriate management of other civilian and military nuclear-energy activities. The safe storage of pits is not, after all, a more demanding task than the safe storage of comparable (and even larger) numbers of intact nuclear weapons, and the latter task has been accomplished in the United States for decades with few ES&H problems.³¹

Temporary storage of plutonium in forms other than pits will of course be involved at the later stages of any disposition option, and ultimately all the plutonium that has not been fissioned will have to be stored essentially permanently

³¹ The ES&H problems that have occurred in some parts of the U.S. nuclear-weapon complex over the years (about which more below) have been associated mainly with steps other than weapon storage; see National Research Council (1989), OTA (1991, 1993), and IPPNW/IEER (1992). While less is known about the corresponding history in the nuclear-weapon complex of the former Soviet Union, the main ES&H problems there that have come to light so far also involved operations other than weapon storage (IPPNW/IEER 1992).

in one or another form of radioactive waste. The ES&H issues connected with radioactive wastes are taken up below in "Radioactive Waste Issues." The prior forms needing temporary storage are likely to include plutonium-dioxide powder and may also include liquid plutonium nitrate solutions and fabricated MOX nuclear fuel. Of these, the plutonium dioxide and the plutonium nitrate will require the greatest care to avoid accidental criticality and release modes that could produce significant worker exposures. (The fabricated MOX fuel is less problematic because (1) the plutonium is diluted by about 20-to-1 with nonfissile and much less radiotoxic U-238, (2) the plutonium is confined by the ceramic-pellet fuel matrix and by metal cladding, and (3) the cladding attenuates somewhat the gamma rays from the contained Am-241.)

Handling of plutonium dioxide and plutonium nitrate in the nuclear-weapon complex has not always been exemplary (National Research Council 1989, IPPNW/IEER 1992, OTA 1993)—all of the criticality accidents that have been publicly described involved plutonium nitrate, for example—but the technologies and procedures for avoiding such problems are well established (see, e.g., PNL 1988, OTA 1993). Levels of awareness and sophistication about ES&H issues are greater in the 1990s, in the relevant government organizations and indeed throughout society, than was the case in earlier decades when ES&H problems in some parts of the nuclear-weapon complex materialized; and specific recent initiatives give further reason to expect that the diligence required for safe plutonium handling in dealing with surplus nuclear weapons will in fact be applied.³² The ES&H risks associated with storage of plutonium from dismantled surplus nuclear weapons must be compared, in any case, with the risks of storage of a wider array of plutonium forms and plutonium-contaminated materials that have been produced over the years in both the civilian and military nuclear complexes; the forms and locations of much of this material make it more problematic from the ES&H standpoint than stored plutonium from surplus weapons is likely to be.

³² For example, DOE has recently carried out an extensive Plutonium Vulnerability Study identifying the most important ES&H problems associated with the various forms of plutonium at the sites in the DOE complex, and has developed an extensive action plan to resolve the problems found. In April 1994, before the vulnerability study was complete, the Defense Nuclear Facilities Safety Board (DNFSB) recommended a wide range of improvements in handling of plutonium in the DOE complex. Intensive work is now underway at several DOE sites to correct the deficiencies identified by the DNFSB and the vulnerability study. In addition, although DOE's facilities have been exempted until now from regulatory oversight by the Occupational Safety and Health Administration (OSHA), DOE's May 1993 Health and Safety Initiative commits the DOE to a transition to regulation by OSHA over the next 3–5 years. The background leading up to these developments is described in considerably greater detail in the recent study of management of nuclear-weapon materials by the Office of Technology Assessment of the U.S. Congress (OTA 1993).

Hazards in Plutonium Transport

The transport links likely to be associated with plutonium disposition were detailed above in "Economic Comparisons," in connection with discussion of the associated security risks. Truck and rail are the modes most likely to be chosen for these transport activities, and in either case special precautions (protective containers, carefully selected routes) will be needed to minimize the probability of a release and the extent of public exposure if a release occurs.

In the case of the transport of intact or deformed pits from the warhead dismantlement plant to the (future) central storage facility and the subsequent transport of the pits from there to a MOX fuel fabrication or vitrification plant (assuming that conversion to oxide takes place at these plants and not adjacent to the storage facility), the obvious comparison is with transport of intact nuclear weapons. Such transport has been managed in the United States for tens of thousands of nuclear weapons over the years with almost no serious ES&H incidents; the only exceptions have involved aircraft crashes. (Less is known about the record in the former Soviet Union.)

Air transport is not likely to be contemplated for the transfer of pits from interim storage to a MOX fabrication or vitrification plant on the same continent (mainly because of the difficulty of designing reasonably lightweight containers that could prevent the dispersal of the plutonium in the event of a crash); and transport of pits between continents is, for political and security reasons, unlikely to be undertaken at all. Truck and rail transport are therefore the modes most likely to be used, and we see no reason why such transport cannot be managed for pits with the same high degree of safety and reliability that has characterized truck and train transport of intact warheads in the United States. If, nonetheless, a transport accident severe enough to breach the shipping containers did occur, the possibility of a fire's converting the plutonium metal in the pits to plutonium-oxide smoke would pose a public health hazard.³³

In the case of the vitrification option, co-location of the processes for converting pits to plutonium oxide with the vitrification plant would avoid further transportation steps except for the eventual transport of the plutonium-bearing glass logs to a permanent geologic repository. Inasmuch as the nominal 50 tons of surplus U.S. WPu could be embedded in fewer glass logs than will need to be produced in any case in order to stabilize the accumulated high-level radioactive wastes from this country's defense programs, choosing this means of WPu disposition would not entail any additional transportation of logs or, hence, any additional probability of transportation accidents.

Nor would the addition of WPu to the glass logs affect significantly the routine doses of radiation to the workers involved with transporting the logs. The routine doses to transport workers would come almost entirely from gamma

³³ Some pits were designed to be fire resistant and would, therefore, be less susceptible to this accident scenario.

radiation; in logs containing about 20 percent by weight of 20- to 40-year-old fission products and 1–2 percent by weight of WPu, the gamma dose from fission product cesium-137 will exceed that from the Am-241 in the plutonium by several orders of magnitude (see Table 6-5 and accompanying notes).

As for doses to the public in the event of an accident, even a crash severe enough to break the transportation cask, followed by a long and intense fire, would be unlikely to mobilize much of the highly refractory glass log. To the extent that such an accident did mobilize radioactive material from the log, however, the contribution of the plutonium to the dose potential to the public could add very substantially to the contribution from the fission products.³⁴ This means that, in order to meet the criterion that the disposition of the WPu should not add substantially to the ES&H risks from the nuclear-energy activities that would be going on in any case, one must ensure that the total expected dose to the public from accidents in the transport of the glass logs will be small compared to the doses to the public from other aspects of civilian and military nuclear-energy activities, if those other aspects are properly managed. Given the relatively small number of glass logs needed to accommodate all of the surplus WPu, the highly refractory character of the glass, and the capacity to make shipping casks that greatly reduce the probability of significant releases even in severe accidents, it seems very likely that this will be the case. Further study of the mobilizability of plutonium from borosilicate glass logs in severe transport accidents is warranted, however.

In the case of the spent fuel option, it is reasonable to assume that conversion of pits to plutonium oxide would be co-located with other fuel fabrication steps, so that the remaining transport links would include only the transport of the fabricated fuel to the reactor site(s)—if the reactors are not co-located with the fabrication plant—plus transport of the spent reactor fuel to any intermediate

³⁴ According to Table 6-5, note 9, the logs to be produced at Savannah River will contain about 13 curies (Ci) each of cesium-137 (Cs-137) and strontium-90 (Sr-90) per kilogram of glass—these two isotopes dominate the hazard of a fission-product mixture of the age of this material—and at 2-percent WPu by weight the logs would contain 20 gWPu/kg. Suppose the respective fractional release rates of plutonium and fission products were similar to those expected from molten spent fuel in a severe reactor accident. The “worst-case” accident releases from oxide reactor fuel are generally estimated to be 10–40 percent for cesium, 5–10 percent for strontium, and 0.3–3.0 percent for plutonium (USNRC 1975, Hohenemser 1988). Taking 25 percent for cesium, 5 percent for strontium, and 1 percent for plutonium would give releases from the glass of about 3 Ci Cs-137, 0.7 Ci Sr-90, and 0.2 gWPu/kg of glass. (It is not necessary to imagine that absolute releases of this magnitude are possible; we are interested just in the relative magnitudes of the fission product and plutonium hazards, assuming they are released in these proportions.) Taking the dilution volume for WPu from Table 6-18 as 4×10^{12} m³/g and those for Cs-137 and Sr-90, based on the January 1, 1994 DCLAs from the U.S. Code of Federal Regulations (OFR 1992), as 5×10^9 m³/Ci and 2×10^{11} m³/Ci, respectively, gives “hazard measures” of 1.4×10^{11} m³ for the released cesium and strontium (combined) versus 8×10^{11} m³ for the released plutonium. If the material were totally vaporized, releasing plutonium and fission products in direct proportion to their weight fractions in the glass, plutonium’s dominance of the hazard (by this crude measure) would be even greater.

storage site and, later, its transport to a final repository. Fresh-fuel transport is a link with no counterpart in the vitrification option, but it probably poses only moderate ES&H risks. Because of the absence of fission products, the relatively low gamma dose rate from WPu, and the integrity of the fuel pellets and cladding under normal conditions, the routine radiation exposures to the workers in transport of fresh MOX fuel should be extremely low.

As for the risk of public exposures in severe transport accidents involving fresh fuel, the use of suitably sturdy and fire-resistant shipping containers should be able to reduce to a very low level the probability of accidents severe enough to mobilize significant quantities of plutonium. That plutonium-oxide pellets have a very high melting point and do not burn is helpful in this respect. (The combustibility of the graphite matrix in MHTGR fuel in a severe accident that includes a fire needs clarification. In the case of those LMRs that use metallic fuel, the fuel would certainly be combustible in a sufficiently severe fire; but since fabrication of such fuel under the IFR scheme is likely to be co-located with the reactor, there would be no highway or railroad accidents in fresh-fuel transport to serve as causes for such fires.) If significant quantities of fresh fuel *could* be mobilized so as to become airborne as particulate matter, the resulting public health risks would be substantial.³⁵ This puts a heavy premium on prevention.

Transport of spent MOX fuel under the spent fuel option would pose transportation risks qualitatively similar to those from transport of glass logs under the vitrification option. The quantitative risks of transport depend on the number of shipments and the distance of transport, on the probability of shipping-container failure in the event of an accident, and on the radiologic hazard potential of the material in each shipment and its mobility under accident conditions.

The spent fuel assemblies will have larger inventories of fission products and larger plutonium hazard potential per kilogram than the glass logs, and some of the fission products in the spent fuel may be in forms more volatile under accident conditions than any of the fission products in the glass logs.³⁶ On

³⁵ Recall from Table 6-1 that the amounts of plutonium in fresh MOX or MHTGR fuel as a fraction of the fuel matrix fall in the range of 1 to 5 percent by weight, with plutonium fractions in LMR fuels being several times higher. A gram of MOX fuel containing 4 percent by weight WPu and mobilized as fine airborne particulate matter would require a dilution volume of $1.6 \times 10^{11} \text{ m}^3$ of air to reach the DCLA corresponding to a whole-body-equivalent annual dose of 0.5 millisieverts (50 millirem) from continuous inhalation. This implies that the whole-body-equivalent dose-commitment from one-time inhalation of 25 micrograms of this material would exceed the 0.25 sievert (25 rem) once-in-lifetime emergency dose limit specified in Nuclear Regulatory Commission regulations. (To derive this result, note that the steady-state annual dose from continuous intake of a curie of a radioactive mixture per year is approximately equal to the lifetime dose commitment from a one-time intake of a curie of the same mixture; see, e.g., APS 1978.)

³⁶ Thermal fission of U-235 makes 3.3 Ci Cs-137 and 2.9 Ci Sr-90 per MWd of thermal energy release, hence per gram of fission products, while thermal fission of Pu-239 makes 3.3 Ci Cs-137 and 1.1 Ci Sr-90. Thus fresh spent fuel from the thermal fission of MOX fuel irradiated to 40

the other hand, the brittleness of the glass logs could contribute to the potential for dispersion of the material in severe accidents, and, at our assumed loadings of plutonium in glass logs and in spent MOX fuel, the tonnage of material to be shipped in the course of disposition of 50 tons of WPu will be about 2.5 times smaller for the MOX option than for the glass option.³⁷ The average shipment distance would be on the order of three times greater for the glass logs than for the spent fuel, assuming that the MOX-burning reactors would be in the western United States, the glass-log manufacture would be at the Savannah River site, and the repository for either waste form would be at Yucca Mountain; other assumptions are possible, of course, and would yield different conclusions about comparative distances of transport.

The actual number of shipments that will be required in each case will depend not only on the mass and bulk of the spent fuel assemblies and of the glass logs in their canisters, but also on the mass and bulk of the respective shipping containers. Such containers must provide shielding to protect drivers, other workers, and the public from the gamma radiation originating in the material inside, as well as providing containment in the event of transport accidents. Shipping casks for LWR spent fuel have undergone extensive development and testing over a period of many years in several countries, and the resulting designs have demonstrated their integrity under the severe stresses imposed by

MWd/kgHM will contain 132,000 Ci Cs-137 and 44,000 Ci Sr-90 per MTHM, or 115,000 and 40,000 Ci of Cs-137 and Sr-90, respectively, per ton of MOX/fission product matrix. This spent fuel will also contain 20–50 kg Pu/MTHM (depending on the plutonium loading in the fresh fuel)—hence 18–44 kg Pu/ton of MOX matrix—which in most cases will be similar in isotopic composition to reactor-grade plutonium produced in once-through LEU fuel. If the spent fuel averages 15 years old at the time of transport, the fission product quantities will be about 80,000 Ci Cs-137 and 30,000 Ci Sr-90 per ton of MOX/fission product matrix. These figures compare to about 13,000 Ci each of Cs-137 and Sr-90 per ton of the borosilicate glass to be produced at Savannah River, and 13 kg of weapons-grade plutonium per ton in this glass (at the 1.3 percent by weight plutonium loading assumed in the reference case of Table 6-3). Then the Cs-137 hazard per kilogram of spent fuel would be about six times greater than that per kilogram of glass, the ratio for Sr-90 would be about a factor of 2, and the ratio for plutonium would be a factor of 7–20 (a factor of 1.4–3.4 from the greater mass of plutonium in the spent fuel and another factor of 5–6 from the greater radiotoxicity per gram of the post-irradiation plutonium compared to the still-weapons-grade plutonium in the glass). A modest additional factor of excess radiotoxicity in the spent fuel compared to that in the glass would result from the lower age of the fission products in the former, meaning that the accompanying fission products other than Cs-137 and Sr-90 would have had less time to decay or leak away. The relative mobility of fission products in spent fuel and glass under accident conditions needs more study.

³⁷ Disposition of the nominal 50 tons of WPu would utilize 2,200 glass logs containing $2,200 \times 625 \text{ liters} \times 2.7 \text{ kg/liter} = 3.7 \times 10^6 \text{ kg glass}$; the mass of the logs including their stainless steel canisters would be $4.7 \times 10^6 \text{ kg}$. For the reference MOX fuel case (CLWR, 100-percent MOX core, 40 gWPu/kgHM in fresh fuel, 42-MWd/kgHM irradiation), the disposition of 50 tons of WPu would utilize 2,700 fuel assemblies containing $1.25 \times 10^6 \text{ kgHM}$, with a total mass, including cladding, spacers, and so on, of $1.8 \times 10^6 \text{ kg}$, or about 2.5 times less than that of glass logs with their canisters.

collisions, falls, fires, and the like.³⁸ As far as we know, comparable testing of shipping containers for glass logs has not yet been carried out. Thus, we cannot yet determine whether glass-log containers designed to the same shielding and accident-integrity standards as are met by spent fuel casks will be more or less massive, in relation to the mass of the contents, than the spent fuel casks. If the ratio of mass of container to mass of contents were the same in both cases, then the number of shipments of glass logs would be larger than the number of shipments of spent fuel by approximately the factor of 2.5 ratio of the masses of the plutonium-bearing material to be shipped. We assume, then, that a somewhat smaller number and shorter average distance of shipments of spent fuel as compared to glass logs will tend to offset a somewhat larger spent fuel risk per shipment (based on higher radiologic hazard potential per kilogram, comparable container integrity, and, perhaps, higher mobility of radioactivity in the event of container failure). Thus we judge the overall transport risk to be roughly comparable.

The spent fuel from the use of WPu in MOX fuel may be somewhat more hazardous, in terms of potential public exposure in transport accidents, than spent fuel from the once-through use of LEU fuel. This is because there is two to four times more plutonium per ton in the WPu spent fuel than in comparably irradiated LEU spent fuel, while the isotopic compositions, and hence the hazard per gram, are practically the same. (The extra plutonium hazard in the WPu spent fuel is partly offset by the threefold smaller production of Sr-90 from thermal fission of Pu-239, compared to that from thermal fission of U-235. The Cs-137 contents are about the same in the two fuel types, for equal burnup.) These figures suggest that the overall extra public risk from transport of WPu-MOX spent fuel compared to that from transport of LEU spent fuel of equal burnup is not more than a factor of two or three.³⁹ Two important conclusions follow from this result:

- (a) Since the volume of spent fuel transport from ordinary civilian nuclear-power generation over the next few decades will be greater than the

³⁸ In the United States, for example, shipping casks for spent fuel must be able to survive, in sequence, a drop through a distance of 9 meters onto a hard surface, a drop of 1 meter onto a vertical, 15-cm-diameter steel bar, a 30-minute fire at 800° C, and 8 hours of submersion under a meter of water (OFR 1992, Sec. 71.73).

³⁹ Consider 15-year-old spent fuels that were irradiated to 42 MWd/kgHM and that contain 1.0 percent plutonium in the case of the LEU fuel and 2.6 percent plutonium in the case of the WPu-MOX fuel (see Table 6-1). The potential hazards from volatilization of portions of these fuels in accidents will be dominated by Cs-137, Sr-90, and the various plutonium and americium isotopes. Based on the production rates for Cs-137 and Sr-90 given in note 36 and the release fractions for cesium, strontium, and plutonium given in note 34, the dilution volume for the radioisotopes releasable from a kilogram of fuel is 1.9 times higher for the WPu-MOX fuel than for the LEU fuel. If one assumes, instead, that mobilization fractions are identical for all elements, the dilution volume per gram of released material is 2.4 times higher for the WPu-MOX fuel than for the LEU fuel.

volume of spent fuel transport from WPu disposition by much more than a factor of two or three,⁴⁰ we can conclude that the extra risk associated with transport of spent fuel from WPu disposition will be small compared to the risks of the same kind that society will be incurring anyway from civilian nuclear-energy activities. (That is, it is not necessary to invoke the fact that spent fuel transport risks are much smaller than some other fuel-cycle risks⁴¹ in order to reach the conclusion that the third of our ES&H criteria is met.)

- (b) Since we have concluded that the transport risks of glass logs containing 1.3 percent WPu would be roughly comparable to the risks of the spent fuel transport connected with disposition of the same total quantity of WPu, it follows from (a) that the glass-log transport risks would also be small compared to risks of the same general kind that society will be incurring anyway from civilian nuclear-energy activities. So the third of our ES&H criteria is met for the transport risks from this option too.

Of course, much more thorough risk assessments for the transport of various plutonium forms will need to be done before such activities are actually undertaken on a substantial scale. We think the rough quantitative comparisons presented here for the MOX spent fuel and vitrification options are sufficiently robust, however, that our conclusion about the incremental transport risks for these materials being small compared to other risks of the same kinds is not likely to be overturned by further work.

Hazards in Plutonium Processing

Metal-to-Oxide Conversion

Conversion of the plutonium metal pits to plutonium oxide would be required as an initial processing step for the vitrification option as well as for the spent fuel option with most reactor types—LWRs, CANDUs, and MHTGRs using the current leading-candidate fuel formulation, and some LMRs. In the case of other HTGR designs using carbide fuels, and in the case of those LMR types that use metallic fuels in which plutonium is alloyed with other metals, the initial processing step would differ in details but would involve similar ES&H hazards. These arise mainly from the radiological toxicity of plutonium and

⁴⁰ Current U.S. nuclear-generating capacity of 99,500 MWe discharges more than 2,500 MTHM in spent fuel per year, hence more than 25,000 MTHM per decade and more than 75,000 MTHM in the nominal 30-year operating lifetimes of this generation of reactors. A campaign to process 50 tons of WPu at a MOX-fuel loading of 5 percent WPu in heavy metal would produce 1,000 MTHM in spent fuel.

⁴¹ See, for example: Smith 1978, Fischer et al. 1987, and Lahs 1987.

from its potential to achieve criticality, as discussed in general terms above in “Relevant Characteristics of Plutonium.”

The most widely used process for plutonium oxide production entails dissolution of the plutonium metal in acid to form plutonium nitrate, followed by steps to precipitate plutonium dioxide from the solution. The main *nonroutine* ES&H hazards associated with these operations—that is, events of low frequency that, by virtue of their severe consequences, one strives to make as improbable as possible—are mainly criticality accidents and fires. The plutonium nitrate solution is particularly problematic in terms of criticality, because it is easy for a liquid to flow accidentally into a more critical geometry and because the reactivity of any given plutonium concentration in the solution is increased by the presence of water, which serves as a moderator. Criticality accidents in plutonium processing pose dangers almost exclusively to workers, since the energy releases involved are not large enough to breach the buildings in which the operations are taking place. In any case, these accidents can be avoided by scrupulous adherence to appropriate procedures and proper design of the containers and transfer systems for plutonium solutions.

Fires, by contrast, have the potential to mobilize plutonium in ways that can lead to public as well as worker exposures, and fires are probably harder to prevent altogether than are criticality accidents. Nonetheless, proper design and operation of the facilities, combined with adequate onsite fire-fighting capabilities, surely can hold the occurrence of fires—and the mobilization of plutonium from those that do occur—to very low levels. This is a matter that certainly will require the most careful attention from the designers, operators, and regulators of any plutonium disposition option.

