Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines

Edited by

Ashot Sarkisov
and Alain Tournyol du Clos

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II. Mathematics, Physics and Chemistry – Vol. 215

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PREFACE

The NATO-Russia Advanced Research Workshop (ARW) "Scientific and technical issues in the management of spent fuel of decommissioned nuclear submarines" was held in Moscow, Russia, on September 22-24, 2004. Attendance at this workshop was approximately 100 with participants from Russia, France, Norway, Denmark, Sweden, Finland, United Kingdom, Japan, United States and NATO.

This was the fourth ARW in this series of the North Atlantic Treaty Organization (NATO)-sponsored workshops in Moscow over the last 10 years. Like the three previous workshops, the fourth NATO-Russia ARW was focused on a very important global challenge of the present – complex decommissioning of taken-out-of-service naval and civil nuclear vessels and environmental rehabilitation of contaminated facilities, and terrestrial and aquatic systems concerned by everyday running of different type of nuclear vessels.

The first NATO-Russia ARW (June 1995) addressed the general issues of decommissioning of nuclear submarines. The second ARW (November 1997) was focused on analysis of the risks associated with withdrawal from service, storage and dismantlement of nuclear submarines. The third workshop (April 2002) considered scientific problems and unresolved issues remaining in the decommissioning of nuclear-powered vessels and the environmental remediation of their supporting infrastructure.

Each following ARW logically went deeper into the problems identified by the previous ones, with special emphasis on the most intricate scientific and technical issues requiring appropriate decisions at a particular instant.

The fourth ARW also addressed the problems necessitating urgent solution. First and foremost those were the issues of safe management of spent nuclear fuel and, especially, of spent fuel of damaged nuclear-powered installations. Secondly, temporary and long-term storage and ultimate disposal of spent nuclear fuel and radioactive waste also necessitated a special consideration. Finally, the ARW examined the challenges of both rehabilitation of the whole decommissioning infrastructure of nuclear-powered vessels and environmental remediation of the affected terrestrial and aquatic systems.

In addition, the participants of the fourth ARW heard the very first results from the team developing a Strategic Master-plan ("Master-plan") for complex decommissioning of Russian nuclear submarines, nuclear-powered surface ships, service vessels and civil-fleet icebreakers.

The fourth NATO-Russia Advanced Research Workshop was sponsored and funded by the Security-Related Civil Science and Technology Program of the Public Diplomacy Division of NATO, the Commissariat for Atomic Energy (France) and the Russian Academy of Sciences. The sponsorship of all of the above organizations is gratefully acknowledged.

The Workshop was perfectly organized in Russia by the Nuclear Safety Institute of the Russian Academy of Science (IBRAE). The technical program was developed between IBRAE and CEA/STXN (French expertise in nuclear propulsion). Through an efficient and friendly cooperation they produced a technical program that was broad in scope, technically deep, challenging, relevant and interesting. The efforts of many individuals from IBRAE and CEA in producing the Workshop are here especially recognized.

The proceedings of the 2004 NATO-Russia ARW taken together with the earlier published papers of the three previous workshops of 1995, 1997 and 2002 represents the most complete set of materials on the issues related to complex decommissioning of nuclear vessels and nuclear-powered ships.

Though this book is primarily intended for scientists, engineers and technicians concerned with the challenges of decommissioning nuclear-hazardous objects, we do hope it will be also found interesting and useful by specialists involved - in one way or another - into development and implementation of standard, managerial and engineering documents and decisions on the problems under consideration.

OVERVIEW OF THE THREE PREVIOUS (1995, 1997 and 2002) NATO-RUSSIA WORKSHOPS GENERAL ISSUES OF SPENT NUCLEAR FUEL AND RADIOACTIVE WASTE MANAGEMENT

SPENT FUEL OF DECOMMISSIONED NUCLEAR SUBMARINES: QUESTIONS TO BE ADDRESSED

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France

Abstract

Since the first ARW on nuclear submarines in 95, management of the spent fuel coming from decommissioned nuclear submarines has become a growing preoccupation.

Spent fuel represents in the same time a valuable resource, a major radioactive hazard for nearby populations and a possible threat if stolen by terrorist organizations.

The Russian Federation has a closed fuel cycle policy which asks for all the spent fuel to be sent to the Mayak reprocessing plant.

Nevertheless implementing this policy proves to be difficult due to many factors: lack of infrastructure for handling and transportation, degraded status of many fuel elements, variety of types of fuel, not all of them being reprocessible, lack of financial resources, etc.

It is then necessary to establish an overall strategy addressing all the questions, hierarchizing the problems and recommending intermediate or temporary solutions when and if needed.

This paper will make a review of communications on spent fuel management made during previous ARW (95, 97 & 02) and give a general frame to the reflections that will be presented during the oncoming workshop.

Closed of open fuel cycle for nuclear submarines?

Each country which has decided to turn to nuclear energy to fulfil its needs, has also to decide what to do with the spent fuel unloaded from nuclear plants.

Spent nuclear fuel can be considered either as a resource, as it contains a significant quantity of Uranium and Plutonium, or as a radioactive waste, as it contains highly radioactive isotopes of minor actinides and fission products.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 3–9.

Some countries, including the Russian Federation and France, have chosen a closed fuel cycle meaning that the nuclear fuel is reprocessed in a dedicated plant, thus allowing the remaining Uranium and Plutonium to be incorporated into fresh fuel.

Other countries, for instance the USA, Sweden, Finland, have decided for an open fuel cycle.