The main routine hazards of the plutonium-processing operations in oxide production are to workers: the mobilization of respirable plutonium aerosols and exposure to gamma irradiation from the Am-241 contained in the plutonium. While carelessness can certainly lead to excessive doses from these sources, available technologies and processes (involving the use of glove-boxes, shielded processing cells, and the like) appear to be adequate—if conscientiously applied—to keep the doses well within existing regulatory guidelines (see, e.g., OTA 1993). Of course, the regulatory authorities with oversight of the WPu disposition campaign will have the task of ensuring that the needed conscientiousness materializes, and that it does not weaken over time under pressures to cut costs and meet schedules.

There is no obvious reason that the ES&H hazards from plutonium metal-to-oxide conversion in connection with the spent fuel and vitrification disposition options should be any larger than similar plutonium-handling hazards encountered in the course of nuclear-weapon production—indeed, with the improved technology and regulatory oversight that can now be expected in comparison to the practices that prevailed when most weapons production was taking place, the hazards should be smaller. Nonetheless, it may well be that these

exposures will prove to be the hazard of WPu disposition most difficult to make small compared to analogous hazards in the other military and civilian nuclear-energy operations that will be occurring in the future—that is, these exposures may be the most problematic ones in relation to our third criterion—so they should (and we assume will) receive particular scrutiny.

Further Plutonium Processing for Spent Fuel Options

In the case of spent fuel options for disposition of WPu, processing steps beyond metal-to-oxide conversion would include: (1) mixing the plutonium in the appropriate proportions with the other constituents of the fuel (e.g., uranium dioxide in the case of MOX fuels, alloying metals for metallic fuels); (2) the actual production of the basic fuel entities (pellets, particles, etc.); and (3) the fabrication of the completed fuel elements. In all of these steps, the fact that plutonium and not just uranium is being used is decisive in governing the ES&H hazards, both because of the much higher radiological toxicity of plutonium compared to uranium and because the plutonium poses greater criticality problems.

The resulting need for special equipment and precautions in the fabrication of MOX fuel, as compared to what is involved in uranium fuel fabrication, is obvious, and this accounts for much of the difference in cost between MOX fuel fabrication and LEU fuel fabrication, as summarized above in “Weapons Plutonium Versus Uranium as Power Reactor Fuel.” At the same time, the hazards at the fuel fabrication stage should be smaller in several respects than those encountered in plutonium metal-to-oxide conversion: once in oxide form, the plutonium is not flammable like the metal nor as prone to criticality accident as a plutonium nitrate solution; and, once diluted with uranium oxide, its radiologic and criticality hazards are further reduced.⁴² If, as we have concluded above, the ES&H hazards of plutonium metal-to-oxide conversion can be expected to satisfy the ES&H criteria set forth in this report, this should also be true of the additional plutonium-processing steps in MOX fuel fabrication.

Further Plutonium Processing for the Vitrification Option

In the case of the vitrification approach to plutonium disposition, the only plutonium-processing step beyond oxide production would consist of incorporation of the oxide into the mix of borosilicate glass and fission products either in the melter, or before the glass frit and fission products are introduced into the melter. In the subsequent pouring, cooling, and handling of the glass logs,

⁴² We note also that the radiologic toxicity of MOX made with WPu will be considerably less than that of the MOX now being made with recycled power-reactor plutonium in France and Belgium.

the presence of 1–2 percent by weight plutonium in the glass would add little to the ES&H risks of these operations, which would be dominated by the fission products in the high-level wastes (HLW) constituting about 20 percent of the mixture. (The dominance of the fission-product hazards in these steps arises from the combination of their larger quantities plus the presence among them of intense gamma emitters, above all Cs-137.) Also, plutonium criticality in intact glass logs will not be a problem, because of the combination of low plutonium concentration and the substantial content of neutron-absorbing boron in the glass. Possible criticality problems in the melter are discussed below, in this subsection, and criticality hazards after thousands of years in a geologic repository, when the boron may have leached away, are discussed below in “Radioactive Waste Issues.”

The hazards presented by the presence of, and operations with, plutonium in the vitrification plant prior to mixing with the melt would be mainly those of plutonium-oxide production, as discussed above. In the vitrification case, however, the context to which these plutonium hazards would be added would be a riskier one than in the spent fuel case—a vitrification plant full of HLW as opposed to a fuel fabrication plant containing only LEU—so the *relative* change in ES&H risks caused by the addition of the plutonium would be smaller.⁴³

An ES&H risk unique to the vitrification case, however, is posed by the possibility of criticality accidents in the melter. These might occur if a substantial part of the added plutonium dioxide somehow coalesced in one part of the melt rather than being well mixed throughout it. A number of approaches to minimizing this risk can be envisioned, such as by using melters small enough that, at the highest plutonium loading contemplated, they would never contain as much as one critical mass of plutonium; it would also be necessary to ensure against malfunctions in the plutonium feed system that might add a higher proportion of plutonium to the melt than is intended. The obvious alternative approach of avoiding criticality by means of mechanical stirring to ensure adequate mixing is not permitted under current U.S. safety regulations.

Determining how best to ensure that melter criticality problems do not contribute significantly to the ES&H hazards of plutonium disposition by the vitrification route is a technical issue that needs to be resolved before this disposition option is embraced. We do not think it will be so difficult to resolve, however, as to pose a significant obstacle to proceeding with this option if that

⁴³ For example, a major fire in a vitrification plant being used for plutonium disposition might mobilize a similar quantity of plutonium to that mobilized by a major fire in a MOX fuel fabrication plant, but in the former case the *extra* hazard posed by the plutonium probably would be smaller than in the latter case because the fire in the vitrification plant would also mobilize significant quantities of fission products. Similarly, the risk of worker radiation exposure from accidental criticality in the course of plutonium-oxide production would be added, in a vitrification plant, to an already significant risk of worker radiation exposure from accidents in the handling of fission products, to which there would be no counterpart in the MOX fuel fabrication plant.

seems otherwise desirable. As for the other ES&H impacts of the addition of plutonium to the vitrification operation (aside from those of metal-to-oxide conversion, which is common to vitrification and most spent fuel options and was discussed separately above), these effects seem unlikely, for the reasons given earlier, to represent large additions to the ES&H impacts that will be encountered in any case in the use of vitrification to stabilize military HLW.

Elimination Options

When and if it is desired to destroy, by fission, a larger fraction of the WPu than is consumed in the once-through reactor operations associated with the spent fuel option, there will have to be added to the plutonium-processing operations described above the additional operations associated with reprocessing spent fuel. (As noted earlier, fissioning more than about 80 percent of a given initial quantity of plutonium can only be achieved by repeated recycle of the plutonium through reactors, and this in turn requires reprocessing.)

Nuclear fuel reprocessing, if carried out using technologies of the type that have been developed on a commercial scale up until now,⁴⁴ would entail routine exposures of workers and the public to radiation that, while presumably within regulatory standards, would not necessarily be as small—per unit of electricity generated—as the analogous exposures from other operations in the commercial nuclear fuel cycle (USNRC 1976, APS 1978, NAS 1979). The plutonium that emerges from these reprocessing operations, moreover, will be more similar to reactor-grade plutonium than to weapons-grade plutonium in isotopic composition, hence will amplify the hazards ascribed above to MOX fuel fabrication with the weapons-grade material. On the other hand, plutonium recycle would reduce the amount of uranium mining and milling required to generate a given quantity of electricity, and hence would reduce the ES&H impacts of those operations. (Whether reprocessing and plutonium recycle would reduce waste management burdens is unclear; see “Radioactive Waste Issues” below.)

Even if reprocessing and recycle of plutonium turn out, on balance, to increase the total worker and public exposures per unit of nuclear electricity generation, that is not to say that the additional exposures from reprocessing and from fuel fabrication with recycled plutonium would exceed regulatory standards (the plants will have to be designed to avoid that) or that the public will deem the exposures unacceptable (it is not easy to judge, in advance, what exposures the public will deem acceptable in exchange for what benefits); it is only to say that their being potentially non-negligible compared to the exposures

⁴⁴ These are based on the PUREX process, which entails mechanical chopping up of the fuel elements (after a cooling-off period of months to years following discharge from the reactor), followed by dissolution in acid and a series of chemical operations to separate the fission products from the actinides and, within the actinides, the plutonium from the rest (see, e.g., APS 1978).

from other nuclear-energy operations means that careful attention should and will be paid to them and the means for their minimization. This, in turn, could be a source of delay before such activities are licensed in the United States—they already are licensed in some other countries—as well as a source of additional costs.

As noted in Chapter 4, some of the advanced-reactor options for plutonium destruction would employ improved reprocessing technologies that are less developed than the PUREX approach now in use where reprocessing is practiced. Among the possibilities are the pyrometallurgical processing techniques envisioned for use with the integral fast reactor (IFR) and the online molten-salt reprocessing technology that might be developed for an accelerator-based conversion (ABC) option employing a molten-salt blanket. These approaches appear to offer some advantages in the security realm compared to conventional reprocessing technology. Perhaps they will offer ES&H advantages, as well, but this will be difficult to confirm until these approaches have been further developed and tested.⁴⁵

It is difficult, in any case, to compare the ES&H hazards of plutonium reprocessing and recycling for WPu disposition with the ES&H hazards that would be associated with civilian and military nuclear-energy activities in the absence of WPu disposition, because of the way in which disposition of WPu and the management of civilian plutonium are linked. If plutonium reprocessing and recycle were practiced only for excess WPu, then even if the ES&H hazards of reprocessing and recycle were a significant addition to other nuclear fuel-cycle risks on a per-unit-energy basis, the small quantity of WPu compared to the scale of civilian nuclear-energy generation would ensure that the plutonium's contribution to total exposures would be modest. Similarly, if society were reprocessing and recycling civilian plutonium on a large scale anyway, then the addition of the WPu would not produce a significant increment to the ES&H hazards of those civilian operations.

But neither of these scenarios seems very relevant: as discussed elsewhere in this report and in the report of the full committee (NAS 1994), there is little security benefit in actually fissioning a large fraction of the atoms of excess WPu (as opposed to merely embedding the WPu in forms that meet the spent fuel standard) if the much larger inventory of civilian plutonium is not fissioned; on the other hand, there is currently no economic incentive in the United States to reprocess and recycle civilian plutonium for energy generation, and there are reasons of proliferation policy not to do so. There will probably not be a cost incentive until uranium is several times more costly than it is today. Reprocessing and recycling plutonium, then, is unlikely to be undertaken in the United

⁴⁵ The forthcoming report of the National Research Council's Panel on Separations Technology and Transmutation Systems (STATS) treats the possibilities in considerably more detail than is possible here (National Research Council forthcoming).

States in the next few decades *unless* it is for purposes of making the plutonium less accessible for weapons use than it would be in spent fuel, and *unless* it is done for civilian as well as for WPu. That this might entail ES&H impacts that are not insignificant compared to those of operating civilian nuclear-energy generation without reprocessing and recycle is a potential liability which, like the additional costs and time requirements of this approach, weighs against the security gain from actually destroying most of the plutonium.

Reactor Safety Issues

The potential influences on safety of the use, in LWRs, of MOX fuel containing reactor plutonium were extensively studied in the United States in the 1970s, when large-scale use of this technology was being contemplated for commercial electric-power production (USNRC 1976). These influences have also been studied in Europe (where considerable operating experience with one-third MOX cores in LWRs has been accumulated), in Japan, and in Russia;⁴⁶ and safety issues for liquid-metal reactors (LMRs) fueled with reactor plutonium have likewise been investigated quite extensively in the United States, Europe, Japan, and Russia, not only in theoretical studies but also in large-scale experiments and in prototype-reactor operations (see, e.g., Planchon et al. 1987, D. Lucoff 1989, Atomic Energy Society of Japan 1992).

In connection with recent interest in the use of reactors for disposition of WPu, additional studies of the associated reactor safety issues for reactors and fuels of various types have been undertaken by reactor manufacturers (e.g., in the vendor reports for the Plutonium Disposition Study of the U.S. Department of Energy, summarized in USDOE 1992), in part for the purpose of clarifying ways in which the different isotopic composition of weapons-grade as opposed to reactor-grade plutonium could affect safety characteristics. Before reactors are licensed to operate with weapons-grade plutonium in any country, moreover, it can be assumed that there will be a further reexamination of the safety implications by the relevant national reactor safety authorities.

We cannot comprehensively summarize, in the time and space available here, the findings of the studies that have been conducted up until now of the safety of the use of plutonium fuels, much less anticipate in detail what may be found in the further layers of studies that will be done before reactors are actually used for WPu disposition. We merely sketch out, in what follows, enough of an overview of the issue to support our belief that it is likely, based on current information, that power reactors of a variety of designs will prove to be operable with WPu fuels without adding significantly to the safety risks that would be associated with operating the same reactors with uranium fuels. (Our primary focus is on LWRs, however, about which the most is known, and which are the

⁴⁶ See, e.g., Kudriavtsev (1993), Schlosser et al. (1993), Shiratori et al. (1993), Levina et al. (1994), Mikhailov et al. (1994), Murogov et al. (1994), and Novikov et al. (1994).

most likely to see use for disposition of WPu in the relatively near future.) We note that the increase in safety risks from operating with WPu fuel would have to be very substantial in order for such operations, in the relatively few reactors that would be needed for the disposition of 50 or 100 tons of surplus WPu, to represent a significant increase in the total risk from civilian nuclear-energy generation—that is, in order to violate our third criterion for the ES&H characteristics of WPu disposition.

The risk from reactor accidents is a matter of both probabilities and consequences—the probabilities of various combinations of mechanical and human failures and the consequences to be expected when and if these failure modes actually occur. The influences on reactor safety from the substitution of WPu for U-235 in fresh fuel—whether as MOX in the case of LWRs, CANDUs, and some LMRs, or as metal in other LMRs, or as pure plutonium oxide in the MHTGR—could take the form of changes in probabilities of particular accident sequences or changes in the consequences of accidents of particular types. Most studies of the reactor safety implications of the use of plutonium fuels have focused predominantly on the potential effects on accident probability, on the supposition that the effects on accident consequences are modest at most. In what follows, we consider probabilities first and then turn briefly to consequences.

Effects on Accident Probabilities

The two main classes of nuclear-reactor accidents with the potential to release significant quantities of radioactivity to the environment are reactivity excursions and loss-of-coolant/loss-of-coolant-flow accidents. The Chernobyl accident in 1986 was of the first type; accidents in this category arise when the nuclear chain reaction in part or all of the reactor core accelerates out of control to reach rates of energy release exceeding the core's capacity to absorb and remove heat. The Three Mile Island accident in 1979 (which, unlike Chernobyl, did not produce a large release of radioactivity to the environment, but which did destroy the reactor core) was of the second type; accidents of this type arise not from an unexpectedly high rate of energy release in the core but from an unexpected reduction in the capacity of cooling systems to remove the decay heat generated in the core (as, for example, when a faulty valve or a break in a coolant pipe allows a substantial fraction of the primary heat-transfer medium to escape and the emergency coolant injection systems fail to replenish the escaping coolant).⁴⁷

⁴⁷ In the most probable loss-of-cooling-accident scenarios, in fact, the main source of energy leading to overheating of the core after cooling fails is not the energy from the nuclear chain reaction, which usually would be promptly quenched. The problematic energy comes rather from "afterheat"—the result of the radioactive decay of fission products, which diminishes over time according to their half-lives but cannot be shut off more rapidly.

As noted in Chapter 2, the characteristics of plutonium in a chain reaction are subtly different from those of uranium in ways that influence the task of protecting against accidents of the reactivity-excursion type: plutonium has a smaller delayed-neutron fraction and a higher thermal-neutron absorption fraction than uranium, both of which factors increase the need for control-absorbers in a plutonium-fueled reactor core compared to what is needed when uranium-based fuel is used; and Pu-239 has a resonance in its fission cross-section that produces a tendency toward a positive temperature coefficient of reactivity that is not present in uranium fuel.⁴⁸ It was also noted in Chapter 2 (see also Wiese 1993) that MOX fuel exhibits levels of radioactive afterheat modestly higher than those exhibited in LEU fuel irradiated to the same burnup, which affects the task of protecting against core damage in the event of loss-of-coolant accidents. Quantitative analysis of these phenomena, however, combined with operating experience using MOX fuels in a number of countries, has shown that most LWR designs can accommodate MOX fuel made from reactor plutonium in at least one-third of their cores, without modification to the reactor, while remaining well within the capabilities of their control systems to safely limit reactivity excursions and the capabilities of their cooling systems to keep the fuel within safe thermal limits.

As discussed in Chapter 4, the prospect of the availability of significant quantities of WPu for use as nuclear fuel following the end of the Cold War led, at the beginning of the 1990s, to a number of studies of the adaptability of LWRs to the use of MOX fuel made from this WPu rather than from the reactor-grade plutonium (RPu) that theretofore had been studied and, to a modest extent, utilized. These initial studies (see, e.g., Omberg and Walter 1993, USDOE 1993a) assumed that most LWR types already in operation would be able to use safely about the same fraction of WPu-MOX fuel in their cores as RPu-MOX (that is, about one third), possibly with some addition of burnable neutron absorbers to compensate for the increased proportion of Pu-239 (with its 0.3 electron volt resonance in fission cross-section) in WPu. This addition could be accomplished in the manufacture of the fuel and would not necessitate modifications to the reactor itself.

⁴⁸ A resonance is a sharp peak in the value of a cross-section—that is, in the probability of a particular reaction—in a narrow range of values of the relative velocity between the target nucleus and an incident neutron. The key resonance in the fission cross-section of Pu-239 occurs at a neutron energy of about 0.3 electron volts, which is well above the average energy of the neutrons in the core of a normally operating LWR reactor. Heating the core results in an increase in the number of neutrons energetic enough to experience the resonance, and it also widens that energy range through the phenomenon called Doppler broadening. Both effects tend to increase the reaction rate as the temperature of the core increases, which if not compensated by other effects acting in the opposite direction would produce what is called a positive temperature coefficient of reactivity (see also Chapter 2).

It was further assumed that most existing LWRs, if they were able to utilize MOX fuel in 100 percent of their cores safely, would need modifications to increase the capabilities of their control systems. (As noted earlier, one existing U.S. LWR type—the ABB-Combustion Engineering System-80, of which three are in operation in Arizona and another has been mothballed in a partial state of completion in Washington state—was designed from the outset to be able to use a 100-percent MOX core. The vendor has indicated that this capability would apply to WPu as well as to RPu; see ABB-CE 1993.) Any new LWR constructed for the purpose of WPu disposition could be designed to take a 100-percent MOX core, as were the “evolutionary” and “advanced” LWR designs presented by vendors in the first phase of the U.S. Department of Energy’s Plutonium Disposition Study (USDOE 1993a).

Analyses performed by vendors in the second phase of that study, which became available late in the panel’s deliberations, suggest—contrary to previous assumptions—that several existing LWR types besides the System-80 could in fact use 100-percent WPu-MOX cores without undergoing significant modifications and without compromising safety. Further analysis and review will be required before this conclusion can be considered firm. In any case, the still unfolding understanding of the circumstances under which existing LWRs would be able to use 100-percent WPu-MOX fuel only underlines the key point that *any* approach to the use of MOX fuel in U.S. power reactors must and will receive a thorough, formal safety review before it is licensed.

While we are not in a position to predict what if any modifications to existing reactor types—or to the designs of newer types that have not yet operated—will be required as a result of such licensing reviews, we expect that the final outcome will be certification that whatever LWR type is chosen will be able, with modifications if appropriate, to operate within prevailing reactivity and thermal margins using sufficient plutonium loadings to accomplish the disposition mission in a small number of reactors. We believe, further, that under these circumstances no important overall adverse impact of MOX use on the accident probabilities of the LWRs involved will occur; if there are adequate reactivity and thermal margins in the fuel, as licensing review should ensure, the main remaining determinants of accident probabilities will involve factors not related to fuel composition and hence unaffected by the use of MOX rather than LEU fuel.

Addressing the implications for accident probabilities of using evolutionary or advanced reactor types other than LWRs for WPu disposition is more difficult, because, necessarily, it is difficult to compare the safety of reactors that have not yet been built with the safety of those that have. (It is usually easy enough to identify some features of alternative reactor types that complicate the task of ensuring safety, compared to the situation with LWRs, while identifying other features that ease the task of ensuring safety; what is hard is predicting what the *net* difference in safety will be, all things considered.) There is, none-

theless, reason to think that advanced reactors for energy supply will not be deployed unless society concludes—through analysis, testing, debate, and the licensing process—that they are at least as safe as current reactors. And since, as we have argued elsewhere in this report, there is no compelling reason to deploy advanced reactors solely for the purpose of WPu disposition, there are only likely to be available for that purpose if they have also passed the safety “test” for use in large-scale electricity generation. If that is so, their use for WPu disposition seems unlikely to so alter their safety characteristics as to perturb significantly the safety of the whole nuclear-energy enterprise, all the more so because advanced-reactor types henceforth will probably all be designed from the outset to minimize any safety problems of plutonium use.

Effects on Accident Consequences

In the event of a large release of radioactivity as the result of a severe nuclear-reactor accident, the consequences generally will include relatively large doses of radiation (say, above 0.1 Sv or 10 rem) delivered in the hours and days immediately after the accident to relatively small numbers of people who are in the path of the radioactive plume near the reactor, smaller doses (0.01–0.1 Sv or 1–10 rem) delivered over a longer period of time to larger numbers of people in the plume path out to distances of some hundreds of kilometers from the reactor, and very small doses (less than 0.01 Sv or 1 rem) delivered to much larger numbers of people at even greater distances and lower dose rates (see, e.g., APS 1975, USNRC 1975, 1987; Hohenemser 1988.) The dominant mechanism for the close-in exposures is inhalation of the mixture of suspended radionuclides in the plume, with a substantial additional contribution by external irradiation from radionuclides deposited on the ground. With increasing distance from the reactor, inhalation from the plume becomes relatively less important and external irradiation from ground-deposited material—along with inhalation of resuspended material and ingestion of radionuclides in food and water—become more important.