In both cases, the final waste (i.e. the spent fuel elements or the waste produced in reprocessing them) has to be stored permanently and safely somewhere.

Spent fuel from decommissioned nuclear submarines differs from spent fuel from nuclear power plants in many ways:

- on one hand, enrichment is higher (sometimes over 90%) and burn-up lower, making the fuel elements a more valuable resource,
- on the other hand, the very conception of fuel (which answers to specific military constraints) renders it more difficult to reprocess.

Nevertheless it seems more cost effective to treat spent fuel from nuclear submarines in the same way than spent fuel from nuclear power plants, even if some adjustments have to be made at the front end of the reprocessing plant.

Technical matters

The Russian policy is to reprocess spent fuel from nuclear submarines, but is it always possible?

There are many types of fuel for submarines, various enrichments, different claddings, etc; all the types are not today acceptable by the Mayak facility, either because the reprocessing technology is not available or because there are safety limitations.

Let’s call Type I the fuel that is reprocessible and Type II the fuel that is not.

The first question is to choose a strategy for Type II fuel:

a) develop a specific facility and/or method to reprocess that type of fuel,

b) send the fuel to a deep permanent repository,

c) send the fuel to an intermediate storage (50 years).

Of course strategy c) can be a first choice, giving time to build a new facility or a permanent repository.

Now as far as Type I fuel (reprocessible) is concerned, the fuel elements must be able to be transported and handled in the Mayak facility; this may be a problem for damaged fuel elements.

The second question is then: how do we know in which condition are the fuel assemblies?

There are three possibilities: A, B & C giving three sub-categories of fuel:

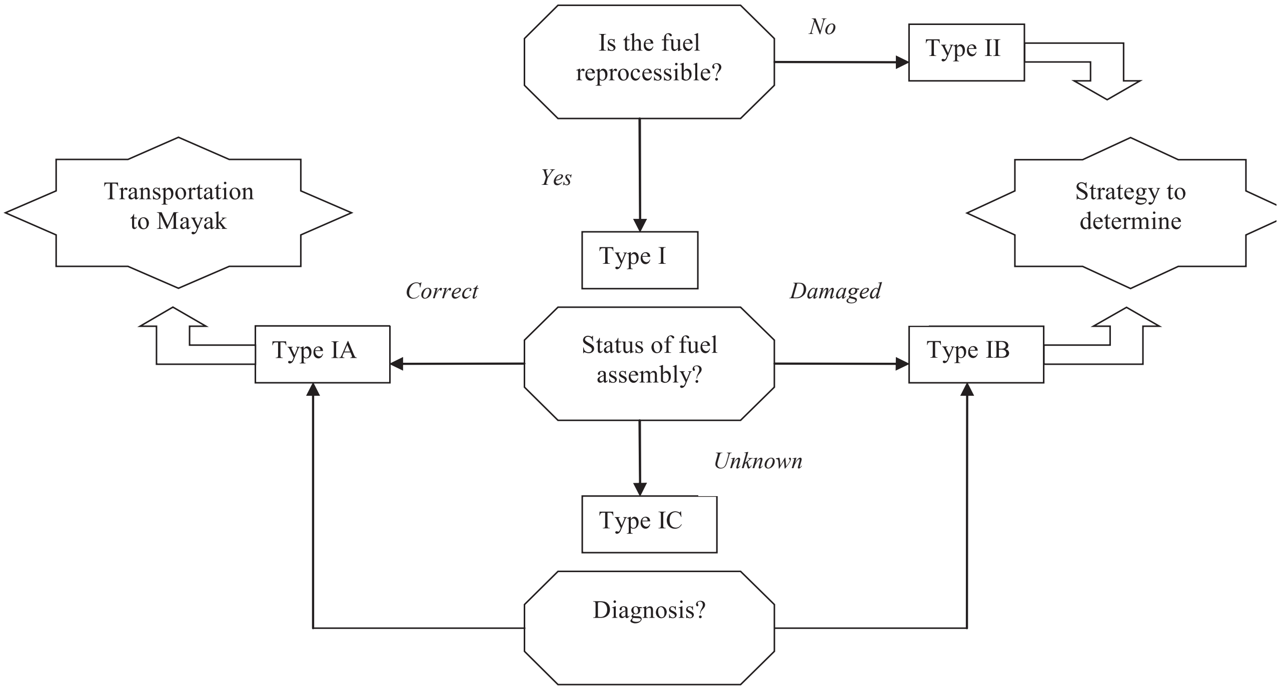
- Type IA are the fuel assemblies of which we know for sure that they are in good condition (for instance, the fuel that is still aboard submarines)
- Type IB are the fuel assemblies of which we know for sure that they are damaged (broken fuel assemblies in pools or in containers)
- Type IC are the fuel assemblies of which we do not know what is their real status (most of the fuel assemblies that have been stored in containers in the open air are in dubious status).

Type IA fuel is due to Mayak.

For Type IB fuel, we have the same need to define a strategy than for Type II fuel.

For Type IC fuel, we must develop a method of diagnosis, before we can put them either with Type IA or Type IB.

The drawing below summarize the decision making process.


Is the fuel reprocessible?
No
Type II
Yes
Type I
Transportation to Mayak
Strategy to determine
Correct
Damaged
Type IA
Status of fuel assembly?
Type IB
Unknown
Type IC
Diagnosis?

Need for an intermediate storage

Normally there is no need for an intermediate storage, but the pace to which nuclear submarines have been decommissioned has created bottlenecks which in turn resulted into a difficult situation.

First bottleneck lies in the unloading capacity: infrastructure for unloading and pools for storage of unloaded fuel (either aboard service ships or in land facilities) are not always available, financial resources are not there, and at the end of the day many decommissioned submarines have still there fuel on board, sometimes more than ten years after they were pulled out active service.

As far as nuclear safety is concerned, this situation is not totally unsatisfactory; criticality accident has been rendered nearly impossible and the residual power is almost negligible.

But when unloading will be possible, it may turn out to be much more difficult than if it had been done in time; specially so for the older submarines for which it may be necessary to rebuild infrastructure, to develop new procedures and tools (this is the case, for instance, for the Alpha class submarines).

Second bottleneck lies in the transportation capacity: fuel elements that have been unloaded stay for many years in pools or in containers stored in the open air by want of transportation capacity. Those *de facto* intermediate storage lack the safety environment that would have been asked for if they had been conceived from the beginning as storage facilities; they also lack correct physical protection and – though spent fuel from submarine is not the easiest way to a nuclear weapon – could thus attract the attention of terrorist organizations.

Were those bottlenecks suppressed, the handling and transportation capacity being significantly increased, it is likely that a new bottleneck would appear at the entrance of the Mayak reprocessment plant.

Altogether, a comprehensive assessment of the necessary time – taking into account the foreseeable cash flow earmarked to the programme – to attain a stabilized situation is necessary. It may well justify the creation of dedicated intermediate storage facilities for spent fuel (for instance one in the North-West area and one in the Pacific area).

Damaged cores

A few submarines have damaged cores that cannot be unloaded by the usual procedures; one of them has been prepared for immersion which renders an intervention still more complicated.

Here again decisions will have to be prepared and taken: will those submarines be considered – partially or in totality – as final waste and hence will they have to be stored permanently somewhere or will it be cost-effective to develop specific processes to recover the damaged fuel?

Previous workshops

All these questions are not new, but, as time goes by, they stir a growing interest.

In the first NATO workshop dedicated to nuclear submarines, in 95, 1 paper out of 42 addressed fuel problems; subject was: “Overview of defuelling approaches used to deal with reactors that have major core damage” and it dealt with the Three Miles Island experience.

In the second NATO workshop, in 97, we had 4 papers out of 36 dealing with spent fuel. Subject were:

- Geological aspects of handling of radioactive wastes and spent nuclear fuel in decommissioning of nuclear submarines
- A potential method for stabilization and packaging of damaged naval spent fuel
- Principal solutions on environment protection from the spent nuclear fuel of stationary and transportable nuclear power plants with the use of metal and concrete containers
- Nuclear fuel cycle for Russian ship spent nuclear fuel : reprocessing or direct disposal

In the third NATO workshop, in 02, 7 out of 43 papers dealt with fuel.

- Radio ecological monitoring of defuelling of damaged spent fuel from storage facilities of floating shops
- Actual status and problems of spent nuclear fuel management at coastal facilities of the north-west region and the far east region of Russia
- Transport and technological flow sheets for management of spent nuclear fuel from nuclear submarines under utilization in the north-west region and the far east region of Russia : problems and solutions
- Storing and shipping of spent nuclear fuel from ships : new engineering solutions and probable radiation effects of an accident
- Options for the handling and storage of nuclear vessel spent fuel
- Disposal of spent nuclear fuel and radioactive wastes in the decommissioning of French nuclear submarines
- Step by step solution to the project on long-term storage of spent fuel from nuclear submarines with heavy liquid metal cooled reactors

As can be seen from the titles, many questions have already been broached, but we need now a fully comprehensive approach

Conclusion: work to be done

Since the last workshop, the G8 members, in Kananaskis, put forward the Global Partnership against the Spread of Weapons and Materials of Mass Destruction.

Dismantlement of decommissioned nuclear submarines and remediation of ex-naval bases were considered as a first priority within this programme, because of the spent nuclear fuel they still contain.

The cleaning up of the present situation demands and will demand in the next years perseverance and a lot of money.

To make sure, and to convince donor countries, that this money is spent in the most effective way, it is absolutely necessary to establish an overall strategy addressing all the questions, hierarchizing the problems and recommending intermediate or temporary solutions when and if needed.

JUSTIFICATION OF PRIORITY LINES AND OBJECTIVES WHEN RESOLVING THE CHALLENGES OF COMPLEX DECOMMISSIONING OF NUCLEAR SUBMARINES

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In the late 1980s and 1990s Russia was facing a serious problem: large-scale withdrawal of nuclear vessels from military service began. The main causes were the expiration of design service life of such vessels and the commitments of the Russian Federation under the START Treaty.

An important aspect of the current situation is the impact of the decommissioned fleet on the environment that is of concern not only in Russia but also in the foreign countries. This concern has its specifics in the context of the role the Arctic Region plays on the global scale.

The global fallouts caused by nuclear weapons tests have appeared to be the major contributor to the Arctic region's contamination (10^{16} Bq). The contribution of Liquid Radioactive Waste (LRW) discharges to marine environment, which had been done before Russia joined the agreements of 1993 on comprehensive prohibition of the discharge of radioactive waste to seas, was significantly less (< 10^{15}Bq). The contribution conditioned by the decommissioning of nuclear fleet and related infrastructure subject to decommissioning is about 10^{14} Bq. After nuclear weapons tests have been ceased, the situation in Arctic seas is gradually improving.