It is to be expected that about 95 percent of the total population exposure in the 50 years following a severe accident (measured in person-sieverts or person-rem, the product of the total number of persons exposed times the average exposure received) would result from one or another form of “ground dose”—external irradiation and inhalation of resuspended material—and from ingestion with food and water. The doses from inhalation of radionuclides from the passing plume and from external irradiation as the plume passed would contribute less than 5 percent to the total population dose. Most of the population dose would be distributed over a large number of people, at substantial distances from the reactor, each experiencing a low individual dose. The details of this pattern of doses would depend on the quantities of various radionuclides released by the accident, the temperature of the plume in which they were re-

Table 6-19 Contributions to Doses from Severe Reactor Accidents

Elements	Calculated LWR Release Fractions		Estimated Chernobyl Release Fractions ^c	Nominal Severe-Accident ^d 50-year Population Dose (10 ⁴ pers-Sv) Delivered Via:		
	WASH 1,400 ^a	NUREG 1,150 ^b		Plume Inhalation	Ground Dose	Ingestion
Kr,Xe	0.8–1.0	0.9–1.0	1.0	negl	0	0
I	0.4–0.9	0.2–0.9	0.2–0.6	0.5	1	1–5
Cs	0.4–0.5	0.05–0.8	0.1–0.6	0.7	50	20–50
Te	0.3–0.7	0.01–0.7	0.1–0.6	0.3	small	1–2
Sr	0.05–0.1	0.005–0.3	0.04–0.06	0.6	small	2–20
Ba	0.05–0.1	0.005–0.3	0.04–0.06	0.3	1	1–4
Ru,Mo	0.02–0.5	0.001–0.1	0.02–0.03	small	1	small
Pu,Am,Cm	3×10 ⁻³ /5×10 ⁻³	1×10 ⁻⁵ /0.05	0.03	0.3	1	1–5

ELEMENTS:

Kr, Xe: krypton, xenon.

I: iodine.

Cs: cesium.

Te: tellurium.

Sr: strontium.

Ba: barium.

Ru, Mo: ruthenium, molybdenum.

Pu, Am, Cm: plutonium, americium, curium.

^a USNRC (1975), with ranges covering release categories PWR 1–2, BWR 1–2.^b USNRC (1987), with ranges for most severe releases considered.^c See Hohenemser (1988). The Chernobyl reactor was a graphite-moderated, light-water-cooled reactor, not an LWR, and lacked a real containment building. Releases of the indicated magnitude are considered much less probable for LWRs.^d Assumed release fractions are Kr,Xe = 0.9, I = 0.7, Cs = 0.5, Te = 0.3, Ba,Sr = 0.06, Ru,Mo = 0.02, Pu,Am,Cm = 0.004. Total population exposure is about 100 × 10⁴ pers-Sv or 100 × 10⁶ pers-rem, divided about equally between ground dose and ingestion dose. Dose contributions estimated based on APS (1975), USNRC (1987), and Hohenemser (1988).

leased, the weather conditions at the time of the accident and afterwards, the distribution of population in relation to the location of the reactor and the wind direction following the accident, the habits of the population (fractions of time spent indoors and outdoors, diet), and so on.

We assume, consistent with the discussion in the preceding section, that once an accident of a given type occurs the conditions governing what proportions of the radionuclides in the reactor will be released will *not* depend on the percentage or type of plutonium in its fuel. (That is to say, for example, that the modest differences in afterheat among fuels with different quantities and isotopic qualities of plutonium are not big enough, in relation to other factors affecting the post-accident condition of the fuel, to affect significantly the release fractions—i.e., the percentages of the various classes of elements present that

actually escape the reactor into the environment.) In that case, the only factor governing accident consequences that *does* depend on how much and what kind of plutonium was used in the reactor fuel is the inventory of radionuclides in the reactor at the time of the accident.

We are interested, particularly, in the quantities of those radionuclides that would tend to make important contributions to the total population exposure resulting from severe accidents. Previous studies have estimated release fractions, for different classes of elements in the fuel, that might be possible in severe accidents at LWRs (see, e.g., USNRC 1975, 1987).⁴⁹ Combining such release-fraction estimates with population-dose models indicates that the largest contribution to the population dose comes from cesium isotopes, followed in importance by strontium, iodine, plutonium and isotopes derived from it, barium, ruthenium, and tellurium, as shown in Table 6-19. Fission of plutonium produces about the same quantity of cesium as does fission of uranium, so the use of plutonium-based fuel will not change very much the releasable inventories of this dominant contributor to accident consequences. Plutonium fission produces somewhat less strontium than does uranium fission, which effect would contribute to a modest reduction in the consequences of a release (for fixed release fractions) from plutonium-based fuel, but the increased quantities of plutonium itself in such fuel—typically 1.5–5 times more than in uranium-based LWR fuel just before discharge—could offset this advantage partly or entirely.

Based on the foregoing considerations and the figures in Table 6-19, it seems unlikely that the switch from uranium-based to plutonium-based fuel could worsen the consequences of a postulated (and very improbable) severe accident in a LWR by more than 10 or 20 percent.⁵⁰ The influence on the consequences of less severe accidents, which probably dominate the expectation value of population exposure per reactor-year of operation (USNRC 1975, 1987), would be even smaller, because less severe accidents are unlikely to mobilize any significant quantity of plutonium at all.

⁴⁹ Notwithstanding significant differences in reactor characteristics, these calculated severe-accident release fractions for LWRs are in rather good agreement with estimates of what was actually released in the April 1986 accident at the graphite-moderated, water-cooled Chernobyl reactor near Kiev (see Table 6-19).

⁵⁰ Using the geometric means of the ranges of values in Table 6-19 for the contributions of different elements to the 50-year population dose from a hypothetical severe accident at an LEU-fueled LWR, one concludes that strontium contributes about 6×10^4 person-sieverts and plutonium and the isotopes derived from it contribute about 4×10^4 person-sieverts to a total, for the accident, of about 105×10^4 person-sievert. Based only on changes in the fuel composition, with release fractions for each element held constant, the substitution of MOX for LEU fuel would shrink the strontium dose about threefold while increasing that from plutonium, in the highest plutonium case, about fivefold. This would produce a net 10-percent increase in the total population dose. Saying, as we do, that an increase of “more than 10 or 20 percent” is “unlikely” allows some leeway for the considerable uncertainty in the severe-accident release fractions for both plutonium and strontium.

We expect that preliminary conclusions of this sort would be revisited in the course of more detailed safety calculations performed for licensing purposes, with up-to-date calculations of radionuclide inventories for different fuel compositions and irradiation histories, and reexamination of population-exposure models for sensitivity to changes in those inventories resulting from the use of WPu in the fuel. We endorse such further investigation, but we would consider it very surprising if the results altered our conclusion that use of plutonium fuels will not change by very much the consequences of such LWR accidents as may occur. Although accident modes and probabilities may differ among reactor types, moreover, it is hard to see how use of WPu in the fuel of, for example, liquid-metal reactors or high-temperature gas reactors could have a large influence on the consequences of accidents with these reactors. And unless there is a *very* large such influence, the use of WPu in a modest fraction of the world's operating power reactors, of whatever types, could not affect significantly the accident risks of the nuclear-energy system as a whole.

Radioactive Waste Issues

All options for the disposition of plutonium from surplus nuclear weapons will generate radioactive wastes of one or more types, and some options will affect the quantities or other characteristics of radioactive wastes that have been or will be produced by other military and civilian nuclear-energy activities. As a basis for surveying the radioactive waste implications of alternative approaches to plutonium disposition, the categories according to which radioactive wastes are described and regulated in the United States—and the approaches for managing the wastes in these categories—can be summarized as follows (APS 1978; OTA 1985, 1989; Holdren 1992; OFR 1992, Pts. 61.2 and 72.3):

- *Spent fuel* consists of fuel rods that have been removed, after nuclear-energy generation, from commercial, defense, or research reactors. It is initially highly radioactive, generates considerable heat, and requires heavy shielding. The volume of spent fuel is 20–30 m³ per year from a large (1,200-MWe) LWR. Current practice is interim storage of the spent fuel in suitably designed pools of water (providing cooling and shielding) at reactor sites, followed in some cases by a second interim storage phase in dry casks. In “once-through” fuel cycles, wherein reprocessing of spent fuel is foregone, the expectation is that such interim storage of the spent fuel would ultimately be followed by emplacement in a geologic repository.
- *High-level wastes* (HLW) are defined, in current U.S. regulatory practice, as the concentrated radioactive waste forms generated when spent

fuel is reprocessed to recover uranium and plutonium.⁵¹ HLW are initially highly radioactive, generate considerable heat, and require heavy shielding. A distinction is made between “defense” and “commercial” HLW, according to whether the material originates in military activities (weapons production and naval reactor operations) or commercial electricity-production activities. The physical form of such wastes depends on the reprocessing technology used and on the choices made about how to prepare the waste for ultimate disposition. Most U.S. defense HLW were initially in the form of slurries, as were the much smaller quantities of commercial HLW produced in this country before commercial reprocessing was suspended. The slurries have since separated into solid and liquid phases. The HLW are expected to be vitrified into highly leach-resistant glass logs—as is already being done in a few other countries—before emplacement into a geologic repository. The volume of HLW resulting from reprocessing the fuel from a 1,200-MWe LWR and vitrifying it would be 5–8 m³ per year, including the glass matrix and metal canisters.

- *Uranium mill tailings* are the residues from the process by which uranium is extracted from mined uranium ore. They contain very low concentrations of radioactive material, but the half-life of the nuclide that governs the longevity of their hazard (thorium-230, parent of radon-226) is about 80,000 years and the volume of these wastes is very large—100,000–200,000 m³ per 1,200-MWe LWR per year for uranium ore of typical characteristics, assuming a once-through fuel cycle, and about 30 percent less for a reprocessing and recycle fuel cycle. (This radioactivity was in the ground all along, but was brought to the surface by mining and rendered more mobilizable by milling.) Current U.S. regulations require covering of newly produced mill tailings with soil and rock to reduce radon release, control erosion, and minimize water infiltration.
- *Low-level wastes (LLW)* are defined in the United States to mean all radioactive waste that is *not* spent fuel, HLW, or uranium mill tailings. LLW originate in nuclear-weapons-related activities, in commercial nuclear fuel-cycle operations (e.g., uranium enrichment, nuclear fuel fabrication, reactor operation and decommissioning, spent fuel handling, and fuel reprocessing), and in medical, industrial, and research applications of radionuclides. The quantities of LLW being generated in the United States in the late 1980s from the operation of LWRs and

⁵¹ The fission products that dominate the radioactivity of HLW are the same ones that were in the spent fuel; only the physical and chemical matrix in which they are embedded has changed. Some authorities define HLW more broadly, as an overarching category that includes both spent fuel and the concentrated waste forms arising from reprocessing.

associated fuel-cycle activities amounted to about 250 m³ per 1,000-MWe PWR per year and 750 m³ per 1,000-MWe BWR per year, not including decommissioning wastes (which would add about 500 m³ of LLW per year, prorated over a nominal 30-year reactor lifetime, for either reactor type) and not including LLW from reprocessing (which, if carried out, would add another 150–300 m³ of LLW per reactor-year).⁵² Most types of LLW in the United States have been disposed of by shallow burial at a modest number of sites licensed for this purpose. LLW containing concentrations of transuranic and certain other long-lived radionuclides above specified limits are not acceptable for shallow burial under current U.S. regulations, and such wastes are being held in interim storage pending the availability of a geologic repository for them.

- *Transuranic (TRU) wastes* are wastes, other than spent fuel or HLW, that contain more than 100 nCi/g of radionuclides with atomic number greater than 92. TRU wastes may be generated in the production and handling of plutonium for nuclear weapons, in the manufacture of sealed radioactive sources, and in the refurbishing or decommissioning of nuclear power plants. TRU wastes are considered a subcategory of LLW. They amount to 100–150 m³/yr for a 1,200-MWe LWR with reprocessing and recycle (essentially entirely from the reprocessing and MOX fuel fabrication operations), none from an LWR using LEU fuel once-through. Although some TRU wastes were formerly disposed of in the United States by shallow burial, this practice is no longer permitted; these wastes are now being held for eventual emplacement in deep geologic storage.

A site at Yucca Mountain, Nevada, is under investigation by the U.S. Department of Energy for use as a mined geologic repository for civilian spent fuel, vitrified defense HLW, and probably civilian TRU waste; actual emplacement of waste at this site, which is in a region of unsaturated volcanic tuff, is not likely to begin before 2015, if then. A mined repository in salt beds near Carlsbad, New Mexico—the Waste Isolation Pilot Plant (WIPP)—probably will be in operation considerably sooner and is to be used mainly for disposal of defense TRU wastes. A few shallow burial sites are in operation at locations around the United States for those parts of commercial and defense LLW that qualify for such treatment; the capacities of the existing sites and the locations of new ones are under continuing and often contentious discussion.

In the subsections that follow, we summarize the ways in which the two primary classes of options discussed in this chapter—(1) once-through irradiation in MOX fuel in reactors of current commercial types and (2) vitrification

⁵² See USDOE (1988a). Current LLW volumes per reactor-year may be smaller because of the incentives for compaction produced by a shortage of licensed LLW disposal sites.

TABLE 6-20 Effects of Plutonium Disposition on Radioactive Wastes

Waste Type	Disposition Option	
	MOX to Spent Fuel ^a	Vitrification with HLW ^b
Spent Fuel	Quantity unchanged, plutonium content increases significantly, fission-product composition changes slightly	No spent fuel involved
HLW	No reprocessed HLW involved	Number and size of HLW-bearing glass logs unchanged, plutonium content increases greatly
TRU	Additional TRU produced in conversion of plutonium metal to oxide and fabrication of MOX fuel	Additional TRU produced in conversion of plutonium metal to oxide and in vitrification process
non-TRU LLW	Some additional plutonium-bearing LLW below 100 nCi/g of TRU isotopes may be produced in plutonium conversion and MOX fabrication	Some additional plutonium-bearing LLW below 100 nCi/g of TRU isotopes may be produced in plutonium conversion

^a We assume the MOX is used in reactors that would be operating in any case, so total amount of nuclear electricity generation is unchanged by this choice of disposition option.

^b We assume that a sufficient plutonium loading is possible from the criticality standpoint to permit accommodating all of the surplus WPu without increasing the number of glass logs already scheduled to be produced for the stabilization of existing defense HLW.

together with military HLW—would affect the quantities and other relevant characteristics of the radioactive wastes in the foregoing categories that would be associated with civilian and military nuclear-energy operations in this country in the absence of WPu disposition. (The main effects are summarized in general terms in Table 6-20.) Implications for radioactive wastes of alternative options are considered more briefly.

Spent Fuel and High-Level Wastes: Dose Potentials

Disposition of WPu by fabricating it into MOX fuel and irradiating that fuel once-through in power reactors of existing commercial types (e.g., LWRs or CANDUs) would produce spent fuel generally similar to the spent fuel that these same reactors would be producing if operated with their usual uranium-based fuels. The characteristics of spent fuel that govern the magnitude of the waste management task and the risks associated with it are mainly its volume, its fission-product content, its content of actinides (including especially those that are fissile), and the chemical and structural characteristics that influence the rate at which it would release these radioactive inventories under conditions encountered in transport and storage.

The volume of spent fuel associated with a given amount of nuclear-energy generation depends only on the burnup, which, in LWRs, is not expected to change as a result of the incorporation of WPu in the fuel. In CANDUs, which ordinarily use unenriched uranium fuel, the burnup would probably be increased—and the spent fuel volume for a given amount of energy generation correspondingly decreased—if WPu were used in the fuel, since exploiting the higher burnups made possible by the addition of the plutonium would partly offset the higher fuel fabrication costs. Thus, disposition of WPu by means of the MOX option in reactors of existing commercial types would not increase spent fuel volume if LWRs were used, and would decrease this volume if CANDUs were used, assuming only that the disposition occurs in reactors that would have operated in any case. If new reactors were commissioned for the purpose of WPu disposition, of course, there would be a net increase in spent fuel volume; but this increase would be just proportional to the fraction of total nuclear electricity generation accomplished by the new reactors, and, as we have seen, the extra electricity generation needed to process 50 or 100 tons of WPu would be only a modest addition to the nuclear energy generation expected to occur in any case over the same period.

The quantity of fission products in spent fuel depends mainly on the total amount of nuclear energy generated from that fuel—that is, on the burnup—and, secondarily, on the neutron-energy spectrum and on how much of the energy came from fissioning U-235, how much from fissioning Pu-239, and so on. The differences in fission-product production between uranium-based fuels and MOX fuels are quite modest (all the more so because, after all, a significant part of the fission even in uranium-based fuels occurs in plutonium that has been produced in these fuels by neutron absorption in U-238), and, in particular, the inventories of the fission-product isotopes that contribute the most to the radiological risks from spent fuel are practically the same in MOX as in uranium-based spent fuels. Thus, as with the volume of the spent fuel, the quantities of important fission products it contains would not be much affected by the use of the MOX fuel option for WPu disposition. (The most important such change might well be the *reduction* in the quantity of strontium-90 in the spent fuel, compared to that from using LEU in the same reactors, as discussed above in connection with hazard potential from reactor accidents.)

The quantity of actinides in spent fuel *would* be substantially different under the MOX option for WPu disposition than for nuclear-energy generation with ordinary uranium-based fuels, however. Typical spent LWR fuel contains about 1 percent plutonium and typical spent CANDU fuel about 0.4 percent; the spent fuel under the MOX option for WPu disposition would contain, for reasonable initial plutonium loadings, from 2.5–5 percent plutonium if LWRs were used and 0.8–1.4 percent if CANDUs were used (see Table 6-1). The amount of americium in discharged MOX fuel would also be greater than in LEU fuel at the same burnup. On the other hand, spent LWR fuel from the MOX option

would contain considerably less of the important actinide, neptunium-237 (Np-237), than ordinary spent LWR fuel, because Np-237 is produced by successive nonfission neutron captures by U-235, of which there is several times more in the uranium-based fuel than in the MOX. (If CANDUs were used, the magnitude in the change of Np-237 content in spent fuel from MOX compared to spent fuel from the usual natural uranium CANDU fuel would depend on whether the MOX was made from depleted or natural uranium.)

The spent fuel characteristics governing releasability of the contained fission products and actinides must be considered in the context of the specific release modes that could be responsible for significant doses to humans. Two such modes are generally assumed to constitute the main hazards from geologic repositories:

- The release mode generally considered to be the most troublesome one is dissolution of the radioactive materials into groundwater that has entered the repository, followed by hydrogeologic transport of the radionuclides into an aquifer used by humans for drinking or irrigation.
- The second release mode usually taken into account is inadvertent intrusion into the repository in connection with mining operations by a future society unfamiliar with the nature or location of the radioactive waste. In such a scenario, waste-bearing material might be handled by miners and ore processors, as well as being piled up on Earth's surface where it could be further dispersed by wind, rain, and surface water.

The details of the chemical properties of the waste are more important in the first case than in the second, where the radiologic potency per unit volume of waste is the key factor.

In the groundwater-intrusion/hydrogeologic-transport release mode, the potential for doses to the public is dominated by relatively soluble long-lived fission products—technetium-99 (half-life 211,000 years), iodine-129 (half-life 15.7 million years), and cesium-135 (half-life 2.3 million years)—the inventories of which are about the same in MOX as in LEU fuel. The rate of dissolution of these radionuclides into groundwater is determined in part by the rate of oxidative solid-state alteration of the fuel matrix (Sadeghi et al. 1991). Although the fuel matrix of MOX fuel is mainly uranium dioxide, the same as for uranium fuel, experimental investigation would be necessary to determine whether the dissolution rates of these fission products is affected by the larger quantities of plutonium and by the presence of distinct grains of plutonium oxide in the MOX fuel. Thermodynamic considerations suggest that the dissolution rates of the soluble fission products could be *less* than for LEU fuel.

The higher actinide content of the MOX fuel as compared with LEU fuel does not contribute directly to the risk from the groundwater-intrusion/hydrogeologic-transport release mode, in the form of extra actinide contributions to the potential doses, because the rate of removal of actinides from the emplaced

wastes is limited by the solubility of the actinides in the local groundwater conditions rather than by the concentration of actinides in the wastes or the rate of alteration of the waste matrix. There could, however, be an indirect effect in which a higher actinide concentration enhances the rate of alteration of the fuel matrix and hence the rate of dissolution of the fission products by virtue of the extra heat generation contributed by the actinides. This effect could only be significant in the long term—that is, at times beyond 50 years from the time the fuel is discharged from the reactor—because at shorter times the heat generation is dominated by the fission products. It is not even completely clear that a higher heating rate is always disadvantageous: it would tend to accelerate the processes tending to break down the fuel matrix over time, but in unsaturated conditions the extra heat could actually confer some advantage in helping to maintain dry conditions in the repository. In some repository concepts, it appears that the increased alpha activity resulting from the higher concentration of plutonium in MOX fuel could modestly increase the dissolution rate of solubility-limited nuclides.

In the case of the mining intrusion scenario for waste mobilization, the transport of plutonium in surface water exposed to the exhumed waste would still probably be solubility-limited, hence not worse for MOX than for LEU spent fuel, but the hazard from weathered airborne plutonium could increase by a factor of three to five for MOX fuel compared to LEU, depending on the plutonium loading of the former. For a repository to be considered satisfactory even for LEU fuel, however, the probability of intrusion by future miners needs to be very low, and given a suitably low intrusion probability the overall hazard from the repository is likely to be dominated by the fission products in the groundwater/hydrogeologic transport scenario—which, as noted above, will *not* depend strongly on whether the spent fuel is MOX or LEU.

In the case of the vitrification option, the disposition of 50 tons of WPu could be accomplished through the addition of 1.35 weight percent plutonium to $50/0.0135 = 3,700$ tons of borosilicate glass, which would correspond to 2,200 of the 1,680-kg glass logs scheduled to be produced in the large melter at Savannah River. (This combination of plutonium concentration and number of logs is only illustrative; larger numbers of logs at lower plutonium concentration, or smaller numbers at higher concentration, are also possible, subject to the criticality constraint discussed below. The total number of 1,680-kg logs needed to accommodate all of the U.S. defense HLW has been estimated at 6,000–7,000.) These glass logs would contain about 20 weight percent fission products; their plutonium content if no WPu were added would be about 0.05 percent.

As in the case of spent fuel, the potential rate of dispersion of the plutonium contained in the glass logs, upon intrusion of groundwater into the repository, would probably be limited by the plutonium's solubility rather than by the plutonium-retention properties of the glass, and the risk to the public in the kinds of

scenarios likely to dominate the overall hazard would be controlled by the rates of release and dispersion of the fission products. There is some evidence that the relevant release rates from the glass would actually be diminished by the addition of plutonium to the glass matrix (Berkhout et al. 1993); if this is so, the use of this approach for the disposition of WPu would diminish the otherwise extant risks from defense HLW.