However a significant amount of Spent Nuclear Fuel (SNF) at Nuclear Submarines (NSs), Coastal Maintenance Bases (CMBs) and other complex decommissioning objects represents a serious potential hazard. The activity accumulated at the decommissioning objects is nearly 40 times higher than that of fallouts resulted from nuclear weapons tests. The situation aggravates due to a high concentration of radiation-hazardous objects in the North-Western Russia. This is clearly seen in Fig. 1 showing locations of decommissioning objects and distribution of radiation potential over the Murmansk coast of the Barents Sea.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 11–33.

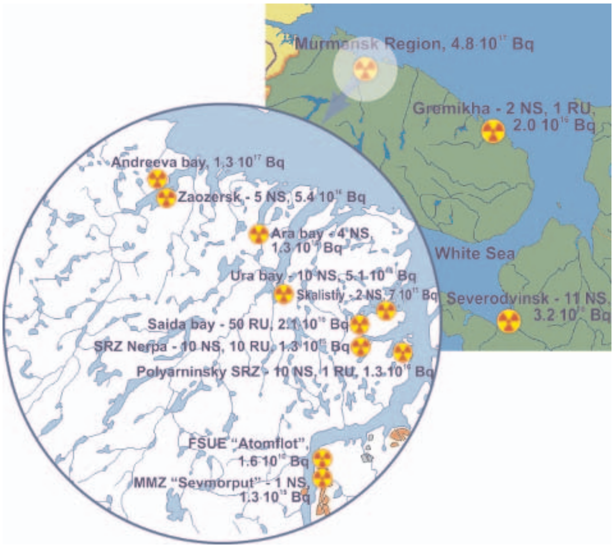

Figure 1. Distribution of radiation potential of the decommissioning objects in the North-Western Russia

Local radionuclide contamination sources in the nuclear fleet waterborne storage, dismantling and repair centers have become more apparent on the background of recent improvements of the radioecological situation in the Northern Seas. As an example, Fig. 2 shows the concentrations of man-caused radionuclides in bottom sediments of nuclear vessel waterborne storage and supporting infrastructure location centers. It is clearly seen that concentrations of ^{60}Co in some locations exceed the background level by 30-70 times, those of ^{137}Cs exceeding the background by hundreds and thousands of times. Despite the fact that so far even such contamination levels have not constrained economic activities in the region as a whole, their adverse impact on the environment will be aggravating due to continuous degradation of technical conditions of nuclear vessels and their maintenance facilities if preventive measures are not taken promptly.

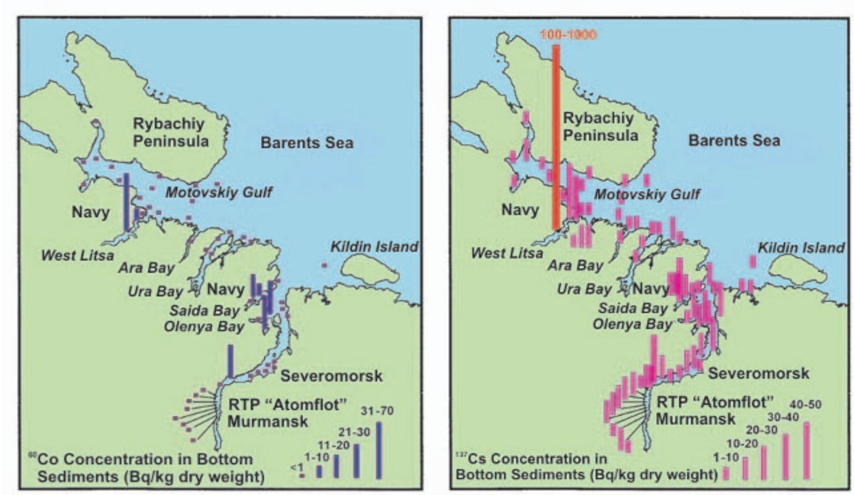

Figure 2. Concentrations of man-caused radionuclides in bottom sediments in nuclear vessel waterborne storage and supporting infrastructure location centers

It should be emphasized that along with acceptable, on the whole, averaged contamination local deviations have been discovered with contamination levels considerably above the permissible limits. The examples are, in particular, CMBs in Andreeva Bay and Gremikha where highly contaminated sections of outer walls of buildings and structures within the sites of these bases are found. At certain areas of the CMB in Andreeva Bay γ-radiation dose rates, surface contamination and specific concentrations of ^{137}Cs and ^{90}Sr exceed the background values by tens of thousand times. Similar unfavorable radioecological situation is observed at some CMB facilities in Gremikha.

The actual situation related to complex decommissioning of nuclear vessels is characterized by the following:

– not a single submarine out of 117 NSs taken out of operation in the North-Western Russia has been decommissioned completely as to form a Reactor Compartment (RC) unit and place it for long-term monitored on-shore storage. Pads for such storage have not been built yet;
– 56 NSs and 62 RC units kept afloat pending decommissioning require continuous buoyancy monitoring while their technical condition degrades;
– two former naval CMBs in Andreeva Bay and Gremikha house a large amount of SNF and Radioactive Waste (RW) which real condition is not quite clear. The CMB sites require a large-scale rehabilitation activities;
– the issue of complex decommissioning of a large number (23) of Maintenance Vessels (MVs) and one Nuclear-Powered Surface Ship (NPSS) berthed in Severodvinsk has not been decided upon yet;

- the problem of SNF of Liquid-Metal Coolant (LMC) reactors, damaged (off-standard) VVER fuel and uranium-zirconium fuel of reactors of nuclear icebreakers requires special decisions and engineering solutions;
- decisions have not been taken yet regarding selection of SRW regional repository locations and construction of related structures.

Despite ten years of works and implementation of related measures, the integral amount of necessary capital investment for elimination of environmental threats issuing from the Russian naval decommissioning objects, as of 2004, is estimated at $ 4 billion. With due regard for real capacities of funding by the Russian Federation (RF), appropriate solution of this challenge over 10-12 years will be only possible if international assistance is used - in particular that provided by the Northern Dimension Environmental Partnership (NDEP) initiative.

Under actual conditions, efficient spending of money and, especially, they investment into the most topical decommissioning and environmental rehabilitation areas are the issues of paramount importance. In this context the problem of justification of priorities, when addressing the complex decommissioning-related challenges, becomes especially acute.

That was the reason for initiation at the turn of 2003 by the European Bank for Reconstruction and Development (EBRD) jointly with Ministry for Atomic Energy of the Russian Federation (RF Minatom, presently Rosatom) of a project “Implementation of the Initial Phase of Development of a Strategic Master Plan (SMP) for Decommissioning of Nuclear Submarines, Ships and Service Vessels Taken out of Operation and Environmental Rehabilitation of Related Radiation-dangerous Facilities in North-Western Russia”. Three leading Russian research institutes in the related area – Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAS), Russian Research Center “Kurchatov Institute” (RRC KI) and Research and Development Institute of Power Engineering (NIKIET) were entrusted with the project implementation. Over 50 experts of the three institutes and other entities participated in the work.

The goal of the first project’s phase was to justify top-priority tasks which implementation should be initiated immediately. In addition, further development of the result was proposed up to generation of top-priority measures (projects).

The SMP Report formulates proposals agreed upon with the RF Agency for Atomic Energy (Rosatom) on the SMP role and place within the framework of the Russian legislation. According to these proposals the Strategic Master Plan should:

- be the basis for selection of projects on complex decommissioning and environmental rehabilitation of radiation-hazardous objects taken out of the Navy service and those of the civil nuclear fleet;
- be the basis for strategic decision-making by the Government of the Russian Federation as regards the decommissioning and rehabilitation and management of SNF and RW;

- facilitate evaluations by the donor countries of technical and economic efficiency of implementation of the decommissioning projects including improvement of nuclear, radiation and environmental safety and physical protection;
- facilitate making of balanced and justified decisions with due accounting of relevant interests of the Russian Federation and the donor-countries; and
- contribute to coordination of activities, perform control over their target-oriented application and the results at all phases of their implementation.

The SMP was developed in close interaction with Rosatom, being the State Customer and Work Coordinator for the complex decommissioning under the RF Governmental Decree, as well as with the Russian Navy, the Russian Federal Agency for Shipbuilding – the Federal Agency for Industry, and other agencies and organizations concerned.

SMP generation goal was to optimize activities targeted to expedite elimination of nuclear, radiation, and chemical hazards and threats in the North-Western Russia for the population and environment taking into consideration the interests of the neighboring territories and Europe as a whole.

The work under SMP (SMP Report) comprises six interrelated chapters (tasks) aimed at attaining an ultimate goal – identifying top-priority measures (projects). General structure of interfaces between the tasks solved at the initial SMP development phase and their representation in the SMP final report is demonstrated in Fig. 3.

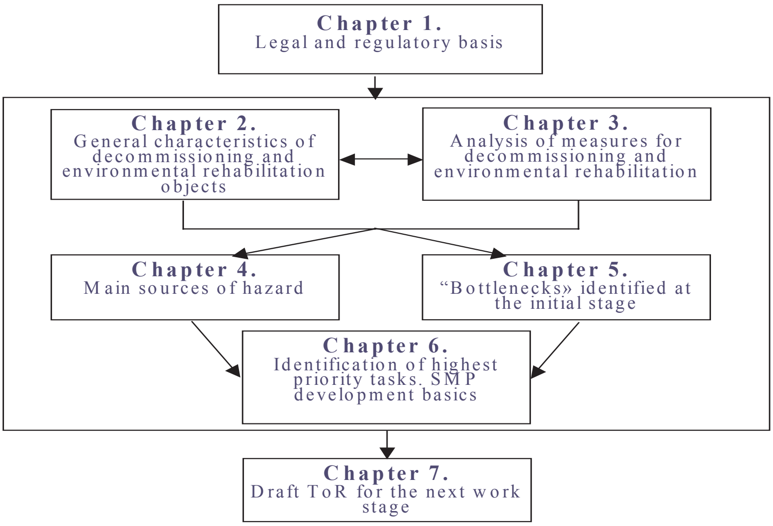

Figure 3. Hierarchy of the tasks solved at the SMP initial development phase

As compared to all previous conceptual documents addressing similar subjects, the SMP has the following important peculiarities:

1. Despite the fact that the SMP was mainly addressed to Rosatom and implemented in close contact and permanent interfaces with the Agency, it should not be considered only as an agency-level document, because the principal institutions-developers were beyond the jurisdiction of Rosatom.

2. Under the SMP the issues of complex decommissioning and environmental rehabilitation of not only Russian naval objects but also of those of the Russian Civil Fleet were considered for the first time.

3. The SMP represents an "open" document due to a huge body of included information and the expected wide distribution.

4. For the first time the challenges of prioritization of the integrity of objects, tasks, activities and specific projects were formulated under SMP.

Within the frames of Task 1 the SMP Report generalizes, systematizes and analyzes a broad range of documents regulating activities in the areas covered by the SMP The completeness and maturity of the legal basis condition effectiveness of organizing, planning and implementing the complex decommissioning and rehabilitation programs: from the level of international cooperation through specific process operations.