Spent Fuel and High-Level Wastes: Criticality Issues

As already noted, spent MOX fuel or plutonium-laden glass resulting from plutonium disposition would have higher plutonium concentrations than would the corresponding waste forms produced in the absence of plutonium disposition. Unless the effect of this additional plutonium is balanced by additional neutron poisons, the plutonium disposition waste forms will have higher nuclear reactivity and hence higher potential for achieving criticality accidentally.

Prevention of criticality in nuclear waste forms relies on a combination of physical configuration (geometry, dilution, spacing), neutron-absorbing materials in the waste and intervening material, and absence of an efficient moderator. (All three factors do not necessarily have to be present at once: spacing and neutron-absorbing materials suffice to prevent criticality in the pools typically employed for preliminary storage of spent fuel at reactor sites, despite immersion of the spent fuel in water—a good moderator—for purposes of cooling and shielding.) Over long spans of time in a waste repository, many factors affecting the possibility of criticality could change: radioactive decay will change the mix of fissile and neutron-absorbing materials; rock movements and migration of the wastes within the repository could change the physical configuration; differential migration could separate the fissile and neutron-absorbing materials; and intrusion of groundwater into the repository could enhance neutron moderation.

Nuclear Regulatory Commission (NRC) regulations require that waste emplacements be designed so that the effective multiplication factor in the repository (k_{eff}) be less than 0.95—meaning that a nuclear chain reaction could not be sustained—in both normal and accident conditions, and that k_{eff} would not reach 1, leading to a chain reaction, “unless at least two unlikely, independent, and concurrent or sequential changes have occurred” (OFR 1992, Sec. 60.131). It has not yet been determined, however, over what period criticality control must be guaranteed,⁵³ or what factors prospective licensees will be permitted to rely on in guaranteeing the absence of a criticality concern. Thus, crucial parts of the

⁵³ The period of potential concern extends to many millions of years. Although the main plutonium isotope, Pu-239, has a half-life of only about 24,000 years—hence would have vanished, for all practical purposes, after a few hundred thousand years in the repository—the product of its radioactive decay is U-235, which also poses a potential criticality problem and has a half-life of over 700 million years.

regulatory basis for examining the licensing of MOX spent fuel and plutonium-bearing HLW glass remain incomplete.

Criticality in the repository would not necessarily constitute a hazard to long-term public health and safety. The result, if the WPU wastes did go critical, would be to create an underground reactor, similar to the Oklo natural reactor that operated in Africa over a billion years ago (Cowan 1976). The Oklo natural reactor did not “blow up,” and neither have any plausible circumstances been identified in which a waste repository containing spent fuel or plutonium-bearing glass could do so. In a moist environment, which is the worst case,⁵⁴ it is expected that waste materials would exhibit a strong negative temperature coefficient of nuclear reactivity, meaning that the nuclear reaction would slow as it heated the surrounding material. The rate of heat generation of the critical system would then be limited to the cooling capacity of the surrounding rock. The heat generation, limited in this way, would likely be less than the fission-product decay heat at the time of emplacement of the wastes now scheduled for Yucca Mountain, and the quantity of fission products that might be generated in this way would be substantially smaller than the amount already scheduled for emplacement in Yucca Mountain.

Nonetheless, the new fission products would be generated at a time thousands of years in the future, when nearly all of the originally emplaced fission products would have decayed away, and when the various engineered barriers to release of these fission products might have failed. Hence it is prudent to avoid long-term criticality, and NRC licensing is likely to require a respectable argument that such criticality will be avoided for very long times. It appears that approaches that can provide the needed assurance are attainable, but further research is required to confirm this. In what follows, we examine the relevant factors more closely for the specific case of the proposed Yucca Mountain repository, and we identify the specific issues on which further work is required.

Waste Emplacements in Yucca Mountain. There are two distinct concepts under consideration for loading waste packages in the proposed Yucca Moun-

⁵⁴ A moist environment is the worst case because an effective moderator is required to achieve criticality at the low fissile-material concentrations characteristic of spent fuel (whether from LEU or MOX) and plutonium-bearing glass logs—see Chapter 2—and water is the best moderator that could plausibly occur in relevant quantities in a repository. (Very pure graphite is a better moderator, but there is no mechanism for such a substance to materialize unexpectedly in a waste repository.) As our report was in the final stages of preparation, a scenario was proposed by C.D. Bowman and F. Venneri of the Los Alamos National Laboratory (LANL) in which Pu-239 from spent PWR fuel, or from glass logs containing vitrified WPU, was hypothesized to separate from the accompanying materials and spread into surrounding rock in a way that would lead to supercriticality and explosive energy release of hundreds of tons of high-explosive equivalent (Bowman and Venneri 1995). A LANL technical review of the Bowman and Venneri analysis, released March 7, 1995, challenged each of the major hypotheses leading to this result and concluded that the probability of such explosions occurring was “essentially zero” (Canavan et al. 1995).

tain repository. One is borehole emplacement, wherein holes large enough to accept a waste container are drilled in the floor of mined cavities, or "drifts," in the repository. To provide for adequate conduction of decay heat into the surrounding rock, the boreholes would be separated by about 10–20 meters. With this spacing, little neutron interaction between emplaced containers is expected, so that the criticality of a single container can be examined without considering the entire system. Recently, attention is turning to drift emplacement, wherein clusters of waste containers would be loaded horizontally on the floor of drifts and later surrounded by clay or crushed rock backfill. In this case, neutron interaction between adjacent containers is likely, and, in considering criticality, all the containers in one drift would have to be considered as an integrated system.

Glass Criticality and Preferential Leaching. In current planning, criticality of HLW glass is not a major concern, as this glass would have only tiny quantities of fissile plutonium. In a glass containing one or a few percent plutonium by weight, such as might be used for disposition of WPu, neutron multiplication would be held down by the presence of large quantities of neutron-absorbing boron in the borosilicate glass. Over the very long term, however, one must consider the possibility that groundwater could intrude into the repository, that the waste container might ultimately fail, and that in the presence of water, the boron and lithium in the glass might leach away more rapidly than the plutonium. In current performance analyses of conceptual geologic repositories, the processes of dissolution⁵⁵ and mass transfer into the surrounding rock and groundwater are analyzed for times extending to hundreds of thousands of years and more. This type of analysis must be applied to these plutonium-bearing waste forms, to examine whether the boron and lithium neutron absorbers would leach away before the plutonium would, causing the neutron multiplication to increase with time.

First, it is important to consider the likely presence of water, which acts both as a leaching agent and as a neutron moderator, increasing criticality. The glass itself will have some porosity initially (and some cracking), allowing some water to intrude within the glass itself. Over time, water would react with silica and other constituents in the glass, resulting in lower-density hydrated reaction products, and potentially increasing the concentration of hydrogen atoms compared to plutonium in the glass (which is very important to the criticality of the system). The area surrounding the waste container may also fill with water, and the surrounding rock has a porosity in the range of about 15–30 percent, which may be partly or completely filled with water. Thus it is possible for a wide

⁵⁵ Here we use "dissolution" of a component to mean the net dissolution of that component, with the dissolved species either increasing the concentration of that species in the water surrounding the waste or being transported away from the dissolution front by diffusion and convection. Other writers in this field sometimes take "dissolution" to mean the reaction of the waste solid with water, even if the reaction product is a new solid phase.

range of densities and amounts of water to exist in the glass and its surroundings over thousands of years. Calculations demonstrate that neutron multiplication will be very sensitive to the amount of water present. Future calculations on these subjects should explore the various possibilities to find the worst case, which can then form a basis for design.

How fast might the boron, lithium, and plutonium in the glass dissolve? The criterion for preferential dissolution can be specified in terms of the fractional dissolution rates, that is, the ratio of the mass dissolution rate of a species to the inventory of that species in the waste. If the atom ratio of two dissolved species is the same as the atom ratio in the undissolved solid at the time of dissolution, we have "congruent dissolution." Here the fractional dissolution rates of the two species are equal, and no preferential dissolution that would increase criticality would occur. If the fractional dissolution rate of species A is greater than that of species B, the ratio of the amount of A to the amount of B in the undissolved solid decreases with time. If A is the absorber and B is the plutonium, decreasing the ratio of A to B could lead to criticality.

Laboratory experiments show that lithium in glass dissolves at a greater fractional rate than does the silica. As the silica in glass reacts with water to form a hydrated siliceous compound, the boron may be incorporated in the solid reaction product, or it may dissolve congruently with the hydration reactions of the glass. However, according to Wicks (1992) boron will be leached from the plutonium glass rubble relatively quickly.

When glass containing plutonium reacts with water to form a hydrated siliceous solid, the plutonium also reacts and forms a hydrated oxide precipitate. These solid phases will be distributed throughout the glass-water reaction products that will eventually fill the volume once occupied by the glass; because of their lower density, they will expand. Plutonium precipitates are expected to have a far lower solubility than the siliceous reaction product. Even though the inventory of plutonium is far lower than that of silica, the plutonium solubility is low enough that its net fractional dissolution rate is expected to be lower than that of silica. Predictions of the net fractional dissolution rate that appear in the literature (National Research Council 1983) are 10^{-6} per year for silica and 4×10^{-7} per year for plutonium, for a waste glass containing 0.007 percent plutonium. From these data we would expect a 140-fold decrease in the net fractional dissolution rate of plutonium for a glass containing one percent plutonium. Therefore, there would be greater chance for preferential dissolution of boron in glass containing higher concentrations of plutonium.

It is possible that most of the boron will have leached out before the Pu-239, with a half life of 24,000 years, will have decayed. If not, the issue of criticality may become that of criticality of U-235, the decay daughter of Pu-239. Even though the U-235 thermal-neutron cross section is less than that of plutonium, criticality can eventually occur if there is sufficient initial concentration of plutonium. After Pu-239 has decayed, and if sufficient boron is still present, the

possibility of later criticality will be affected by the relative fractional rates of net dissolution of boron and uranium. In the Yucca Mountain environment the solubility of uranium may be greater than that of plutonium, because of the oxidizing environment. Therefore, there may be less chance that preferential dissolution of boron will enhance criticality after the Pu-239 has decayed.

The lithium that is present in the glass may or may not form precipitates during the hydration reactions of glass and water. Most lithium compounds are soluble in water, so preferential dissolution of the lithium is also possible.

If the possibility that criticality could arise tens of thousands of years after the waste was emplaced is considered a significant problem, a number of potential solutions seem possible. One possibility would be to limit the plutonium concentration in the glass to a level low enough that the package would remain noncritical even if all of the boron and lithium in the glass leached away. Within the constraints of our study, we have only been able to begin exploring what the maximum concentration of plutonium that would be reliably noncritical in the worst case would be.⁵⁶ In a dry system, glass logs with no boron or lithium would remain non-critical even with concentrations of 3 percent or more plutonium by weight (corresponding to 60 kg or more of plutonium in each log).⁵⁷ As noted above, however, adding water to the system can greatly increase the neutron multiplication. At the other extreme, if all the glass leached away, leaving only plutonium (or its uranium decay daughter) dissolved in water in the volume that once contained the glass container, 10 kg of plutonium would be critical.⁵⁸ That system, however, may be overmoderated; there may be an intermediate water percentage in the glass that would result in even lower maximum plutonium concentrations. If we assume, for example, that for this reason only 5 kg of plutonium could be incorporated in each log, disposing of 50 tons of plutonium would require 10,000 logs, compared to a total of just over 6,000 currently scheduled for production at the Savannah River Site (McKibben et al. 1993).

Another possible approach is to add a neutron-absorbing material which, unlike boron, would not dissolve away more rapidly than the plutonium. From the theories of mass transfer described above, it is possible to design such a

⁵⁶ We would like to thank several parties for their help in providing calculations and advice on this subject, including the group at Westinghouse Hanford Company led by Ron Omberg; William Culbreth of the University of Nevada, Las Vegas; Jor-Shan Choi of the Lawrence Livermore National Laboratory; and Pete McGrail of Pacific Northwest Laboratories. Responsibility for any errors or omissions in this discussion, however, is our own.

⁵⁷ This was confirmed in calculations by William Culbreth using the Keno code (version 4) and in calculations by the Hanford group using the MCNP code (Culbreth 1993, Omberg 1993). The results of the two codes were consistent. Because of the lower fission cross-section of U-235, the system would be even less critical if the plutonium had decayed to U-235 by the time the system had reached such a point.

⁵⁸ The minimum amount of plutonium that can reach criticality in an idealized system is about half a kilogram. The amount of plutonium's decay daughter, U-235, needed to reach criticality is substantially larger.

system: what is needed is to ensure that the ratio of solubility to inventory of the absorber is lower than that of the plutonium in the glass. There are several rare-earth oxides that are possible candidates. Gadolinium, for example, appears particularly promising, as it has a neutron-absorption cross-section much larger than that of boron, and a solubility believed to be comparable to that of uranium in the chemical environment expected in Yucca Mountain. Within the constraints of our study, we have not been able to explore the full range of such possibilities, or develop reliable information on the solubility of these potential additives, but we believe that this approach could be successful in alleviating long-term criticality concerns relating to plutonium in borosilicate glass. In combination with approaches limiting the amount of plutonium in each log (either by limiting the concentration or reducing the size of the logs), we feel confident that an acceptable solution to this potential problem could be developed within a few years at quite modest expense.

Criticality of plutonium dissolved in the water surrounding the waste packages would not appear to be an issue for the proposed Yucca Mountain repository. A typical value of the plutonium solubility used in analyses for unsaturated tuff is about 10^{-3} grams per cubic meter of water. This is far less than the single-parameter concentration limit for criticality of plutonium in water. Sorption of plutonium on the porous rock could increase the local concentration of plutonium in the rock-water mixture. However, even adopting a conservatively high sorption retardation coefficient⁵⁹ of 1,000 for plutonium, the effective concentration of plutonium would be well below the single-parameter concentration limit. Neutron absorption in the solid phase would further reduce multiplication. Another possibility is that plutonium might be precipitated by a local redox front. While this needs to be examined, we do not expect that this will be a serious problem—and if it is, the problem would also arise with plutonium that had leached away from ordinary spent fuel over very long times.

Spent Fuel Criticality. Potential criticality of normal LEU LWR spent fuel is being examined as part of the development of the Yucca Mountain repository, but planning is still in the early stages. As currently envisioned, as many as 20 spent fuel assemblies might be placed in a single large waste container. There is some concern that if water found its way into the repository, and ultimately into the container, and if the structure of the assemblies failed—so that the plutonium-bearing materials were in a water-moderated system concentrated at the bottom of the container—a criticality problem might arise. The Yucca Mountain project is examining a variety of approaches to address this concern, including means for filling the volume surrounding the assemblies (so that less water could enter the container, and failure of the assemblies would not result in con-

⁵⁹ The retardation coefficient is the ratio of the amount of plutonium in a unit volume of saturated porous rock to the amount of plutonium in the pore water in the rock.

centration of the material) and the addition of neutron-absorbing materials such as boron or gadolinium.⁶⁰

Our study is not the place, however, to examine issues concerning the spent fuels already scheduled for emplacement in the Yucca Mountain repository. On the assumption that some combination of the methods just described will be successful in demonstrating the absence of a criticality problem for normal LEU LWR spent fuel, we believe that an extension of the same methods should make it possible to resolve similar issues for MOX spent fuel. If necessary, a smaller number of MOX assemblies could be placed in each waste container, or the MOX assemblies could even be emplaced individually. We note that the quantity of plutonium in this fuel (as in the case of the glass), would be a small fraction of the total of more than 600 tons of plutonium in spent fuel already scheduled for emplacement in the Yucca Mountain repository.

Low-Level Wastes, TRU Wastes, and Tailings

In the case of the once-through MOX spent fuel option, the primary influence of plutonium disposition on the character of the LLW that would otherwise be produced by the corresponding amount of electricity generation would be the production of TRU wastes from conversion of plutonium metal to oxide and from MOX fuel fabrication, to which steps there is no counterpart in a once-through LEU fuel cycle. In the 1992–1993 Plutonium Disposition Study of the U.S. Department of Energy, the following estimates of TRU waste production in MOX fabrication were developed by vendors:

- ABB-Combustion Engineering (ABB-CE 1993): ABB-CE estimated that “total alpha contaminated wastes” would be 100 m³/yr for a 15-year campaign absorbing 50 tons of WPu, hence 30 m³ per ton of WPu. (How much of this total would be TRU waste by the 100 nCi/g definition is not clear from their report.) The ABB-CE study estimated that 0.5 percent of the MOX would be scrap, yielding 250 kg plutonium in MOX scrap for a campaign absorbing 50 tons of WPu. MOX scrap has a density of about 2 g/cm³, which if plutonium were 4 percent by weight of the MOX would imply 0.08 gPu/cm³, or 0.08 tons plutonium per m³, or 0.25 tons Pu / 0.08 tons Pu/m³ = 3.1 m³ of this particular form of TRU waste for the 50-ton campaign.
- General Electric (GE 1993): GE’s fuel fabrication complex was said to produce 60 m³/yr of LLW in connection with a MOX output of 58

⁶⁰ DOE also plans to use Yucca Mountain for disposal of a variety of spent fuels with greater enrichments than ordinary LWR spent fuel, which have been generated in defense and research programs. These spent fuels, which include highly enriched naval-reactor fuel and certain research-reactor fuels, will pose greater criticality difficulties and are likely to pose greater criticality difficulties and may require additional compensating measures.

MTHM/yr (sufficient to absorb 50 tons of WPu in 29 years), hence 35 m³ per ton of WPu. Of the 60 m³/yr, "a small fraction" was said to be TRU wastes. The small production of such wastes, compared to past experience with MOX fabrication, was attributed to minimizing the usual main sources of TRU waste production at MOX fuel fabrication plants, namely the chemical analysis lab and the scrap fuel area. Specifically, GE argued that: (1) modern nondestructive assay techniques will reduce use of chemical analyses and thus reduce the amount of TRU-contaminated chemicals; (2) since the prime objective is disposal of plutonium and not maximum fuel burnup or lifetime, nearly all pellets fabricated will be accepted for irradiation, minimizing scrap waste.

We have not found detailed estimates of the production of LLW or its TRU component from the conversion of plutonium metal to oxide. (In arriving at the above-cited estimates, the vendors assumed that the government would provide the plutonium to the fabrication plant in oxide form. See GE 1993, p. 3.6, and ABB-CE 1993, p. III-82.) As an upper limit, the volume of TRU wastes from metal-to-oxide conversion could hardly be greater than the volume associated with dealing with the same amount of plutonium in a fuel reprocessing plant, less the part of the reprocessing-plant TRU wastes that occur in the form of fuel-cladding hulls. Based on past reprocessing experience using the PUREX process, this volume was on the order of 50 m³ per 1,000-MWe reactor-year, hence per 250 kg of plutonium, thus 200 m³ per ton of plutonium (APS 1978, Holdren 1992). Non-TRU LLW from fuel reprocessing has amounted to another 50–100 m³ per 1,000-MWe reactor-year, or 200–400 m³ per ton of plutonium. We believe the actual totals are likely to be considerably smaller than these upper limits.

The amount of defense TRU wastes accumulated in the United States through 1990 was about 290,000 m³ and the total amount of defense LLW about 250,000,000 m³. Even if the above-cited vendor estimates of the production of LLW and TRU wastes from MOX fuel fabrication were to prove to be too optimistic by severalfold, and even if our upper-limit estimates of LLW and TRU wastes for plutonium metal-to-oxide conversion were correct, the contribution of WPu disposition by the MOX/spent-fuel route to the preexisting burden of defense LLW and TRU wastes would be tiny in relative terms. Depending on what fraction of the total LLW from MOX fuel fabrication actually turns out to be TRU wastes, and on the actual TRU-waste production from plutonium metal-to-oxide conversion, the corresponding TRU claim on repository volume could range from about the same to several times larger than that from the spent fuel produced by this disposition option, but in any case it would be a small fraction of the repository volume claimed by spent fuel from civilian nuclear-energy generation in total in the same period.

We have not seen or developed any detailed estimates of the production of LLW and TRU wastes for the vitrification option. Starting with plutonium oxide, the number and complexity of the processing steps needed to incorporate the PuO_2 into glass logs appear to be smaller than in the case of MOX fuel fabrication, which would suggest that the volume of LLW and TRU wastes should also be smaller. This tentative conclusion would need to be verified by closer investigation, but in any case it does not seem possible that the LLW and TRU wastes from this operation could exceed a percent or so of those already produced by the U.S. defense complex. It may well be, for the vitrification option as well as for the MOX/spent fuel option, that the largest quantities of LLW and TRU wastes will come from the plutonium metal-to-oxide conversion step that is common to both options.

With respect to uranium-mill tailings, use of the MOX/spent fuel option would obviate the need for the uranium mining and milling associated with 30–60 1,200-MWe reactor-years of electricity generation, hence would reduce by

$$(30\text{--}60 \text{ reactor-years}) \times (100,000\text{--}200,000 \text{ m}^3/\text{reactor-year}) = 3\text{--}12 \times 10^6 \text{ m}^3$$

the quantity of mill tailings that otherwise would be produced. This reduction would be an environmental benefit, albeit a modest one in relation to the quantities of tailings that already have been produced in connection with civilian and military nuclear-energy operations and that continue to be produced in connection with the civilian ones. If the vitrification option were chosen, there would be no effect on quantities of mill tailings.

Waste Implications of Other Reactor Approaches

If an advanced-reactor type—e.g., a liquid-metal or high-temperature gas reactor—were employed for WPu disposition, the characteristics of the wastes would differ in greater or lesser respects from those of the wastes of LWRs, depending on the reactor type and the fuel-cycle in which it was used. Differences could include the volume, physical form, chemical properties, and to some extent even isotopic composition of both high-level and low-level wastes. Some of these changes could be advantageous in reducing waste management burdens and the associated risks to workers and the public, while others could be disadvantageous. Given that the repository behavior of even the most familiar and much-studied types of spent fuel and HLW—for instance, those from existing commercial reactor types, reprocessing options, and vitrification processes—has not yet been fully characterized, it is only possible at this point to discuss in general terms the implications for waste management of other reactor and fuel-cycle approaches.