As a whole, the legal and regulatory basis of the Russian Federation complies with and corresponds to generally acknowledged approaches to nuclear, radiation and environmental safety, to management of SNF and RW that creates favorable prospects for international cooperation in the field of NS complex decommissioning. The Russian legislation in force assures safety of planned/to be implemented activities on decommissioning of NS and radioecological rehabilitation of coastal facilities, terrestrial and aquatic systems.

However, an analysis of the existing legal and regulatory documents has revealed a number of shortcomings and bottlenecks in the legislation. The main of them are:

1. The Federal Law “On the Use of Atomic Energy” does not cover nuclear power installations of defense applications, whereas the Federal Law “On the Defense-Purpose Nuclear Power Installations”, which is called for to supplement it, has not been adopted yet.
2. In the field of radiation safety of the environment - regulated, in particular, by the Federal Law “On the Environmental Protection” - the legislation does not address the issues of standardization of radiation quality of objects of environment. Moreover, a number of the law provisions contradicts the existing legislation and real practice of radiation protection of the environment which use the sanitary-hygienic approach where “if man is protected, the environment is protected too”, as per the ICRP Recommendations in force (Publication 60).
3. The RF legislation does not determine SNF as an independent object of regulation.
4. Some provisions of other legislative enactments regulating radiation and environmental safety need to be revised, for instance, the Water Code of the Russian Federation and the Law “On the Radiation Safety of the Public”.
5. “The Joint Convention on the Safety of Spent Nuclear Fuel Management and on the Safety of Radioactive Waste Management” signed in 1999 has not been ratified yet; there is no law “On the Management of RW and SNF”.
6. The standard and legal base in the area of managing low-level and medium-level Solid Radioactive Waste (SRW) needs further development and extension.
7. Procedures for the Russian Federation joining the amendment LC/51 and “The 1996 Protocol to the London Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter” (1972) have not been fulfilled.
8. The Vienna Convention on Civil Liability for Nuclear Damage” has not been ratified. The Federal Law “On the Nuclear Liability” which establishes responsibilities for causing nuclear damage and regulates the mechanisms of financial compensation for the caused damage (incl. the state guarantees’ mechanism), including international cooperation and assistance projects, has not been adopted yet.

The SMP Report formulates specific proposals on improvement of standard and regulatory basis in the nuclear fleet complex decommissioning area at all phases.

In recent decade many international and national scientific conferences and workshops have been held, a lot of research done, and many papers published with data on the general condition of the decommissioning objects at the Russian Federation nuclear fleet. However the published materials are fragmented, sometimes contradictory and out-of-date. In addition, some published information has gaps including the data topical for SMP development.

The second part of SMP Report generalizes, clarifies and supplements, if necessary, this vast and unstructured information. In a number of cases special calculations were done to obtain additional data. Controversial information was clarified directly with Minatom of Russia, Navy and the Russian Agency for Shipbuilding, which are the official holders of the decommissioning-related information.

The SMP Report analyzes and presents data on the following objects:

– 56 NSs including 31 NSs with SNF on board;
– 62 reactor compartment units including 2 RCs with SNF;
– 2 coastal maintenance bases;
– 27 nuclear maintenance vessels;
– 1 nuclear-powered surface ship (cruiser *Admiral Ushakov*);
– 5 shipyards;
– 2 TUK accumulation pads;
– storage locations of about 44000 Spent Fuel Assemblies (SFAs);
– storage locations of about 24000 m^3 of SRW;
– storage locations of about 10000 m^3 of LRW.

Besides, the following civil nuclear-powered ships and objects of related supporting infrastructure are also considered:

– 8 nuclear icebreakers;
– 1 nuclear-propelled lighter ship;
– 5 nuclear maintenance vessels;
– RTP Atomflot enterprise.

Detailed data on number, location, radiation potential and technical condition of all objects of complex decommissioning and environmental rehabilitation are presented and analyzed.

Those data formed a basis for subsequent analysis of hazard sources, the “bottlenecks” and problem issues typical for the current situation with radiation-hazardous facilities in the North-Western Russia.

The SMP Report formulates specific proposals on improvement of standard and regulatory basis in the nuclear fleet complex decommissioning area at all phases.

In the third part of SMP Report:

- the ultimate goals of decommissioning and environmental rehabilitation of all types of objects have been formulated;
- the main work areas and process stages for NS, MV, NPSS and CMB as well as SNF and RW have been identified;
- the capabilities of all floating and coastal defueling equipment have been considered along with production capacities of enterprises performing NS dismantlement and RC unit making up;
- the conditions and capabilities of process and transportation systems of SNF and RW management in Murmansk and Arkhangelsk Regions have been analyzed.

It should be emphasized that the ultimate goals of complex decommissioning and environmental rehabilitation of radiation-hazardous objects have been formulated on basis of a detailed technical and economic analysis of possible ways to solve the existing problems, which was carried out at the earlier work stages by Russian institutes pertaining to various agencies.

The fourth part of SMP Report addresses all main sources of real and potential hazard when conducting complex decommissioning and environmental rehabilitation activities. All complex decommissioning and environmental rehabilitation objects considered in the SMP Report have been distributed over three categories in terms of their radioecological and any other hazard (risk) for personnel, population and environment, i.e. over the following risk categories:

- real risks conditioned by the current status of the objects of complex decommissioning and environmental rehabilitation;
- potential risks non-associated with technologies of any works on decommissioning and environmental rehabilitation (two sub-groups have been considered under this category: the risks growing with time and the risks constant in time); and
- potential risks associated with technologies on complex decommissioning and environmental rehabilitation.

The performed analysis of main sources of hazard has shown that all decommissioning and environmental rehabilitation objects are related - to certain extent - to all considered risk categories.

The SMP Report comprises a variety of data on all mentioned types of real and potential hazard in the complex decommissioning area. In the course of analysis of potential sources of hazard the most important – for every type of objects – emergency situations have been identified, their simulation using mathematical models performed, the related integral damage and the risks for population and servicing personnel estimated.

The performed *under Part 4 of SMP Report* analysis of potential hazards associated with the current situation allows the following statements:

The complex decommissioning activities do not produce a significant radiation impact on the population and environment. By contrast, a significant pollution of the environment with noxious chemical substances generated in the course of NS dismantlement and accumulation of hazardous waste takes place. Near the pollution sources at the sites of shipyards concentrations of noxious chemicals in the atmosphere and seawater exceed Maximum Admissible Concentrations (MAC) by several times.

1. The operations with SNF and RW at CMB in Andreeva Bay and Gremikha are most hazardous for personnel, where in individual buildings, storage facilities and at open-air pads local radioactive contamination areas are found with rather high radiation levels.

2. In terms of radiation potential, SNF stored in unsatisfactory conditions at CMB in Andreeva Bay and Gremikha is comparable with that of afloat-stored taken-out-of-service NSs. Taking into account actual condition of protective barriers at non-defueled NSs, the work on developing and implementing in expedite manner the projects for safe preparation for off-shipment and mere off-shipment of SNF beyond these facilities should be considered as a high priority task.

3. An increase in risk of NS sinking with time of their keeping afloat sets the priority to the task of defueling the NSs with the longest waterborne storage times.

4. The main source of real and potential hazard is SNF stored at CMBs, on NSs and Floating Service Vessels (FSVs) and civil nuclear fleet objects: FSVs *Lepse*, *Lotta* and *Imandra*. SNF storage at CMBs and on FSV *Lepse* is the most hazardous.

A consideration under *Task 5* of a list of topical problems and “bottlenecks” for every object became a logic continuation of Tasks 3 and 4 of the SMP Report. Both the “bottlenecks” and the “problem issues” were identified on basis of integrated analysis of a variety of factors with emphasis on safety factors and the general logic of process technologies.

At that stage of SMP development the identification of problem areas was done through: -expert evaluations by leading experts of Minatom, Russian Academy of Sciences and other agencies and enterprises involved; and -analysis of work progress under main areas and tasks which solution is required to implement the program of complex decommissioning of NS, NPSS, MV and environmental rehabilitation of radiation hazardous coastal facilities.

The main results of the work on identifying the “bottlenecks” in each work area regarding the concerned decommissioning and environmental rehabilitation objects are presented below.

Work area - “*Decommissioning of NS, NPSS and MV*”

Of the main tasks considered in Section 5 of the SMP Report, which solution is required for implementation of this work area, the experts identified the following principal problem issues:

- ensuring safe haulage of NSs from the waterborne storage centers to the dismantling enterprises;
- establishing a coastal Long-term Storage Facility (LSF) for reactor compartment of dismantled NSs and NPSSs;
- solutions to the issues of management and reprocessing of SNF from LMC reactors of *Alpha* class NS;
- construction of a modern high-efficiency complex for reprocessing, conditioning and long-term storage of SRW;
- solution of the issues of management and complex decommissioning of damaged FSV “*Lepse*”;
- ultimate solution to management of damaged and non-reprocessible (uranium-zirconium) fuel stored on MV pertaining to Murmansk Shipping Company; and
- solutions to safe management and disposal of toxic waste.

Work area - “*Environmental rehabilitation of radiation-hazardous facilities in Andreeva Bay and Gremikha*”

Under this work area the most important problem issues requiring immediate strategic decision-making are due to the lack of:

- appropriate equipment to ensure safety of personnel at the facilities involved into the related works;
- confident data on amount, type and condition of SNF and RW stored at the said objects;
- conceptual solutions to management of SNF, including damaged fuel stored at these facilities;
- an accepted ultimate solution regarding management of *Alpha* class NS reactor fuel;
- a classification of “very-low-active SRW” stipulated in the Russian legal, regulatory and technical documents;
- clearly formulated and justified criteria to assess the ultimate state of CMB buildings and sites after the rehabilitation is completed;

- functional requirements and acceptability criteria for ultimate disposal of RW;
- modern, high-efficiency center for reprocessing, conditioning and storage of SRW;
- approved concept and taken decisions on selection of a type and location, conducting of necessary research, development and design works to support establishment of a regional repository for ultimate disposal of low- and medium-level RW, as well as of a storage facility for high-level RW; and
- physical protection of objects in compliance with the present-day requirements.

Work area – *Ensuring extrinsic safety during complex decommissioning and environmental rehabilitation of radiation-hazardous facilities*

Being of general-system nature, the problems of ensuring extrinsic safety concern all work areas, decommissioning objects and tasks. It is obvious that ensuring reliable safety of all-types is an indispensable condition when addressing any works with radiation-hazardous objects. Solution of safety problems is considered as the top-priority measures and should begin already at the initial phase of deploying large-scale works on complex decommissioning and environmental rehabilitation.

In this area the following tasks should be considered as the most topical ones:

- ensuring safe working conditions of personnel during complex decommissioning and rehabilitation of radiation-hazardous facilities;
- ensuring physical protection of valuable materials and radiation materials; and
- conducting radiation monitoring of the environment.

Thus, when addressing Task 5, more than one hundred of different measures focused on solution of major decommissioning tasks were examined, of which a limited list of the most urgent measures was drawn up.