In general, it would be the case that approaches that actually destroy a high fraction of the WPu would be associated with some reduction in the criticality concerns connected with waste management, since the amount of plutonium

consigned to wastes would be relatively low; but because most of the high-destruction options would entail repeated reprocessing and MOX fuel fabrication steps, there would be a tendency to offset this criticality advantage with the production of greater quantities of TRU waste per ton of initial WPu treated than would be associated with the once-through MOX approach.⁶¹ On the other hand, some of the advanced-reactor/fuel-cycle combinations reviewed earlier in this report—notably those with online or integrated pyrometallurgical fuel reprocessing—have been designed to minimize the production of radioactive wastes associated with fuel reprocessing; as noted earlier, however, the technological practicality and economic feasibility of these approaches, not to mention the details of their waste-generating performance, remains to be proven. And, while it may be argued that greatly improved performance in waste generation—if it should turn out to be achievable—would be a great advantage in a reactor system intended for a major role in power production, the radioactive waste burdens associated with more conventional approaches to the disposition of 50 or 100 tons of WPu are not large enough to constitute an important incentive to develop advanced reactors and fuel cycles for the narrower and smaller-scale purpose of plutonium disposition.

Summary of Waste Issues

Spent fuel resulting from the use of MOX in LWRs and borosilicate glass containing WPu as well as defense HLW would be different enough from spent fuel derived from LEU and the borosilicate-glass/HLW combination, respectively, that separate licenses would be required to certify these new forms as acceptable for waste disposal; it would not be possible to rely on the licenses and associated reviews required for commercial LEU spent fuel and currently planned HLW glass. The licensing processes for the new waste forms would entail investigations of the mobilization of radioactivity and of criticality potential under conceivable repository conditions that would be much more extensive and thorough than the preliminary considerations undertaken so far and summarized above. If the outcome of these investigations were unsatisfactory, the option in question would not proceed.

Based on the considerations summarized here, however, we think that an irreparably unsatisfactory outcome is not likely. Most probably, if it proves possible to certify spent fuel from LEU and borosilicate glass with defense HLW for disposal in a geologic repository, it will also prove possible to certify spent fuel from MOX and borosilicate glass containing WPu as well as defense HLW. It could turn out, of course, that some modifications in repository design or waste form are desirable in order to accommodate the extra plutonium, but we

⁶¹ There would also be a trade-off within the category of criticality issues, since there are criticality hazards associated with reprocessing.

doubt that these would be of a sort to impose substantial additional delays on a repository program that is unlikely to be in operation before 2015 in any case.⁶²

As for LLW, including especially their TRU component, the disposition of WPu will certainly produce some additional TRU (compared to what would be produced in the absence of the disposition program), and under current regulations these will require disposal in a geologic repository. Conversion of plutonium metal to oxide, a common step for both the MOX/spent fuel and the vitrification options, might well be the dominant source of this increase, minimizing the importance of any difference in TRU generation between the two options in their subsequent steps. In any case, neither the absolute magnitude of the TRU wastes that could plausibly be generated by these two disposition options, nor any plausible differences between them, are likely to be large enough to constitute a significant impediment to choosing either one, or a basis for choosing between them. With respect to uranium-mill tailings, the MOX/spent fuel option has the advantage of reducing the tailings burden from what it would be if all nuclear electricity generation were based on LEU fuel, while the vitrification option would have no impact on tailings quantities; but the magnitude of this advantage of the MOX option is too small, given its tiny impact on the total tailings burden that will exist in any case, to have any significant influence on the choice between MOX and vitrification.

THE COMPARISONS IN SUMMARY

We summarize here the principal conclusions from the comparisons we have undertaken in this chapter. For brevity's sake, we will refer to the three main classes of disposition options we have considered as (1) "current-reactor" options (meaning the use of currently operating light-water or heavy-water reactor types, or evolutionary adaptations of them, to incorporate WPu into spent fuel, at typical commercial burnups, on a once-through basis), (2) "vitrification" (meaning incorporation of the WPu into glass logs containing high-level fission-product wastes), and (3) "advanced-reactor" options (meaning use of such reactor types as LMRs, MHTGRs, MSR, PBRs, and ABCs either to incorporate

⁶² It has been asserted in at least one analysis (Shaw 1992) that borosilicate glass is less complex and easier to analyze than spent fuel in terms of repository behavior, which could be taken to imply that the possibility of reaching an earlier conclusion about its suitability constitutes a significant advantage of choosing the glass route for plutonium disposition. We are not convinced that this distinction is either clear or significant—or if it were that it would remain so when the complication of additional plutonium content is imposed. Further, the intensive effort that has been mounted by DOE to design and evaluate the repository package for LEU spent fuel probably will give a licensing edge to the MOX/spent fuel option over the WPu/borosilicate-glass option. In any case, we think that attempts to characterize the repository behavior of both waste types in their plutonium disposition forms should proceed energetically and in parallel, as part of a process of intensive parallel investigations of both aimed at choosing the one that is best overall.

the WPu into spent fuel on a once-through basis or to destroy it to a greater extent by means of multiple recycles).

Security

Security considerations include both “direct” and “indirect” effects of a disposition option. The former are the ways an option influences the barriers against diversion or theft of the WPu. The latter are the ways it influences non-proliferation and arms reduction incentives and the protection afforded stocks of nuclear-explosive materials other than those dispositioned.

Timing is crucial to both of these dimensions of security. Minimizing the time until the start of operations to transform the surplus WPu into less accessible forms, and minimizing the time until this transformation is completed, are of obvious value in reducing the direct risks of diversion and theft. An expeditious approach brings major indirect security benefits, moreover, by signaling both commitment to irreversible arms reductions and seriousness in addressing proliferation hazards.

In terms of this crucial timing aspect of security, the current-reactor options and the vitrification option are roughly comparable to each other, and both are greatly superior to the advanced-reactor options. Under the most optimistic assumptions that are defensible, loading of WPu into current-reactor types could begin between 2002 and 2004 and be completed between 2015 and 2025; loading of WPu into waste-bearing glass logs could begin around 2005 and be completed as early as 2013. The timing uncertainties in both cases—relating more to resolution of institutional issues in the reactor case and to resolution of technical issues in the vitrification case—are bigger than the differences in the best-case point estimates we have provided; thus it would not be meaningful to say more than that the two sets of options are comparable.

Under the most optimistic assumptions that we consider defensible, any of the advanced-reactor options would be at least a full decade slower to get started; the delay could easily be longer even for the most well developed of these options (MHTGRs, LMRs), and it would probably be two decades or more for the least well developed of them (MSRs, PBRs, ABCs). We believe that the direct and indirect security risks of delays of this magnitude should be considered unacceptable, given that the current-reactor options and vitrification option provide the means to avoid these risks and given that the advanced-reactor options do not appear to offer advantages in other aspects of security, economics, or ES&H (as summarized in what follows) that could offset their timing liability.

The main factors besides timing that affect the comparative security of disposition options are (1) the extent of exposure to theft or diversion in the processing and transport steps that an option entails and (2) the theft and diversion risks posed by the plutonium in its final form and location. Among processing

and transport steps, those involving forms of material with low barriers to use in weapons and high portability are most problematic. Vitrification entails fewer such steps than the current-reactor options but, for the circumstances likely to apply in the United States, longer transport links. (As discussed in the preceding sections, there can be substantial differences—in these respects and in others—among variants *within* an option, depending for example on how many reactors at how many sites are employed, whether fuel fabrication facilities and reactors are co-located, and so on.) Advanced-reactor options do not appear to offer significant advantages, with respect to vulnerability of processing and transport steps, over the better variants among once-through current-reactor options, or over vitrification.⁶³

With respect to the security of the final plutonium forms that disposition options produce, we have concluded that meeting the “spent fuel standard” is both necessary and, for the decades immediately ahead, sufficient: if the WPu in its final form is not substantially more accessible for weapons purposes than the larger quantities of plutonium that will continue to exist in spent fuel from commercial electricity generation in this period, it will not represent a significant additional security hazard; but there is no great security advantage to be had from making the WPu much *less* accessible than the rest of the plutonium in commercial spent fuel, since the latter would then dominate the overall risk.

The current-reactor options obviously meet the spent fuel standard, and we judge that the vitrification option meets this standard also. The plutonium in the spent fuel assembly would be of lower isotopic quality for weapon purposes than the still weapons-grade plutonium in the glass log, but since nuclear weapons could be made even with the spent-fuel plutonium this difference is not decisive. Under middle-of-the-road assumptions,⁶⁴ the radiological barrier presented by glass logs would be about three times smaller than that presented by a fuel assembly (but still very high), and the mass of a glass log—containing, coincidentally, about the same amount of plutonium as a fuel assembly—would be about three times greater. The difficulty of separating the plutonium from the accompanying materials would be roughly comparable in the two cases.

Use of advanced reactors could produce reductions in the quantity of residual plutonium from the disposition process (even on a once-through basis in the

⁶³ If recycling of plutonium in order to burn up more of it were deemed important, some of the advanced-reactor options might offer security advantages by being able to recycle plutonium without separating it completely from fission products. In the once-through mode we favor to reduce the clear and present security danger of WPu, however, advanced reactors do not offer reductions in handling and processing of vulnerable plutonium forms.

⁶⁴ Here we compare PWR fuel assemblies initially containing 5.5 percent WPu (55 g WPu per kgHM, 461 kgHM per assembly, 658 kg MOX plus hardware per assembly), irradiated to 40 MWd/kgHM, with the large logs currently planned to be produced at the Savannah River site (1.3 percent WPu and 20 percent fission products in 1,700 kg of glass, 2,200 kg of glass plus canister) (see Table 6-5).

case of the MHTGR, and with the assistance of plutonium recycle in other cases) and in some cases (particularly the MHTGR) could produce decreases in the plutonium's isotopic quality for weapons purposes to below that characteristic of spent fuel. As just noted, however, these changes would not bring much reduction in overall security risk unless commercial spent fuel stocks were similarly transformed. Although society might eventually decide to do this and might choose advanced reactor types for the purpose, transforming today's very dangerous stocks of surplus WPu to meet the spent fuel standard does not require advanced reactors and should not wait for them.

Economics

Estimates of the economic consequences of alternative reactor-related disposition options depend strongly on the assumptions that are made about a wide array of factors: the real interest or discount rate; the time required for design, construction, and licensing; the treatment of interest during this preoperational period; the market value of any electricity produced; whether the facilities are owned by the government or the private sector; and so on. What should be assumed about these matters is not always obvious, and any analyst's choices can be criticized. If different disposition options are to be compared on an even-handed basis, however, the assumptions made about these factors must at least be consistent across the cases analyzed, and, similarly, the estimates of construction costs and operating costs that are at the core of any economic analysis must have been generated for the different options with a consistent degree of conservatism or optimism. We have sought, in our economic analysis, to impose a degree of consistency in these respects and to explain the associated assumptions in a way that will permit interested readers to determine the origins of any differences between our results and those of other analysts.

Among many ways to present the results of economic analyses, we have chosen as our primary figure of merit the discounted present value, as of the start of WPu disposition operations at a reactor (or melter, in the case of the vitrification option), of the stream of costs produced by the associated WPu disposition activities before and after this time, after subtracting revenues from associated electricity generation where appropriate. We express this net discounted present value at start of reactor (or melter) operation in 1992 dollars. Table 6-21 summarizes the results for all of the options for which such estimates were developed.

The most important conclusions that emerge from the cost calculations summarized in Table 6-21 are as follows:

- The central estimates of the net costs of the reactor-related options considered, expressed as net present value as of the start of reactor or melter operations, range from about \$0.5 to about \$6 billion (1992 dollars) for disposition of 50 tons of WPu. When the uncertainty ranges

TABLE 6-21 Net Economic Impact of Options for WPu Disposition (in millions 1992 dollars)

MOX Use in Currently Operating U.S. LWRs: ^a						
Fuel Source:	No Modifications Required			Modifications Required		
	FMEF	New Plant no Tax	New Plant with Tax	FMEF	New Plant no Tax	New Plant with Tax
	450±250	900±300	1,100±300	1,500±400	1,900±400	2,100±400
MOX Use in Currently Operating CANDU Reactors in Canada, Fuel from FMEF: 950 ^b						
Completion of Mothballed U.S. PWRs for MOX Use (fuel from FMEF): ^c						
	1 reactor, 6.8% Pu, 30 years:			1,300±1,600		
	2 reactors, 4.0% Pu, 25 years:			2,200±3,000		
Construction of New Evolutionary or Advanced Reactors for MOX Use: ^d						
		Reactor Pays no Ins, Prop Taxes, Fuel from FMEF ^e		Reactor Pays Ins, Prop Taxes, New Fuel Plant		
ELWR-1 (GE ABWR)		2,600±3,600		5,500±3,800		
ELWR-2 (ABB-CE System-80+ PWR)		1,600±1,800		3,200±1,900		
ALWR (Westinghouse PDR-600)		3,100±2,100		5,100±2,400		
MHTGR (GA)		3,900±2,700		5,800±3,200		
ALMR (GE)		3,900±2,500		5,600±3,000		
Vitrification with Defense HLW Savannah River Site: ^f 1,000±500						

NOTES: Figures are discounted present value of net costs of disposition of 50 tons of WPu by the indicated means, accounting for electricity revenues where relevant. Present values are calculated as of start of plutonium disposition operations at the reactor (or melter, in the vitrification case). For details of calculations and justifications of assumptions, see the economic evaluation section in Chapter 3 and "Economic Comparisons" in this chapter. The ± ranges are the panel's judgmental 70-percent confidence intervals.

^a Two 1,200-MWe PWRs dispositioning 50 tons of WPu in 21 years (rounded from Table 6-13).

^b Two 769-MWe CANDUs dispositioning 50 tons of WPu in 24 years (no confidence interval estimated) (see section "Weapons Plutonium Versus Uranium as Power Reactor Fuel" in this chapter).

^c One or two 1,256-MWe PWRs (see "Completing Existing LWRs" in this chapter).

^d For the MHTGR and ALMR, the figures in this column correspond not to FMEF but to a new fuel fabrication plant that does not pay property taxes and insurance.

^e See "Building New Reactors for Plutonium Disposition" in this chapter.

^f See "Economics of Vitrification" in this chapter.

on these estimates—here expressed as judgmental 70-percent confidence intervals—are included, the ranges extend from a negative \$1.2 billion (i.e., a profit in this amount) to a cost of \$9.0 billion.

- The lowest central estimate, at about \$0.5 billion, is for the MOX/spent fuel option using currently operating U.S. LWRs that need no modification to use MOX safely, with the fuel fabricated at the FMEF at the Hanford site. Four of the options studied have central-estimate costs around \$1 billion: use of MOX fuel from FMEF in currently operating CANDU reactors in Canada; use of MOX from FMEF in a single, currently mothballed, partly completed PWR that would be completed for this purpose; use of MOX from an entirely new fuel fabrication plant in currently operating U.S. LWRs that need no modification to use MOX safely; and vitrification with defense HLW at the Savannah River site.
- Although the central estimates in all cases considered correspond to net costs, our judgmental 70-percent confidence intervals include a possibility of profits from WPu disposition for the case in which currently mothballed, partly completed PWRs are completed for the purpose of plutonium disposition and these PWRs use MOX fuel from FMEF, and for cases when new, evolutionary LWRs are built for this purpose at government facilities (paying no property taxes or insurance) and use MOX from FMEF. These profit possibilities depend not only on the costs associated with MOX use falling at the low end of our judgmental 70-percent confidence ranges, but also on the additional electricity generated by the plutonium disposition reactors being marketable at a price of 5.5 cents per kilowatt-hour or higher (1992 dollars) at the busbar.
- Central estimates for the net costs of the use of advanced reactors for WPu disposition, at \$3–\$6 billion, are considerably higher than the central estimates of using currently operating or mothballed reactors for this purpose. The circumstances needed to realize costs at the low end of the uncertainty ranges for the advanced reactors include a high market value for the electricity they produce—a circumstance that would also lower the net costs of the other options.

It is important to note that all of these cost estimates represent sums of money that are modest in relation to the security benefits of plutonium disposition. For example, the best-estimate, net-cost figure for a typical current-reactor MOX/spent fuel option—say, \$1–\$2 billion (1992 dollars) net present value at the start of reactor operations in, say, 2002—could be paid for by setting aside less than 0.4–0.8 percent of the fiscal year 1995 U.S. defense budget (even without allowing for interest on this sum between 1995 and 2002). The range of \$0.5–\$5 billion (1992 dollars)—covering the best estimates of net present value, at reactor or melter startup, of most of the options considered—corresponds to

\$10,000 to \$100,000 per kilogram of WPu, or \$40,000 to \$600,000 for a nominal “bomb’s worth” of 4–6 kg. Even the higher figure is probably less than what this weapon material once cost to produce, as well as much less than would be spent in the attempt to recover such material if it went astray and incomparably less than would be spent to try to deter or otherwise prevent its use in the form of a bomb in the hands of a potential adversary.

Environment, Safety, and Health

The ES&H characteristics of the current-reactor options and the vitrification option for disposition of surplus are summarized in Table 6-22, based on the treatment of this topic in “Building New Reactors for Plutonium Disposition” above. Organized by types of activities common to the two classes of option, this table compares the options to each other with respect to the ES&H issues they raise in each activity, and then places these issues in the larger context of the ES&H impacts of analogous nuclear-weapon and nuclear-energy activities in which the United States has been and will continue to be engaged irrespective of a WPu disposition program.

The main conclusions that emerge from this table and from the more detailed analysis in the section “Building New Reactors for Plutonium Disposition” are that:

- most of the ES&H impacts of WPu disposition using either of these options can be expected to represent modest additions, at most, to the routine exposures to radiation and risks of accident associated with other civilian and military nuclear-energy activities underway in the United States;
- there is no apparent reason that the activities involved in WPu disposition using either of these approaches should not be able to comply with all applicable U.S. ES&H regulations and standards;
- while there are differences in detail in the ES&H challenges and risks posed by the two options in some of the activity categories—e.g., a somewhat more complicated set of plutonium-handling operations for the reactor options than for the vitrification option, and a greater relative increase in plutonium content of the final waste form for the vitrification option than for the reactor options—these differences do not consistently favor one class of options or the other, and none is large enough in relative or absolute terms to justify choosing one class of options over the other;
- the ES&H issues to which the greatest attention ought to be given in the next phase of study of these options are (1) ensuring adequate safety against criticality accidents in the melter for the vitrification option, (2) confirming the conditions under which full-MOX cores can be

TABLE 6-22 Summary of ES&H Issues for Current-Reactor and Vitrification Options

Activity	Current-Reactor Options vs. Vitrification Options	WPu Disposition vs. Other Nuclear-Energy and Nuclear-Weapon Activities
Pit storage	No difference	Not more demanding than intact-weapon storage already done on a larger scale
Transport of pits or plutonium metal	No difference	Not more demanding than intact-weapon transport already done on a larger scale
Plutonium metal-to-oxide conversion	No difference	No equivalent in current U.S. nuclear-weapon or nuclear-energy practice, but technology for doing it safely is well established
Storage of plutonium oxide or nitrate	No difference	Less demanding than storage of wide array of plutonium forms and plutonium-contaminated materials in nuclear-weapon and nuclear-energy complexes
MOX fuel fabrication	Vitrification option lacks an equivalent step	No equivalent in current U.S. practice, but technology well established and less demanding in ES&H terms than metal-to-oxide conversion
Transport of fresh MOX fuel	Step does not occur in vitrification option	Plutonium in fresh fuel adds some risk to current nuclear-energy practice, but risk is small compared to transport of pits and weapons
Irradiation in reactor	Effects of plutonium addition more complex than in vitrification counterpart, but well studied	Plutonium addition unlikely to significantly affect safety of the small subset of commercial reactors involved
Addition of plutonium to vitrification melt	See comparison above; main plutonium effect is on criticality potential, which needs more study	Except for criticality issue, plutonium addition adds little to ES&H risks from melter operation that would occur without WPu program
Transport of spent fuel or glass logs	Spent fuel more radioactive and probably more easily dispersed	Extra plutonium from WPu disposition adds only moderately to spent fuel or glass-log transport risks that would occur without WPu program
Geologic repository storage of spent fuel or glass logs	Relative increase in plutonium content from WPu disposition is greater for glass logs than for spent fuel	WPu adds little to population dose except in very unlikely intrusion scenarios; added criticality needs more study

used without adverse impacts on safety in reactors of currently operating commercial types, and (3) clarifying the conditions for avoiding long-term criticality in geologic repositories containing either spent fuel or glass logs from WPu disposition operations.

It is worth emphasizing here that the task of ES&H assessment is considerably simplified by the circumstance that both of the leading-candidate classes of options would simply add WPu to a set of nuclear activities that would be going on in any case. In addition, both would leave the residual WPu in a waste form—spent fuel in one case and HLW-bearing borosilicate glass logs in the other—which will exist in large quantities and will need to be safely managed whether used for WPu disposition or not, and the relevant properties of which will not, for the most part, be very much affected by the addition of WPu in the quantities foreseen.⁶⁵

Some of the advanced-reactor options could plausibly reduce some of the ES&H impacts associated with use of the current-reactor options, such as the magnitude of the uranium-mill-tailings burden and, perhaps, the quantities of low-level TRU wastes produced in MOX fuel fabrication. Such potential ES&H gains from the use of advanced-reactor options are far too modest, however, to justify the security liabilities of postponing WPu disposition until the advanced-reactor options are available.

⁶⁵ The main caveat relates to the potential for criticality in the repository in the very long term. The higher plutonium concentrations in spent fuel associated with using current-reactor/spent-fuel options for disposition, and the much higher plutonium concentrations in glass logs used for WPu disposition compared to those not so used, may add to the repository criticality problem. Nonetheless, the plutonium content in ordinary spent fuel is sufficient to necessitate very careful attention to the avoidance of repository criticality even in the absence of WPu disposition, and it is likely that the effort required to provide assurance against repository criticality for ordinary spent fuel will lead the way to measures that will provide this assurance for WPu-MOX spent fuel and for plutonium-bearing glass logs as well.

**APPENDIX:
APPROVAL AND LICENSING ISSUES IN WEAPONS
PLUTONIUM DISPOSITION**

Regulatory and licensing issues will be a critical pacing factor in accomplishing plutonium disposition in the United States (and, to differing extents, in other countries that might be involved). An administration decision to burn or bury the WPu will involve material that was very costly to produce, and is viewed by some as a potential asset because of its energy value. Such a decision will also raise important issues for nuclear waste disposal. DOE has already determined that a Programmatic Environmental Impact Statement (PEIS) is required. Congressional authorization and appropriation will also be necessary. The process of decision and approval is likely to be extended and controversial.

As noted in the text, the regulatory environment that now exists in the United States affords multiple opportunities for opponents of large nuclear projects to intercede with regulatory agencies, the courts, and Congress to slow or stop their implementation. While some progress has been made in streamlining the licensing process for such projects in recent years, the overall process of pursuing a large project to completion—including not only the licensing process but the fundamentally political process of gaining funding and associated approvals for such efforts—remains a difficult and uncertain one.

Any new activity at a nuclear-reactor site or major DOE nuclear site generates local public interest and, usually, opposition. Plans to process tens of tons of plutonium at a particular site, or to introduce MOX fuel into certain reactors, can be expected to produce such interest. Added to the reactor-related opposition is the strong anti-DOE feeling in many communities. Although the current DOE has made efforts to change these attitudes, any DOE initiative is likely to be scrutinized closely and to face public opposition. Any flaws in a required PEIS or EIS probably would serve as the basis for court challenges. The length of time required, the probability of success, and the cost involved in gaining licensing and regulatory approval for relevant plutonium disposition options are difficult to predict.

Reaching Agreement on an Option

The first and most fundamental steps in the process will be to bring the various differing interests within the administration and Congress to agreement on a specific choice of disposition options, and to gain sufficient funding from Congress. This will take time (the current hoped-for schedule is to reach a decision in the spring of 1996) and is likely to be contentious. Committed advocates

of particular options are likely to take whatever opportunities present themselves to attempt to reverse decisions that go against them.⁶⁶

The National Environmental Policy Act and the EIS Process

The first major regulatory step will be meeting the requirements of the National Environmental Policy Act (NEPA), which requires an Environmental Impact Statement (EIS) for all major federal actions likely to have a significant environmental impact. NEPA was intended to ensure that environmental issues would be appropriately considered in the process of reaching decisions on preferred options for tasks such as this one. An EIS must consider all reasonable alternatives for accomplishing the objective, and must in particular examine the "no action alternative." DOE is currently preparing a Programmatic Environmental Impact Statement (PEIS) analyzing the options for both plutonium storage and plutonium disposition (as well as storage and disposition of the other surplus fissile materials, such as HEU, U-233, and miscellaneous actinides). A draft is expected to be released for public comment in the fall of 1995, with the final document published in early 1996. Subsequent EISs for specific activities at particular sites may then be necessary.

There is an extensive history in recent years of court challenges to the adequacy of EISs. These challenges have sometimes been successful in delaying projects by several years—which, with changing circumstances over time, has sometimes led to cancellation of the project concerned. Whether the PEIS that DOE is now preparing, or subsequent site-specific EISs, would suffer such a fate is impossible to predict. It should be noted that the EIS is only the first step in this process, not the last: in addition to the EIS and gaining formal license approvals from federal regulatory agencies, for example, many state permits may be required unless Congressional action removes these requirements.

Are New Legislation or Generic Rules Needed?

A major question is whether plutonium disposition (particularly the use of plutonium fuels in U.S. reactors) would require new legislation.

It appears that new legislation is not essential, but that some form of congressional approval would be highly desirable. In 1983, Congress barred the use of commercial fuel for nuclear explosive purposes. This legislation was specifi-

⁶⁶ In the fiscal year 1993 and 1994 budgets, for example, advocates of particular reactor options inserted Congressional "earmarks" directing that specific amounts of the money allocated to plutonium disposition be spent on studying particular reactor types, though DOE resisted this earmarking. In the fiscal year 1995 budget, the earmarking is more general, specifying a substantial amount of funding that should be spent only on nuclear reactors, and calling specifically for a fast turn-around study of the so-called "triple-play" reactor options—reactors that would burn plutonium, produce electricity, and produce tritium for the weapons stockpile.

cally aimed at preventing plutonium from reprocessed civilian spent fuel from being used for nuclear weapons. There is no prohibition, however, on the use of material from weapons for commercial fuel. (Indeed, the HEU purchase agreement, in which LEU produced from HEU from Russian nuclear weapons will be purchased by the United States for commercial use, will have established this principle before plutonium disposition begins.) Nor is there any general legislation forbidding the use of plutonium fuels in U.S. commercial reactors. However, the overall process (including NEPA determinations) would be smoothed if there were explicit Congressional approval of the particular disposition options chosen. Given that Presidential Decision Directive 13 indicates that the United States does not encourage reprocessing of plutonium,⁶⁷ Congressional approval would be particularly desirable if reactor options were chosen, to emphasize the national interest in accomplishing the plutonium disposition mission. The panel was informed unequivocally by the CEOs of two nuclear utilities that clear congressional approval would be a requirement before commercial utilities would allow WPu-MOX to be used in their currently operating commercial reactors. (The continued conflict in Nevada concerning opening a repository at Yucca Mountain, however, demonstrates that it is not enough merely to have Congress pass a law stating that an action is in the national interest.)

Another question is whether, in the absence of congressional legislation, a generic rule-making by the Nuclear Regulatory Commission (NRC) would be necessary before plutonium could be used in U.S. reactors. In the mid-1970s, when industry and government predicted that reprocessing and recycle of plutonium in reactors would soon be necessary, the NRC began a lengthy regulatory process to address the associated environmental issues, known as the Generic Environmental Statement on Mixed Oxide (GESMO). The GESMO public hearings were halted by the NRC in 1977 in response to the Carter administration's opposition to reprocessing.⁶⁸ Since that time, case law has changed so that a hearing would not be required were such a rule-making to be undertaken today. Even more important, it probably would not be necessary to develop a generic rule at all if only a few reactors were involved and the reprocessing issue was not raised. Instead, license amendments for the few reactors involved would be required (see below).

⁶⁷ White House (1993). This statement has been widely interpreted as opposing use of plutonium in general, but appears to be specifically directed at use that involves additional reprocessing, which need not be the case here.

⁶⁸ Since the NRC is an independent agency, the administration could only request, not direct, the NRC to halt the GESMO hearings. The administration did make clear that there would be no federal funding requested for any aspect of the reprocessing regime. Because the potential reprocessing industry depended on federal funding, including development of breeder reactors, the administration's positions made the GESMO hearing irrelevant.

Federal and Nonfederal Reactors: Who Regulates?

Any plutonium disposition option is likely to involve construction, modification, and operation of important facilities that must be regulated—particularly fuel fabrication plants and reactors for the reactor options, or vitrification facilities for the disposal option. Technically, the licensing issues related to the reactors themselves are only a matter of degree, and do not appear to pose any major technical obstacles. But the procedural issues—how these various facilities would be regulated, and by whom—will be important.

Reactors owned by the U.S. Department of Energy (either government-owned, government-operated [GO-GO] or government-owned, contractor-operated [GO-CO]) would be a special case—one in which current practices are under discussion and review. Currently, the NRC does not regulate any federally owned nuclear plants used for purposes connected with the defense mission. The Defense Nuclear Facilities Safety Board (DNFSB) was established by Congress in 1988 to provide a form of regulatory oversight for DOE weapons facilities. As currently constituted, however, the DNFSB is a purely advisory body with no authority to issue binding regulations or orders. Congressional legislation has been introduced to provide external regulation of some aspects of DOE. This legislation likely will be reintroduced in the 1995 Congressional session, with possible modifications based on DOE recommendations following advice from an advisory committee established to review all aspects of external regulation of DOE.

Were DOE to build a nuclear plant for this mission, or were DOE to buy and complete or take over an existing commercial nuclear plant, DNFSB could be asked to provide advisory oversight. It is not completely clear that DNFSB would be the appropriate agency, however, because such a plant would be for the purpose of generating electricity (using MOX fuel which happens to be fabricated using weapons material), rather than part of the weapons complex. On the other hand, the NRC currently has no statutory authority to regulate DOE facilities, so further legislation would be required if it were to serve in anything more than an advisory role comparable to that of the DNFSB. Absent such legislation, a full-fledged regulatory regime would not exist for a DOE-owned reactor in either the DNFSB oversight or the NRC advisory case. Ultimately, it appears both likely and desirable that reactors burning WPu-MOX will have to meet the same NRC safety and licensing standards applied to commercial reactors.

There are major differences in the operation of the NRC and the DNFSB, including an order of magnitude difference in the size of their staffs. The DNFSB has no hearing process, does not operate under the Administrative Procedures Act, and has not developed the practice of involving the public in its deliberations (an issue on which it has been taken to court). Under current procedures, the length of time required to license a facility, or to get approval for

operation, would probably be shorter under the NRC than the DNFSB, if the review and hearings processes were similar. Certainly this would be the case if the DNFSB's actions were challenged in court.

If an existing licensed commercial reactor were taken over by DOE for plutonium disposition, the issue of transferring its license would be raised. Although licenses have been transferred from one utility to another, a license or a construction permit has never been transferred from a commercial facility to the government.⁶⁹ If a transfer to DOE were proposed, it is most likely that the NRC would require the utility to apply for a license termination. The NRC's role then would be to make a finding of the effect of such action on the safety of the public.

If a new reactor were to be constructed with DOE ownership, under current procedures the DNFSB would provide oversight. Overseeing construction and operation of such a reactor or reactors would pose considerable challenges for the DNFSB. The time required to build such a reactor, however, would give the DNFSB the time to develop gradually the expertise to oversee operation.

If the NRC were asked to provide advice concerning a DOE-owned reactor (as was done for the Fast Flux Test Facility, for example), there would be no formal NRC licensee, but there is the issue of who would pay for these NRC reviews. The NRC is required by law to recover all its costs from the applicants and licensees, but the NRC has no authority to charge DOE. DOE, however, could provide the funding for such reviews. Or, if DOE could characterize the reactor as a "demonstration" reactor, and it were operated on a commercial grid, the reactor would be licensed by the NRC, and as a licensee, would pay the costs of regulation.

The licensing procedure for a reactor that was privately owned would be more clear-cut. Nonfederal reactors operate under license from the NRC. The operating license review includes consideration of the core. Major changes in the core, such as introduction of MOX fuel, would be reviewed by the NRC for potential effect on the safety of the reactor. Although a claim might be made that a license amendment would not be required,⁷⁰ it would be prudent to assume that one would be necessary. The license amendment process would offer the opportunity for challenges, leading to public hearings. Under current law, the NRC could decide that no significant hazard would result from the amendment and allow the amendment to take effect, with the hearing held afterwards. This would be somewhat more likely for one-third MOX cores, which have been used overseas, than for full-MOX cores, but would still be a politically difficult decision for the Commissioners.

⁶⁹ The only exception is that ownership of uranium-mill tailing piles, after stabilization, is transferred to the federal government.

⁷⁰ Such a claim is made in GE (1994, p. 5.2-8).

To summarize, the two reactor options being considered can be assessed as follows:

1. Privately owned reactors would be regulated by the NRC, would require some type of environmental impact document, and, if currently holding licenses, would require license amendments. Barring a major technical safety problem (unforeseen at present), the licensing process should be completed well before the time required to build a MOX fabrication facility and obtain approval for its operation. Congressional endorsement would be highly desirable and may be required by utility owners of current reactors.
2. Government-owned reactors (either GO-CO or GO-GO) would be overseen by the DNFSB if current procedures were used, but the Secretary of Energy could ask the NRC to provide informal advice or the Secretary could ask Congress to amend the Atomic Energy Act to enable the NRC to regulate these reactors. Congress may act in 1995 or 1996 to provide external regulation of operations now managed by DOE. If the oversight process were handled by the DNFSB, its small size and unfamiliarity with regulation could make the process longer than if handled by the NRC. Nevertheless, the MOX fabrication facility still is likely to be the pacing item.⁷¹

Regulating Fuel Fabrication Facilities

For the reactor options, fuel fabrication facilities would also be required, and these too would require regulation. Procedurally, if the fuel fabrication facility were owned by the Department of Energy, or operated under a DOE contract with DOE oversight, the same considerations described above for the case of government-owned reactors would apply (even if it supplied fuel to a nonfederal utility). If the facility were privately owned and the only contractual obligation to the government were to provide fuel, the facility would have to be licensed by the NRC—even if all the fuel it produced were to be used by the federal government (as a precedent, the NRC licensed and regulated the privately owned facilities that made nuclear fuel for the U.S. Navy).

Technically, NRC licensing of a plutonium fuel fabrication facility might be somewhat more difficult than licensing reactors to use MOX. Given the absence of civilian reprocessing and MOX use in the United States, it has been many years since the NRC licensed a large plutonium bulk-handling facility. It may take some time to develop the necessary regulatory expertise. The DNFSB does

⁷¹ However, if use is to be made of a commercial reactor which had been closed rather than being modified to meet new NRC requirements, then DOE would come under great pressure from Congress to make such modifications and get NRC (informal) approval before operation. This could add several years and hundreds of millions of dollars to the process.

provide oversight of existing federal facilities handling plutonium in bulk forms, but large-scale plutonium processing in the DOE complex has been virtually shut down since before the DNFSB was established.

Regulating Vitrification Facilities

Vitrification would presumably be done in DOE-owned facilities (possibly operated by private contractors), as is currently planned for vitrification of U.S. HLW. Currently, the NRC does not have regulatory oversight of these facilities, which are reviewed by the DNFSB. (The NRC, however, does plan on monitoring vitrification facilities, to assure that the product will meet the NRC deep geological disposal regulations.) DNFSB oversight of plutonium vitrification in these facilities could raise issues similar to those described above in the case of possible DNFSB oversight of DOE reactors. Since the current waste already includes some plutonium, no fundamental regulatory changes would need to be made to review the inclusion of WPu. The principal technical issues would relate to criticality safety, the potential for plutonium releases, and worker exposures, all of which are issues DNFSB has addressed extensively in other areas. Presuming a vitrification facility was designed to provide adequate assurance of safety in these areas, gaining DNFSB approval for its operation should not pose special obstacles.

Repository Regulatory Impacts

Either spent MOX fuel or vitrified HLW glass would ultimately have to be disposed of in a geologic repository licensed by the NRC under regulations written to conform with other regulations written by the EPA. In both cases, the products are sufficiently different from the analogous products now scheduled for disposal—LEU spent fuel and vitrified HLW without significant quantities of plutonium—that it is likely they will have to be independently certified as acceptable waste forms for disposal.

Although EPA has issued final regulations for disposal of wastes in the Waste Isolation Pilot Plant (WIPP), there are as yet no final regulations which a Yucca Mountain repository must meet. A National Research Council committee will recommend a regulatory approach for a Yucca Mountain repository, which EPA must consider. EPA and NRC regulations will follow. Neither the addition of WPu into the vitrification process, nor the use of MOX fuel with its generally higher plutonium content, has been examined for its impact on the repository license. The criticality issues addressed elsewhere in this report appear to be the most important technical questions that would be of concern.

Licensing and Russian Plutonium Disposition

If the future regulatory process in the United States is somewhat uncertain, that in Russia is even more so. In the wake of the collapse of the Soviet Union, the recently established Russian nuclear regulatory agency, GOSATOMNADZOR (GAN) is still finding its role. At this writing (early 1995), a new Atomic Law is still being debated in the Russian parliament. It is inevitable that the regulatory environment in Russia will evolve substantially between now and when plutonium disposition is actually underway on a large scale, in ways that are difficult to predict. At the moment, GAN is much weaker politically than the Ministry of Atomic Energy, which it is intended to regulate. As a result it appears unlikely that GAN objections to particular plutonium disposition facilities, even were they to arise, would have a major impact on the timing or cost of plutonium disposition in Russia.

Licensing and Plutonium Disposition in Other Countries

If Canadian CANDU reactors were used for plutonium disposition, these would be regulated by Canada's Atomic Energy Control Board. As noted in Chapter 4, the regulatory environment in Canada tends to be less adversarial and involve more cooperation between regulators and utilities than is the case in the United States.⁷² As in the U.S. LWR case, reactor licensing would probably not be a major source of delay in the case of the CANDU option. Just as in the United States, however, the broader issue of overcoming all the potential political and regulatory barriers would be a difficult one in Canada if there were significant public opposition to MOX use in Canadian reactors.

If U.S. or Russian WPU were used in Europe or Japan, the relevant regulatory agencies and publics would be substantially more familiar with civilian use of plutonium. While the use of weapons-grade plutonium does involve somewhat different issues, there would not appear to be any major problems in the licensing process itself. The more difficult problems would arise from the political issues involved in shipping large quantities of WPU from the United States or Russia to these countries.

⁷² Ahearne (1988).

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Conclusions and Recommendations

In this study we have analyzed the possibilities for (1) using nuclear reactors to process surplus weapons plutonium (WPU) into forms resistant to diversion or theft for reuse in nuclear weapons, (2) using reactors to consume the plutonium altogether, and (3) rendering the plutonium resistant to diversion or theft by immobilizing it in waste forms similar to those contemplated for disposal of high-level wastes originating in nuclear reactors. This work has been motivated by the direct and indirect security dangers that would be posed by long continuation of the status quo in management of the increasing stocks of surplus plutonium from dismantled nuclear weapons, which consists of guarded and monitored storage of plutonium pits (the nuclear-explosive cores of the weapons) at various sites in Russia and at the Pantex site (near Amarillo, Texas) in the United States. The direct dangers of the prolonged storage of the pits are the risks of their diversion (by the possessor states) or theft (by or for other states and subnational groups) for use in nuclear weapons; the indirect dangers are the adverse effects on incentives for further arms control, and for nonproliferation, that will arise if the United States and Russia fail to put additional barriers in the way of reuse of this plutonium in their own arsenals.

In considering the possibilities for using nuclear reactors or immobilization technologies to provide such barriers, we have tried to address, on a comparative basis, technological readiness, institutional requirements, economics, and environment, health, and safety characteristics. We have given by far the greatest weight, however, to the security characteristics of the various possibilities—their capacity to reduce, in a timely way, the direct and indirect security dangers

posed by prolonged storage of the plutonium as pits, while minimizing the new security problems arising from the disposition operations themselves. This chapter summarizes our main conclusions and recommendations. (This is also done more briefly in the Executive Summary at the beginning of the report.)

DISPOSITION OPTIONS AND END-POINTS

Options entailing the incorporation of surplus plutonium into fuel that is then irradiated in nuclear reactors (for brevity, “reactor options”) can be subdivided by the type of reactor to be used and by the condition of the plutonium at the conclusion of the disposition operation (end-point). Under *reactor types*, the categories of possible interest are:

- (a) electric-power reactors of currently operating commercial types, using fuels of current designs, or evolutionary adaptations of these reactors and fuels (for brevity, “current-reactor types”);
- (b) electric-power reactors of more advanced varieties (“advanced-reactor types”);
- (c) current or future naval propulsion reactors;
- (d) current or future research reactors; and
- (e) new reactors and/or fuels designed specifically for plutonium disposition, or for plutonium disposition in combination with tritium production.

With respect to the *end-points* of reactor options for plutonium disposition, there is a continuous spectrum of possibilities in terms of residual plutonium quantity, isotopic composition, and quantity of accompanying fission products, but it is useful to distinguish three general classes of outcomes with somewhat different purposes, as follows:

- (i) The “spent fuel” outcome is the result of a once-through fuel cycle in which a moderate fraction of the WPu is destroyed and the remainder is embedded in spent fuel that is similar—in bulk, radioactivity, and isotopic composition of the contained plutonium—to the spent fuel that already exists, in considerably larger quantities, from civilian nuclear-power generation. Net destruction of plutonium in this option, taking into account all plutonium isotopes, can range from slightly negative (in that more plutonium is produced from uranium-238 than is consumed from the initial stock of WPu) to as high as 50 percent in advanced light-water reactors and 80 percent in gas-cooled reactors. The purpose of this approach is to create substantial physical, chemical, and radiological barriers to use of the WPu in nuclear explosives by the original owner of the material or by others. This way of doing so would reduce the WPu management problem, at the end point, to a modest

part of the reactor spent fuel management task that will exist in any case.

- (ii) The “spiking” outcome results when briefer irradiation of WPu in a nuclear reactor processes the plutonium more rapidly (but with correspondingly less destruction of the plutonium and smaller changes in the isotopic composition of the remainder) into an irradiated fuel form with enough radioactivity to provide a moderate degree of protection for a few years after discharge. Net destruction of plutonium in this option is small. The purpose of this approach is to achieve a degree of physical, chemical, and radiological protection for the entire stock of surplus WPu more quickly, or with fewer reactors, than in the spent fuel option. It achieves this aim at the cost that the resulting degree of protection is not as high as in the spent fuel approach.
- (iii) The “elimination” outcome results from the use of fission and transmutation through multiple recycles of plutonium-bearing fuel to convert a very high fraction of the initial surplus WPu into other elements. The purpose of this approach is to eliminate as completely as practicable the possibility of reuse of the WPu in nuclear explosives.

The options that entail immobilizing the WPu in waste forms similar to those contemplated for disposal of fission-product wastes (for brevity, “immobilization options”) are somewhat less diverse than the reactor options, but they can vary in the type of waste form, in its dimensions, in the content of fission products, and in weight percent of contained plutonium. A particularly important distinction is between (1) immobilization options that add the plutonium to a waste form heavily laden with fission products, as planned in connection with the stabilization of U.S. defense high-level wastes in borosilicate glass, and (2) immobilization options in which the plutonium is incorporated into a waste form without fission products. The first class of options would serve a similar purpose to that of spent fuel reactor options, that is, to create substantial physical, chemical, and radiological barriers to further use of the WPu in nuclear explosives. (This way of doing so, unlike the reactor spent fuel approach, would not change the isotopic characteristics of the plutonium.) Vitrification options without fission products would offer quicker, easier processing at the cost of lower barriers to reuse in weapons; in that sense these options are analogous to the “spiking” variants of the reactor options.

NARROWING THE RANGE OF OPTIONS

We have concluded that an appropriate goal for WPu disposition operations to be undertaken over the next few decades is to convert the surplus WPu into forms approximately as resistant to diversion or theft for reuse in weapons as is the plutonium in spent fuel from commercial nuclear reactors. Achieving less

resistance than this would mean that the WPu remained a unique security hazard. Achieving more resistance than this would not bring much gain in security until a comparable degree of increased resistance was provided to the commercial spent fuel as well. That is so because the plutonium in commercial spent fuel is, despite some disadvantages of its isotopic composition for nuclear-weapon purposes, nonetheless usable for crude but quite powerful nuclear weapons by unsophisticated bomb-makers and for still more powerful weapons by sophisticated ones. If WPu were harder to obtain than plutonium from commercial spent fuel, then the latter—which exists in considerably larger quantities—would become the dominant security risk.

Acceptance of this “spent fuel standard” for disposition of surplus WPu over the next few decades rules out of consideration, for this purpose, the use of reactor options in the “spiking” mode and the use of immobilization variants that embed the plutonium into waste forms lacking fission products. Because of the low radiological barrier associated with the final plutonium form in these cases, they would not meet the spent fuel standard. We can imagine circumstances in which the “spiking” mode or a no-fission-product immobilization variant might be worth considering as a preliminary step prior to further irradiation in reactors or to later reimmobilization with fission products, in order to gain a modicum of protection quickly if carrying out the campaign to the spent fuel standard was expected to take a very long time; but neither of these approaches is acceptable as a stand-alone approach.

Even as preliminary steps, these low-radiation-barrier approaches have significant liabilities. For example, using reactors in the “spiking” mode increases MOX fuel fabrication capacity requirements—which are likely to be a limiting factor in any case—roughly in proportion to the speed-up in plutonium processing compared to the “spent fuel” mode. In the case of most reactor types, moreover, use of the spiking approach also significantly reduces reactor capacity factors (hence electrical output) because of increased downtime for refueling. (This is not the case for CANDU reactors, which can be refueled while they are operating.) In the case of immobilization without fission products as a preliminary step, it can be questioned whether the temporary barriers added in this way, compared to storage as pits, are sufficient to offset the security risks and economic costs of the extra handling and processing steps involved.

Acceptance of the “spent fuel standard” also effectively removes the “elimination” options from consideration as the primary disposition approach for the decades immediately ahead, although these options deserve continued study for their possible role in reducing the security hazards of all plutonium—military and civilian—in the longer term. The main reasons for this conclusion are (1) the elimination options are less developed technically and more demanding institutionally than many “spent fuel” options, and hence could not be initiated nearly as quickly; and (2) the elimination options, once started, would require a much longer operating time to achieve any reasonable elimination

standard for the surplus WPu, compared to the operating time required by spent fuel options to achieve the spent fuel standard for this material. Choosing an elimination option as the main approach to disposition, then, would be a prescription for great delay in both starting and completing the disposition campaign. We judge the direct and indirect security risks of such delay to be unacceptable. It makes far more sense to use one of the current-reactor/spent-fuel options or immobilization with fission products to bring the surplus WPu relatively quickly to a level of protection comparable to that of plutonium in commercial spent fuel and *then* consider, in the light of evolving technological capabilities and evolving conceptions of the nuclear-energy future, how the residual security risks of all of the plutonium at the spent fuel standard, military and civilian alike, might be subsequently reduced.

CURRENT-REACTOR OPTIONS FOR MEETING THE SPENT FUEL STANDARD

Commercial reactor types currently operating in the United States offer the technical possibility of transforming U.S. WPu into spent fuel within a few decades:

- (a) In mid-1994 the United States had 109 operating light-water reactors (LWRs) totaling 99,500 electrical megawatts (MWe) of generating capacity. Most if not all of these reactors would be capable, without significant modification, of operating with at least one-third mixed-oxide (MOX) fuel. (The remainder of the core would contain ordinary low-enriched uranium [LEU] fuel.) Assuming an initial weight fraction of 4-percent plutonium in heavy metal (uranium plus plutonium) in MOX fuel, just over 7 percent of the U.S. LWR capacity—for example, six 1,200-MWe reactors—would suffice to process 50 tons of WPu into spent fuel in 25 years of operation, assuming one-third MOX cores.
- (b) Three of the operating U.S. LWRs—the 1,221-MWe pressurized-water reactors (PWRs) at Palo Verde, Arizona—and one 1,240-MWe PWR that is 75-percent complete in Satsop, Washington, were designed to use 100-percent MOX cores. U.S. reactor manufacturers have indicated that a number of the other operating reactors could also use 100-percent MOX cores without major modification. Assuming favorable safety review of this capability, two 1,200-MWe-class PWRs using 100-percent MOX cores with 4-percent plutonium in heavy metal could process 50 tons of WPu into spent fuel in 25 years of operation.
- (c) If the use of 100-percent MOX cores were desired but the existing U.S. reactors suitable for this turned out to be unavailable for the WPu disposition campaign, modifications to one or more of the other operational or under-construction U.S. LWRs would make it possible, at tol-

erable cost, to use these in the 100-percent MOX mode (again assuming favorable safety review). The modifications would entail addition of more control absorbers and corresponding changes to the hardware at the top of the core, and if applied to reactors already operating would require a substantial shutdown period to complete.

- (d) If a plutonium loading of 6.8-percent plutonium in heavy metal in a 100-percent MOX core passed safety review, a single 1,250-MWe-class PWR could process 50 tons of WPu into spent fuel in 30 years.

The limiting ingredients on the timing of the current-reactor/spent-fuel approach in the United States would be providing the needed MOX fuel fabrication capacity (no such capacity is currently operational in the United States) and obtaining the necessary approvals and licenses (use of MOX fuel in U.S. power reactors is not now licensed).

- (a) The most expeditious solution to the MOX fabrication problem appears to be to bring to operability the partly completed government MOX fabrication facilities at the Fuel and Materials Examination Facility (FMEF) at Hanford, Washington. The panel estimates that the FMEF could be made operational as soon as 2001 at a level of 50 tons heavy metal per year (MTHM/yr) of MOX fuel fabrication for an investment of $\$180 \pm \40 million (1992 dollars), including licensing and interest during construction. Given this start date, the last of the nominal 50 tons of WPu would be transformed into MOX fuel in 2025 and loaded into reactors in 2026. The sum of all incremental costs for disposition of 50 tons of WPu by the current-reactor/spent-fuel option—based on MOX fabrication at 50 MTHM/yr at FMEF and use of 100-percent MOX cores in two PWRs that needed no modifications for this role—are estimated at $\$450 \pm \250 million; if the reactors needed modification, the incremental costs would be $\$1,500 \pm \400 million. (These figures are the present values, as of the start of plutonium disposition operations at the reactors, of the net cost streams—i.e., total costs of electricity generation using WPu-MOX, minus the costs of generating the same electricity in the same reactors using LEU fuel—measured in 1992 dollars and based on a real cost of money of 7 percent per year.)
- (b) If the MOX fuel for the same reactors, under the same assumptions about fuel loading and burnup, were fabricated at a newly constructed MOX plant with capacity sufficient for processing 50 tons of WPu in 25 years, transformation of the WPu into MOX could begin as soon as 2003, the last WPu would then be transformed into fuel in 2027, and the last batch of WPu-MOX would be loaded into reactors in 2028. If the reactors used did not require modification for this role the incremental costs (net present value at start of reactor operation, 1992 dollars) would be $\$900 \pm \300 million if the new fuel fabrication plant did

not pay property taxes and insurance and $\$1,100 \pm \300 million if it did; if the reactors required significant modification, these costs would become $\$1,900 \pm \400 million and $\$2,100 \pm \400 million, respectively.

- (c) The problem of approvals and licenses for the use of MOX reactor fuel might become easier if the U.S. government chose to purchase and convert to federal facilities, for the purpose of WPU disposition, commercial reactor installations that were either never completed or whose continued operation is becoming economically less attractive for their utility operators. One such option would be the purchase of the mothballed, 75-percent complete Washington Public Power Supply System WNP-3 reactor, which is of the type and size capable of processing all 50 tons of surplus U.S. WPU in 30 years, and/or the purchase of the 63-percent complete WNP-1 reactor, located at the Hanford site, which is of the same size and could be modified during the completion work, if necessary, to permit its operation with 100-percent MOX. Purchase and use of one or both of these reactors in conjunction with MOX fuel fabrication at the Hanford FMEF would have the advantage of confining the handling of unirradiated MOX fuel to a single federal site (if WNP-1 were purchased) or to two federal sites in the same state (if WNP-3 or both reactors were purchased). WPPSS has decided to cease maintaining these reactors and to sell the parts, so timely government action would be required to keep this option open. Given a timely decision to proceed with this option, it should be possible for the WNP reactor or reactors to begin loading MOX fuel in 2002, consistent with the schedule assumed for fuel fabrication at the FMEF. We have estimated the incremental costs of this option (net present value at start of reactor operation, 1992 dollars) at $\$1,300 \pm \$1,600$ million if one of the reactors is used at 6.8 percent plutonium in heavy metal to load the 50 tons of WPU in 30 years (last WPU-MOX loaded in 2031) and $\$2,200 \pm \$3,000$ million if both reactors are used at 4.0 percent plutonium in heavy metal to load the 50 tons of WPU in 25 years (last WPU-MOX loaded in 2026).¹

If, for some reason, no combination of currently operating and partly completed U.S. LWRs was deemed attractive for the WPU disposition mission, it

¹ If, in the two-reactor case, the reactors are operated for another 5 years on LEU after completing 25 years of WPU-MOX operation, the maximum profit (in our 70-percent confidence range) increases to \$0.8 billion and the maximum loss shrinks to \$5 billion. As discussed in detail in Chapter 6, the cost uncertainties in the case of reactors that would not otherwise operate are very large, because they must take into account sales of electricity that would not otherwise be produced, and future electricity prices are uncertain enough that at the high end, such reactors might show a net profit, while at the low end, the net cost could run to billions of dollars.

would be possible at the cost of a few years' delay to construct a new dual-purpose (plutonium disposition/electricity generation) reactor or reactors on a government site. The logical choice of reactor type for this function, given adoption of the spent fuel standard and given the desirability of minimizing the delay, would be an evolutionary LWR. Given a timely decision to proceed, fuel loading in such a reactor or reactors could begin as soon as 2005. As examples of this approach, we evaluated cases involving two reactor types: use of a 1,256-MWe ABB-Combustion Engineering System-80+ PWR using 100-percent MOX with 6.8 percent plutonium in heavy metal at an average burnup of 42.2 megawatt-days per kilogram of heavy metal (MWd/kgHM) (loading the 50 tons of WPU in 30 years) and use of two 1,300-MWe General Electric (GE) advanced boiling-water reactors (ABWRs) using 100-percent MOX with 3.0 percent plutonium in heavy metal at an average burnup of 37.1 MWd/kgHM (loading the 50 tons of WPU in 29 years). We estimated costs of the ABB-CE PWR option to range from \$1,600 ± \$1,800 million if the fuel is fabricated at FMEF and the facilities do not pay property taxes and insurance to \$3,200 ± \$1,900 million if the fuel is fabricated in a new plant and the reactor and reprocessing plant do pay property taxes and insurance (present values in 1992 dollars as of start of reactor operation, net of electricity sales); the corresponding cost estimates for the GE ABWR option are \$2,600 ± \$3,600 million and \$5,500 ± \$3,800 million for the cases with and without property taxes and insurance.²

Heavy-water-moderated reactors in commercial operation in Canada (known as CANDU reactors, where CANDU stands for Canadian deuterium-uranium) appear to be compatible, without physical modification, with the use of 100-percent MOX fuel. (Favorable regulatory review of the safety of such operation would of course be required.) Two typical currently operating CANDUs of 769 MWe each could transform 50 tons of WPU into spent fuel somewhat less radioactive than that from U.S. LWRs in about 24 years of operation. Canada has 20 CANDU reactors totaling about 14 GWe (46 gigawatt-thermal; GWt). As with U.S. LWRs, the pacing elements of a plutonium disposition scheme based on existing CANDU reactors would be provision of the needed MOX fuel fabrication capacity and obtaining the needed permissions and licenses for burning such fuel. Canada has no MOX fuel fabrication capacity; fabricating MOX fuel for CANDUs at the FMEF is technically feasible and would be the most expeditious approach. The fabrication capacity needed for a 24-year campaign in two CANDUs using fuel with 1.2 percent plutonium in heavy metal is 170 MTHM/yr, which is within the capability envisioned for FMEF for this fuel type. Given a timely decision to proceed, the FMEF would

² Note that possibilities for negative net costs (i.e., profits) occur only in the cases where no property taxes or insurance are paid and then, as examination of Table 6-16 reveals, only in cases where the value of electricity at the busbar is above 5.5 cents per kilowatt-hour (1992 dollars). The profits made in these cases would be less than those that would be earned by operating the same reactors on LEU (again, see Table 6-16).

be able to begin operation producing CANDU fuel as soon as 2001, and in the two-reactor scenario just described the last of the 50 tons of U.S. WPU would be loaded into the reactors in 2025. The panel estimates the incremental costs of this CANDU disposition option at about \$1 billion. Because of their continuous refueling capabilities, CANDUs also offer the possibility of very rapidly “spiking” the entire WPU inventory and then reirradiating the fuel on a longer time scale to bring it to typical spent fuel burnup levels, all without adverse impact on electrical output. This approach would require MOX fuel fabrication capacity substantially larger than could be readily provided at the FMEF facility, however.

Currently operating commercial reactors in Russia and Ukraine also have the technical capacity to implement the “spent fuel” option in a timely way for a nominal 50 tons of surplus WPU from the stockpile of the former Soviet Union. For safety reasons, if this option is selected the only currently operating Soviet-designed reactors that should be used are the 3,130-MWt VVER-1000 LWRs, which are similar to Western PWR designs. Russia has six such reactors in operation (of which, according to one official of the Russian Ministry of Atomic Energy, only four are suitable for MOX fuel) and Ukraine has nine. Depending on the results of a safety analysis of high plutonium loadings in these reactors, and on the acceptability of high plutonium content in the spent fuel, it might be necessary to bring into operation some of the additional VVER-1000 reactors currently standing unfinished in Russia if the goal is to process 50 tons of WPU in 30 years of reactor operations within that country alone.³ The potentially limiting factors governing the timing of plutonium disposition in VVER-1000 reactors thus include this possible need to complete additional plants, as well as the successful upgrading of the safety features of these reactors and the provision of MOX fuel fabrication capacity. No such capacity is currently operating in the former Soviet Union, although a facility with an intended capacity of about 100 MTHM/yr—enough to feed 12 VVER-1000s using one-third MOX cores—stands unfinished at the Chelyabinsk-65 site. The timing of disposition of 50 tons of Russian plutonium in VVER-1000s could, in principle, be similar to that envisioned above for the disposition of 50 tons of U.S. plutonium in LWRs of currently operating commercial types.

The “spent fuel” option could also be implemented for U.S. and former Soviet Union WPU in currently operating Japanese and European LWRs that already use or are licensed to use one-third MOX fuel. (As of 1993, eight LWRs in France, seven in Germany, and two in Switzerland are using MOX fuel, and more are licensed to do so; Belgium and Japan plan to begin loading MOX fuel

³ Based on one-third MOX fuel and average burnup of 41,000 MWd/MTHM, the number of VVER-1000 reactor-years to process 50 tons of WPU is 270 (nine reactors operating for 30 years) if the fuel plutonium content is 2.4 percent and 135 (four reactors operating for 30 years and another operating for 15) if the fuel plutonium content is 4.8 percent.

in commercial reactors later in the decade.) This approach would be of (at best) limited value if the input of WPu to these reactors merely displaced the use of separated civilian plutonium, which is usable for bomb-making and probably not as well guarded as WPu. If WPu is to be fabricated into MOX in the facilities producing MOX for these reactors, its use should be phased in with, rather than replacing, the consumption of already separated civilian plutonium; either WPu should be fabricated using MOX capacity that would otherwise be idle (such as the Hanau plant in Germany, which might be opened specifically for the purpose of plutonium disposition), or existing reprocessing contracts should be renegotiated to delay separation of additional plutonium until existing stocks of civilian plutonium and WPu are consumed.⁴ This approach would also require international agreements and safeguards for the considerable international shipment of WPu that would be entailed. The timing of plutonium disposition in this mode could be similar to that envisioned above for the use of U.S. LWRs of currently operating commercial types.

Experimental and prototype liquid-metal fast reactors exist in a few countries and offer some near-term capacity for plutonium disposition with the "spent fuel" option—without reprocessing—although most are not in operation at the present time. The largest such capacity is the French Superphenix reactor (3,000 MWt), which in principle could process 50 tons of WPu into spent fuel in about 20 years. MOX fuel fabrication capability sufficient to support this operation exists in France and Belgium. In practice, however, operating Superphenix in this plutonium-burning mode would require modifications that could take many years to design and implement; and controversy can be expected about whether it should be restarted at all. Reaching agreement on the desirability and terms of the associated international transfer of the plutonium could also prove difficult. Russia and Kazakhstan have two liquid-metal fast reactors totaling about two-thirds the capacity of Superphenix, but they are too old to complete the plutonium disposition mission in their expected lifetimes, and questions have been raised about their safety.

Use of reactors on U.S. naval vessels for disposition of WPu is not practical on the time scales of interest. Such reactors now use high-enriched uranium fuel that is replaced only at very long intervals, if at all. To switch naval reactors to plutonium would first require a long (probably 10–20 year) program to develop and certify plutonium fuels for such use. Because of the extremely high reliability requirements for naval reactors and the uncertainties introduced by such a change, the navy would oppose it. The current rate of loading new fuel into naval reactors is essentially zero, moreover, meaning that the capacity for WPu

⁴ Under a scheme put forward by the Natural Resources Defense Council, for example, money from MOX-burning electric utilities that would have been paid, under existing reprocessing contracts, for reprocessed civilian plutonium in MOX fuel would instead be divided between paying a fair rate of return to the investors in commercial reprocessing plants that would not be operated and paying for WPu that would be processed into MOX.

disposition in this way is negligible at present; and there are no prospects for a net increase in the number of nuclear-powered naval vessels.

Existing research reactors, similarly, do not offer an attractive option for the disposition of WPu. These reactors are generally small in capacity and in duty factor, they refuel only rarely or not at all, and they are highly dispersed geographically and often located in institutional settings that would be difficult to safeguard. Given the availability of many more attractive possibilities, then, research reactors do not deserve serious consideration for the disposition mission.

ADVANCED REACTORS AND SPECIALTY FUELS

If the “spent fuel standard” is adopted, there is no need to develop and deploy an advanced-reactor type or nonfertile fuel type to achieve that aim. As indicated above, the numbers and characteristics of existing reactor types, using ordinary MOX fuels, are more than adequate to carry out the spent fuel option, and the limitation on the reactor-spiking option is fuel fabrication capacity, not inadequacies in the numbers or characteristics of reactors or the characteristics of fuels. Advanced reactors and nonfertile fuels do not offer sufficient advantages over existing reactor and fuel types, for achieving the spent fuel end-point, to offset the liabilities of longer lag times and additional costs before loading of WPu into reactors could begin. Advanced-reactor types and nonfertile fuels thus are of interest for the WPu mission only if the “elimination” end-point is to be sought in the longer term. (As noted above, such an approach would bring large security gains only if it were applied to reactor plutonium as well as to WPu.)

To illustrate more specifically the timing and cost penalties associated with the use of advanced reactors for the spent fuel mission, we constructed scenarios and cost estimates for using, in a once-through mode, the three advanced-reactor types that would be the least difficult to bring to the point of operation if a decision were made to do so:

- (a) As an example of an advanced light-water reactor (ALWR), we took the Westinghouse PDR-600, of which two 610-MWe units using 100-percent MOX cores with 5.5 percent plutonium in heavy metal and average burnup of 40 MWd/MTHM could load the nominal 50 tons of WPu in 34 years.⁵ The panel estimates that loading of WPu-MOX fuel

⁵ The plutonium net destruction fraction in this option is about 29 percent of the plutonium loaded, or about 460 kg of plutonium destroyed per gigawatt-electric-year (GWe-yr) of electricity generation. (This compares to about 280 kg of net *production* of plutonium per GWe-yr of electricity generation in LEU-fueled LWRs.) Use of an advanced MOX fuel in the PDR-600, which would permit burning 4 percent plutonium in heavy metal fuel to 50 MWd/kgHM, could achieve a net plutonium destruction fraction of about 50 percent. We did not attempt to analyze the timing or economics of this variant.

in these reactors could begin, given a timely decision, as early as 2008. We estimated costs of this option to range from \$3,100 ± \$2,100 million if the fuel is fabricated at FMEF and the facilities do not pay property taxes and insurance to \$5,100 ± \$1,900 million if the fuel is fabricated in a new plant and the reactor and reprocessing plant do pay property taxes and insurance (present values in 1992 dollars as of start of reactor operation, net of electricity sales).

- (b) As an example of a modular high-temperature gas-cooled reactor (MHTGR), we took a General Atomics design employing conventional steam turbines, in which eight 169-MWe units using a nonfertile, particle fuel containing 100-percent plutonium and burning it to an average of 580 MWd/kgHM could load the nominal 50 tons of WPu in 28 years. The net plutonium destruction in this mode would be about 65 percent, about two times higher than the corresponding figure for typical LWR options; at the higher burnups that may be attainable in this fuel type, once-through net plutonium destruction fractions could be as high as 80 percent. The isotopic composition of this plutonium would be less attractive to bomb-makers than that of typical LWR plutonium, but it would still be usable in bombs. In short, the quantities and isotopic quality of the plutonium remaining in spent fuel from the MHTGR would still represent a significant security risk and would require safeguards comparable to those required for spent fuel from LWRs. Even given an early decision to use this option, we do not think the start of plutonium fuel loading into MHTGRs could occur before 2013 (Table 6-2). We estimate the cost of this option at \$3,900 ± \$2,700 million if the reactor and its fuel fabrication plant do not pay property taxes and insurance and \$5,800 ± \$3,200 million if they do (present value of net cost stream, after subtracting electricity revenues, in 1992 dollars as of start of reactor operation).
- (c) As an example of an advanced liquid-metal reactor (ALMR), we considered a GE design of which four 303-MWe units using fuel containing an average of 10.5-percent plutonium in heavy metal and burning it to an average irradiation of 69.1 MWd/kgHM could load the nominal 50 tons of WPu in 36 years. In this case there would be a net production of plutonium amounting to an increase of a few percent of the amount of plutonium loaded, and the isotopic quality of the discharged plutonium for weapon purposes would be higher than that of typical LWR plutonium. Our estimates of the earliest possible start of fuel loading into the reactors, and of the costs of this option, are essentially identical to our corresponding estimates for the MHTGR.

These findings are consistent with the view, expressed above, that use of the advanced-reactor options in a once-through mode to achieve the spent fuel

end-point would be significantly slower and costlier than accomplishing much the same thing with reactors of currently operating commercial types. We find, further, that other advanced-reactor options—such as the molten-salt reactor, particle-bed reactor, and accelerator-based convertor—would entail even longer development times and greater development investments, with no apparent prospect of gains in performance in the spent fuel mode that could offset the liabilities in timing and cost. Concerning the potential merits of all these advanced reactor types as contributors to national and world electricity supply in the future, we have made no investigation and offer no judgments; that is a question that deserves study in the context of the wide range of considerations appropriate to energy planning. Our conclusion is simply that we have not identified a need to develop and deploy these reactors for the purpose of bringing surplus WPu to the spent fuel standard, and indeed that there would be significant security costs (in the form of delay), as well as monetary costs, in choosing the advanced reactors over currently operating types for this mission.

Concerning the possible use of advanced reactors and/or advanced fuels to pursue the elimination of the WPu, which as noted above might eventually be deemed attractive as a step to follow the transformation to the spent fuel standard and to be applied to plutonium of civilian as well as military origin, we note that:

- (a) Plutonium elimination fractions significantly above 80 percent appear attainable only with the help of fuel reprocessing and plutonium recycle.
- (b) With suitable reprocessing and recycle capability, both thermal and fast reactors can be used for the “elimination” option. While some actinide isotopes cannot be fissioned in a thermal spectrum, thermal reactors can transmute most of them into isotopes that can. The repeated reprocessing and recycle that would be necessary for the elimination option could raise significant safeguards and security concerns.
- (c) ALMRs are superior to current LMR types for the reprocessing-based plutonium elimination mission mainly in that some ALMRs employ integral reprocessing, which does not significantly change the net elimination rate but does offer safeguards advantages over separate reprocessing.
- (d) Accelerator-based conversion (ABC) systems have been under study as a means of fissioning actinides and transmuting fission products in order to reduce the longevity of radioactive wastes. Development of this option is only at the early paper-study stage, and both the proposed subcritical-reactor technology and its fuel-cycle technology are extremely challenging and unproven. If the estimated performance could be attained, however, such systems—subcritical reactors with the needed additional neutrons provided by an accelerator-driven spallation

system—could eliminate plutonium at rates (per thermal gigawatt of capacity) comparable to those estimated for the best of the other elimination-oriented options. This approach's continuous online reprocessing would offer some advantages in waste reduction and in safeguards against plutonium theft (but not against diversion by the system's operators)—shared in varying degrees by other advanced systems that use such reprocessing. The availability of this option to receive plutonium is decades away.

- (e) Particle-bed and molten-salt reactor concepts have also been proposed as eliminators of plutonium. These concepts are both in the preliminary stages of development, and their performance as plutonium eliminators does not appear better than that of other options that could be brought into operation substantially sooner. Like the ABC concept, the molten-salt reactor does offer some safeguards advantages against plutonium theft by virtue of its online reprocessing, and one version of ABC indeed uses a molten-salt subcritical assembly.

IMMOBILIZATION OPTIONS

We believe that WPu immobilization by vitrification in borosilicate glass represents a feasible technology that could meet the spent fuel standard, could be available in the relatively near future (within about a decade hence), and could potentially immobilize all of the nominal 50 tons of U.S. excess WPu in glass in a relatively short time once the vitrification campaign had begun (i.e., in a few years, very likely less than 10).

The vitrification option on which we base this view would entail mixing the WPu with high-level radioactive wastes in the course of vitrifying those wastes preparatory to long-term storage or geologic disposal. The glass logs produced by the vitrification scheme would be resistant to theft by virtue of their large size and mass (3 meters long and nearly 2,200 kg for the large-log variant, 0.5 m long and 250 kg for the small-log variant), and their high radioactivity levels; additional barriers to theft eventually would be provided by isolation in a waste repository and, perhaps, intermixing with outwardly similar waste logs that do not contain plutonium; and, even with a plutonium-bearing log or logs in one's possession, substantial chemical processing in a shielded facility would be required in order to extract the plutonium from it.

The earliest possibility for implementing this vitrification option in the United States would seem to be integrating the vitrification of WPu with the currently planned campaign at the Department of Energy's Savannah River plant to vitrify that plant's defense high-level waste (HLW) onsite at its Defense Waste Processing Facility (DWPF), now expected to begin operation in 1996. Before WPu could be added to this scheme, it would be necessary to resolve

questions about the risk of criticality in the melter (a function of plutonium concentration and melter design) and to make such other modifications in the facility as would be required to handle large quantities of WPu there. These tasks probably could not be completed in much less than a decade from now. The Savannah River Site staff has proposed a schedule in which 50 tons of WPu would be vitrified with HLW during the last eight years of the currently planned 20-year DWPF campaign, approximately from 2005 to 2013. Under this schedule, all 50 tons could be emplaced, at about 1.3-percent plutonium by weight, in the 2,200 large logs already scheduled to be produced in the eight-year period. Since this estimate was made, there have been additional delays in the startup of the DWPF, and it remains possible that currently unforeseen major problems in the HLW vitrification campaign there could substantially stretch out this estimated schedule. We estimate the cost of adding the WPu to the DWPF operation at $\$1,000 \pm \500 million (present value in 1992 dollars at the start of plutonium vitrification operations). These costs are comparable to those of the less expensive among the current-reactor/spent-fuel options.

A smaller waste-vitrification facility is under construction in West Valley, New York—the West Valley Demonstration Plant (WVDP)—with startup expected in 1995 or 1996. With a melter about half the size of the DWPF melter, but similar in technology, the WVDP is to vitrify HLW remaining at that site from the previous commercial fuel reprocessing activities there into about 300 logs similar in size and waste content to the 6,100 logs to be produced at Savannah River. While it would be possible in principle to vitrify some of the surplus WPu in this West Valley melter, it clearly could not do the whole job, and there seems little point in paying for the plutonium storage and handling facilities that would be needed at this site to do only a small part of the job.

Groundbreaking for a vitrification facility similar to Savannah River's DWPF is expected sometime this year at the Department of Energy's Hanford site. It is to be used to vitrify the military HLW now stored at that location, which is roughly comparable in quantity to that at Savannah River. By virtue of its later time schedule, the Hanford facility might be more readily and economically modifiable than the DWPF to accommodate WPu in the vitrification process, all the more so if criticality considerations prove to require extensive changes to the current DWPF design.

A waste-vitrification facility with a nominal output of 1 ton of glass per day has been in operation at the Chelyabinsk-65 site in Russia since 1987. The phosphate glass composition employed at this facility appears to be both less durable and less resistant to recriticality if plutonium is embedded in it than is the borosilicate glass planned for U.S. vitrification facilities. To our knowledge, the costs of modifying the Russian facility to make borosilicate instead of phosphate glass, and to integrate WPu with its process stream, have not been estimated, but these requirements seem unlikely to differ greatly from those we have estimated for the modifications that would be needed to U.S. vitrification

facilities. Russian authorities, however, have so far displayed strong resistance to WPu disposition schemes that “throw away” the plutonium without generating any electricity from it, irrespective of arguments that electricity generation with WPu is costlier than with LEU.

Waste-vitrification plants are operating or soon to operate in France, Great Britain, and Japan, but their use for disposition of U.S. or former Soviet Union WPu would be impeded by the logistical and political problems of international transport of this material, as well as by the cost and difficulty of add-on modifications to integrate WPu with their glass-production process streams.

Certification of plutonium- and radioactive-waste-bearing borosilicate glass as suitable for ultimate disposition in a geologic waste repository will depend on resolution of issues involving the properties of the glass itself (physical integrity, leachability) and the potential recriticality of the plutonium at distant future times if other glass constituents such as boron were to leach out preferentially. A substantial development effort, requiring a decade or more, might be necessary to decide these questions fully, although important insight into them probably can be developed in a shorter time. Depending on the results of such studies, it is possible that plutonium loadings might be constrained to percentages low enough to substantially affect the timing and cost of a vitrification campaign, or that alternative glass compositions might have to be developed.

A number of waste forms other than borosilicate glass have been proposed by various groups for consideration as alternatives for the immobilization of surplus WPu in ways that would, it is argued, meet the spent fuel standard. Besides the phosphate glass mentioned above, these candidates include synthetic rock (“synroc”), cements, and pyroprocessed metals. We do not favor further consideration of phosphate glass, synroc, or cements for the WPu disposition mission in the United States, unless unforeseen developments require it, because we believe this country has chosen borosilicate glass for the waste-immobilization function for sound (and extensively documented) reasons, the suitability of this material for containing significant quantities of plutonium in addition to HLW has already been studied rather extensively with favorable results, and a high cost in delay—hence in security as well—would be paid for reopening the question.

The pyroprocessing approach, which would use technology developed as part of the U.S. Integral Fast Reactor program, would require substantial additional engineering development and construction of major new facilities (including what would amount to a sizable LWR fuel reprocessing plant to provide feed material), and it would produce a waste form that has not been characterized at all for long-term disposition and would probably be unsuitable for emplacement in Yucca Mountain. All this strikes our panel as a prescription for long delays and big investments in pursuit of highly uncertain prospects for solving a problem for which satisfactory approaches—the current-reactor/spent-fuel and borosilicate-glass/vitrification options—are much closer to hand.

Yet another approach that has been proposed would combine the surplus WPu with beryllium so as to generate substantial neutron radiation from alpha-n reactions followed by neutron multiplication in the plutonium. This approach would yield a material with much higher plutonium concentration and lower radiation barrier than either standard spent fuel or borosilicate glass logs containing HLW, hence a much more attractive target for diversion or theft. It would also pose much greater criticality problems than the other final plutonium forms considered here, and would be unlikely to be certifiable for long-term storage in a geologic repository. Although its production might be less costly than that of WPu-bearing spent fuel or borosilicate glass logs, it does not come close to meeting the spent fuel standard and does not, in our view, deserve further consideration.

COMPARISON OF THE CURRENT-REACTOR/ SPENT-FUEL AND VITRIFICATION OPTIONS

It is clear that the current-reactor/spent-fuel option and the vitrification-with-wastes option are the most attractive candidates for reducing expeditiously the security risks of surplus WPu. Our comparative evaluation of these two leading options (drawing heavily on Chapter 6 section “The Comparisons in Summary”) is discussed in the following paragraphs.

In terms of the crucial timing aspect of security, the current-reactor/spent-fuel options and the vitrification-with-wastes option are roughly comparable to each other (as well as superior to all other options). Under the most optimistic assumptions that are defensible, loading of WPu into current-reactor types could begin between 2002 and 2004 and be completed between 2015 and 2025; loading of WPu into waste-bearing glass logs could begin around 2005 and be completed as early as 2013. The timing uncertainties in both cases—relating more to resolution of institutional issues in the reactor case and to resolution of technical issues in the vitrification case—are bigger than the differences in the best-case point estimates we have provided; thus it would not be meaningful to say more than that the two sets of options are comparable.

With respect to those aspects of security that depend on the details of handling, processing, and transporting various plutonium forms, vitrification-with-wastes entails fewer and somewhat simpler steps than the current-reactor/spent-fuel options, and hence may be somewhat easier to safeguard.

With respect to security of the final plutonium forms, the current-reactor options obviously meet the spent fuel standard, and we judge that the vitrification-with-wastes option meets this standard also. The plutonium in the spent fuel assembly would be of lower isotopic quality for weapon purposes than the still weapons-grade plutonium in the glass log, but since nuclear weapons could be made even with the spent fuel plutonium this difference is not decisive. Under typical assumptions, the radiological barrier presented by glass logs would be

about three times smaller than that presented by a fuel assembly (but still very high), and the mass of a glass log—containing, coincidentally, about the same amount of plutonium as a fuel assembly—would be about three times greater.⁶ The difficulty of separating the plutonium from the accompanying materials would be comparable in the two cases.

In terms of security, then, we believe the two options are comparable in timing; the vitrification-with-wastes option has, perhaps, a modest advantage in safeguardability of the handling, processing, and transport steps; and the current-reactor/spent-fuel option has a modest advantage in the barriers associated with the final plutonium form because of the difference in plutonium isotopics. We conclude that the two approaches are comparable in security overall, that either would be adequate, and that no other option known to us is better.

With respect to economics, our estimates indicate that the most likely costs for the vitrification-with-wastes option and for the less costly among the current-reactor/spent-fuel options are comparable at \$0.5–\$2 billion, expressed as the present value, in 1992 dollars as of the start of plutonium operation at the reactor or melter, of the stream of incremental costs associated with plutonium disposition by these means (subtracting electricity revenues where appropriate). The range of best estimates for all of the current-reactor/spent-fuel options extends from \$0.5 billion to about \$5 billion. The lowest central estimate, at about \$0.5 billion, is for the MOX/spent fuel option using currently operating U.S. LWRs that need no modification to use MOX safely, with the fuel fabricated at the FMEF at the Hanford site. Four of the options studied have central-estimate costs around \$1 billion: use of MOX fuel from FMEF in currently operating CANDU reactors in Canada; use of MOX from FMEF in a single, currently mothballed, partly completed PWR that would be completed for this purpose; use of MOX from an entirely new fuel fabrication plant in currently operating U.S. LWRs that need no modification to use MOX safely; and vitrification with defense HLW at the Savannah River site.

Although the central estimates in all cases considered correspond to net costs, our judgmental 70-percent confidence intervals include a possibility of profits from WPU disposition for the case in which currently mothballed, partly completed PWRs are completed for the purpose of plutonium disposition and use MOX fuel from FMEF, and for cases when new, evolutionary LWRs are

⁶ For example, a large glass log of the type expected to be produced at Savannah River's DWPF would contain 1,700 kg of glass in a 450 kg steel jacket; at 1.3-percent WPU in glass, it would hold 22 kg of plutonium; and, at the expected defense HLW content of 20 percent by weight, it would produce a gamma-ray dose rate of 2,600 rem/h at the surface of the container 30 years after the log was produced. By comparison, a MOX fuel assembly from a Westinghouse pressurized-water reactor would have a mass of about 660 kg, would contain about 18 kg of plutonium after irradiation to 40 MWd/kgHM (assuming initial WPU content of 4.0 percent of heavy metal), and would produce a gamma-ray dose rate of 7,900 rem/h at the surface of the assembly 30 years after discharge from the reactor.

built for this purpose as government facilities (paying no property taxes or insurance) and use MOX from FMEF. These profit possibilities depend not only on the costs associated with MOX use falling at the low end of our judgmental 70-percent confidence ranges, but also on the additional electricity generated by the plutonium disposition reactors being marketable at a price of 5.5 cents per kilowatt-hour or higher (1992 dollars) at the busbar. (Given such prices and the same financial assumptions, however, a *higher* profit would be available from a program that completed reactors or built new ones from scratch and used them to generate electricity using LEU rather than MOX fuel, without addressing the problem of plutonium disposition.)

The range of \$0.5–\$5 billion (1992 dollars)—covering the best estimates of net present value, at reactor or melter startup, of most of the options considered—corresponds to \$10,000 to \$100,000 per kilogram of WPu, or \$40,000 to \$600,000 for a nominal “bomb’s worth” of 4–6 kg. Even the higher figure is probably less than what this weapon material once cost to produce, as well as much less than would be spent in the attempt to recover such material if it went astray and incomparably less than would be spent to try to deter or otherwise prevent its use in the form of a bomb in the hands of a potential adversary.

With respect to environment, safety, and health (ES&H), the panel believes that options for plutonium disposition should:

- comply with existing regulations, of the country in which disposition takes place, governing radioactivity and radiation from civilian nuclear-energy activities;
- comply with existing international agreements and standards on the disposition of radioactive materials in the environment; and
- not add significantly to the ES&H burdens that would result, in the absence of programs for disposition of WPu, from appropriate management of civilian nuclear energy generation and of the environmental legacy of past nuclear weapons production.

The panel believes that both the current-reactor/spent-fuel and vitrification-with-wastes approaches, suitably designed, can meet these criteria. Most of the ES&H impacts of WPu disposition using either of these options can be expected to represent modest additions, at most, to the routine exposures to radiation and risks of accident associated with other civilian and military nuclear-energy activities underway in the United States, and that there is no apparent reason that the activities involved in WPu disposition using either of these approaches should not be able to comply with all applicable U.S. ES&H regulations and standards.

While there are differences in detail in the ES&H challenges and risks posed by the two options in some of the activity categories—for example, a somewhat more complicated set of plutonium-handling operations for the reactor options than for the vitrification option, and a greater relative increase in

plutonium content of the final waste form for the vitrification option than for the reactor options—these differences do not consistently favor one class of options or the other, and none is large enough in relative or absolute terms to justify choosing one class of options over the other. ES&H issues that will need further attention in the next phase of study of these options include: developing and testing the systems to ensure adequate safety against criticality accidents in the melter for the vitrification option; confirming the conditions under which full-MOX cores can be used without adverse impacts on safety in reactors of currently operating commercial types; and determining the conditions that will provide adequate assurance against long-term criticality in geologic repositories containing either spent fuel or glass logs from plutonium disposition operations. These issues differ considerably in the nature and complexity of the work that will be needed to settle them to the satisfaction of the technical and regulatory communities, but it is the panel's judgment that suitable approaches exist for all of them.

Both from the standpoint of ES&H and from the standpoint of security, analysis and comparisons are facilitated by the circumstance that both of the leading-candidate classes of options would be adding WPu to a set of nuclear activities that would be going on in any case, and both would leave the residual WPu in a waste form—spent fuel in one case and HLW-bearing borosilicate glass logs in the other—which will exist in large quantities and will need to be safely managed whether used for WPu disposition or not. We emphasize, in this connection, that a U.S. geologic repository is not likely to be ready to receive wastes of any kind before 2015. Vitrified waste logs, with or without plutonium, will need to be stored in engineered facilities until a geologic repository is ready to receive them; and plutonium-containing spent fuel from nuclear-reactor operations, whether WPu has been incorporated in some of it or not, will need to be stored at reactor sites or at other commercial spent fuel storage facilities until a repository is ready.

RECOMMENDATIONS

Two options have emerged from our investigations as the most promising ones for the disposition of surplus WPu: the use of LWRs or CANDU reactors, employing MOX fuel in a once-through mode, to embed the WPu in spent fuel that would be emplaced eventually in a geologic repository (together with the larger quantity of such spent fuel that will exist in any case from ordinary nuclear electricity generation); and vitrifying the WPu, together with defense high-level radioactive wastes, in borosilicate glass logs of the type planned for use in the immobilization of defense HLW in any case, again for eventual emplacement in a geologic repository.

These options were selected on the basis of evaluation criteria spelled out in the 1994 report of the parent committee and in Chapter 3 of this report, above

all the importance of being able to proceed expeditiously to reduce what the parent committee characterized as "the clear and present danger" posed to national and international security by the existence of large quantities of surplus WPu in the form in which it emerges from the weapon dismantling process. The superiority of these two options to the many others in the panel's purview has been established not only by our own analyses and those of the parent committee (which drew on our preliminary findings), but also by evaluations conducted independently by the U.S. Department of Energy and by a number of other groups.

Although both of these options are technically feasible and have been recommended by the panel in part because they can be deployed comparatively rapidly, some significant uncertainties accompany both of them. The areas of uncertainty in the current-reactor/spent-fuel option are primarily in licensing and public acceptance. Those in the borosilicate-glass/vitrification option relate mainly to the technical issues of plutonium-HLW mixing and criticality. Further paper studies will not significantly reduce these uncertainties, but the experience gained in the implementation process would.

Since it is crucial that at least one of these options succeed, since time is of the essence, and since the costs of pursuing both in parallel are modest in relation to the security stakes, we recommend that project-oriented activities be initiated on both options, in parallel, at once. DOE should assign sufficient resources (both funding and personnel) to manage pursuit of both options in parallel. In connection with the current-reactor/spent-fuel option, work should be started to seek out specific reactors and MOX fabrication options that would minimize multiple plutonium transportation steps so as to reduce this aspect of security risk, to identify locations that are most amenable to public acceptance, and to ascertain the willingness and conditions of the plant owners to participate. The detailed engineering studies should be completed and licensing applications submitted to the Nuclear Regulatory Commission. From that base, the sound cost estimates, schedules, and financial plans could be prepared that are essential to considering full project authorization. In connection with the borosilicate-glass/vitrification option, laboratory work and realistic testing should be started to address the technical uncertainties. Research and development plans, program schedules, and key milestones should be defined. As the results of the research and development are obtained, detailed cost estimates and ES&H analyses can be submitted to the Department of Energy and the Nuclear Defense Facilities Safety Board in pursuit of project authorization.

The pursuit of the two options in parallel, as project-oriented tasks with near-term milestones and aggressive schedules, should be aimed at bringing both processes on-line by the end of the century or as shortly thereafter as possible. It would be a mistake to spend tens of millions of dollars and additional years of paper studies to try to demonstrate, in the absence of actual work toward deployment, which of the two options should be selected over the other. It

will likely be less expensive in the long run, and clearly superior from the security and ES&H perspectives, to proceed with both now. If either option falters due to technical, licensing, or other difficulties, the pursuit of the other option can continue without the loss of time that would have been associated with choosing early and choosing wrong. Indeed, it may prove to be desirable to implement both options for different parts of the stockpile: as one example, it may turn out that reactor approaches are preferable for the relatively pure plutonium in the components of dismantled weapons, while vitrification may prove to be the better alternative for the tons of plutonium that exist in scrap, solutions, and other forms.

We recommend, therefore, that the Department of Energy's Programmatic Environmental Impact Statement process (scheduled to be completed in 1996) should be oriented toward a decision to pursue both the current-reactor/spent-fuel option and the vitrification-with-wastes option, not toward a decision to eliminate one or the other. Subsequent preparation of Environmental Impact Statements (EISs) for both options, and the participation of the public in these processes, should proceed in parallel too. The inclusion, in the EIS and public participation processes, of the results of the specific project-oriented activities mentioned above will be essential to the success of those processes. Full project authorization for one of the options would not be granted until the EIS is completed and approved.

The fundamental objective of the WPu disposition program will not be achieved unless the Russians carry out a disposition program in parallel, on a similar time scale, and adhering to disposition standards equivalent to those of the United States. The above project-oriented activities would lend themselves to forming joint projects with the Russians to assure such a parallel approach. Joint projects will also serve to develop a technical consensus on the disposition process and standards, which, as pointed out elsewhere in our report, does not exist today. The panel recommends that the United States immediately initiate joint project-oriented activities with Russia covering both the MOX and the vitrification options.

Studies of a follow-on nature should continue on the longer-range questions of whether and how the residual security risks of weapon and other plutonium should eventually be reduced beyond the spent fuel standard. It is essential, however, that such longer-range studies not be allowed to draw resources or attention from the pursuit of the two options closest to hand for moving rapidly toward achieving that standard for the weapons plutonium that poses a "clear and present danger" today.