Considering a large scale and complexity of the problems related to nuclear fleet complex decommissioning as well as time constraints to resolve these tasks, the identification of the most priority areas - where the available funding and industrial capacities should be focused on first of all - acquires the paramount importance.

The following circumstances complicate obtaining a sufficiently justified and confident answer to this question:

- a great number and variety of objects of analysis: one has to do with nuclear submarines, nuclear-powered surface ships, coastal maintenance bases, reactor compartment units, maintenance vessels, industrial enterprises involved into decommissioning and transport utilities; and
- many various-in-nature determining factors which must be taken into account when developing recommendations. Importance of these factors and their effects on the justification of priorities also continuously changes over time.

The most important factors are illustrated in Fig. 4.

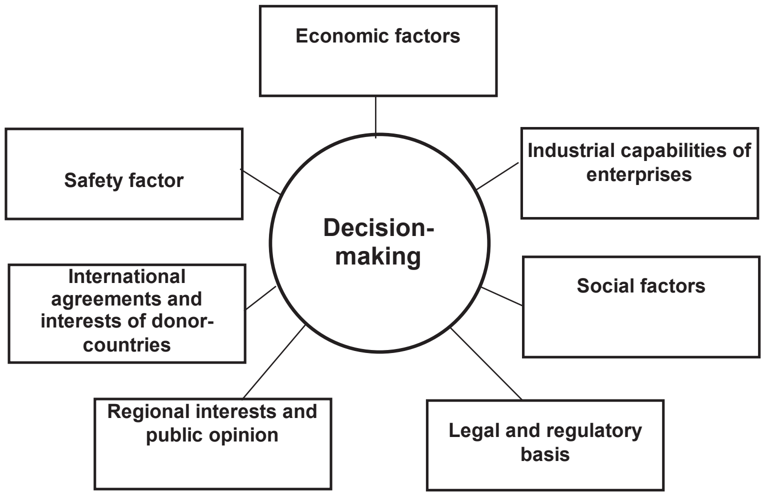

Figure 4. Main factors influencing decision-making while identifying NS complex decommissioning priorities

The justification of priority objects and works on NS complex decommissioning should be unconditionally based on a due regard for the integrity of determining factors. However different nature and sophisticated interfaces between these factors do not allow even approaching a possibility of their simultaneous consideration under a single analytical approach.

Considering the variety of tasks to be solved during complex decommissioning, at the initial stage structuring of these tasks with respect to the objects and measures associated with their implementation was very important.

Work lines (type of activities) were selected as the upper classification level. The following (below) classification level comprises the ***objects*** of complex decommissioning and environmental rehabilitation, such as: NS, NPSS, MV, CMB and RC units.

It is worthy of notice that RC units were distinguished as individual decommissioning objects only due to the specificity of the Russian decommissioning technologies accepted at the initial stage of the related works: for lack of LSFs three and multi-compartment reactor units have been stored afloat.

In the course of works each of the decommissioning objects can be a source of SNF, SRW, LRW, noxious chemicals or other non-radioactive waste reprocessible and re-usable in industry. Still, the main feature of the complex decommissioning objects is the presence of SNF and generation of RW during work execution. Considering a special importance of SNF, SRW and LRW for the work to be done and for justification of priorities, at some work phase they can be attributed to the category of independent objects of management. This is a quite natural decision because in the course of the process-and-management cycle SNF and RW loss their pertinence to one or other decommissioning object and quantitatively merge with other similar materials and thus are considered in most cases in aggregate.

At the same time SNF, SRW and LRW, as independent subjects of management, have one common feature. They cannot be subjects of safety analysis without consideration of their locations.

The next - in terms of scale - level is the ***tasks,*** which represent an integrity of different interrelated, sufficiently large-scale measures which implementation allows achieving ultimate goals regarding each object.

And, finally, the last classification level includes the lists of specific ***measures*** related to the objects and tasks and aimed at task solving.

In turn, a measure can become a basis for one or several ***projects*** when many economic and organizational issues related to their implementation are agreed upon with an investor, state coordinator of work and executors.

When justifying the methodology for identification of priority tasks, the following two extreme approaches were considered:

- a quantitative approach: account of the integrity (or majority) of determining factors in frames of an analytical multi-factor analysis with application of up-to-date mathematical methods of analysis of operations, and
- a qualitative approach: account of opinions of leading experts representing agencies, industry and science.

Complex nature of the set tasks predetermined inevitable use of multi-factor analysis. The issue was to what extent the combination of many determining factors should be accounted for at a specific stage of the studies where sufficiently rigorous and confident analytical approaches could be used.

At first glance, the first approach, which appeared attractive due to its seeming rigorousness, in fact could not provide for any confident results due to extreme complexity of the set tasks, as noted above.

The other extreme approach also appeared unacceptable since it did not allow progressing further to achieve more depth and rigorousness in justifying priorities than it is done at present at practical managerial level.

Therefore, the authors have adopted a compromise option where opinions of experts and positions of agencies are combined with the use of confident quantitative methods comprehensively tested in nuclear power industry, widely acknowledged and sufficiently reliable.

The general logic and sequence of the priority-setting methodology used in the work is illustrated in Fig. 5. A specific feature of the accepted approach is that the ranking is done at all classification levels from top to bottom: objects, tasks and measures. The justification of priorities at each of these levels is carried out using specific methodological approaches complying with the specificity of the relevant levels.

A justification and brief description of these approaches at each classification level, as well as interim and final results of the performed analysis, are given below.

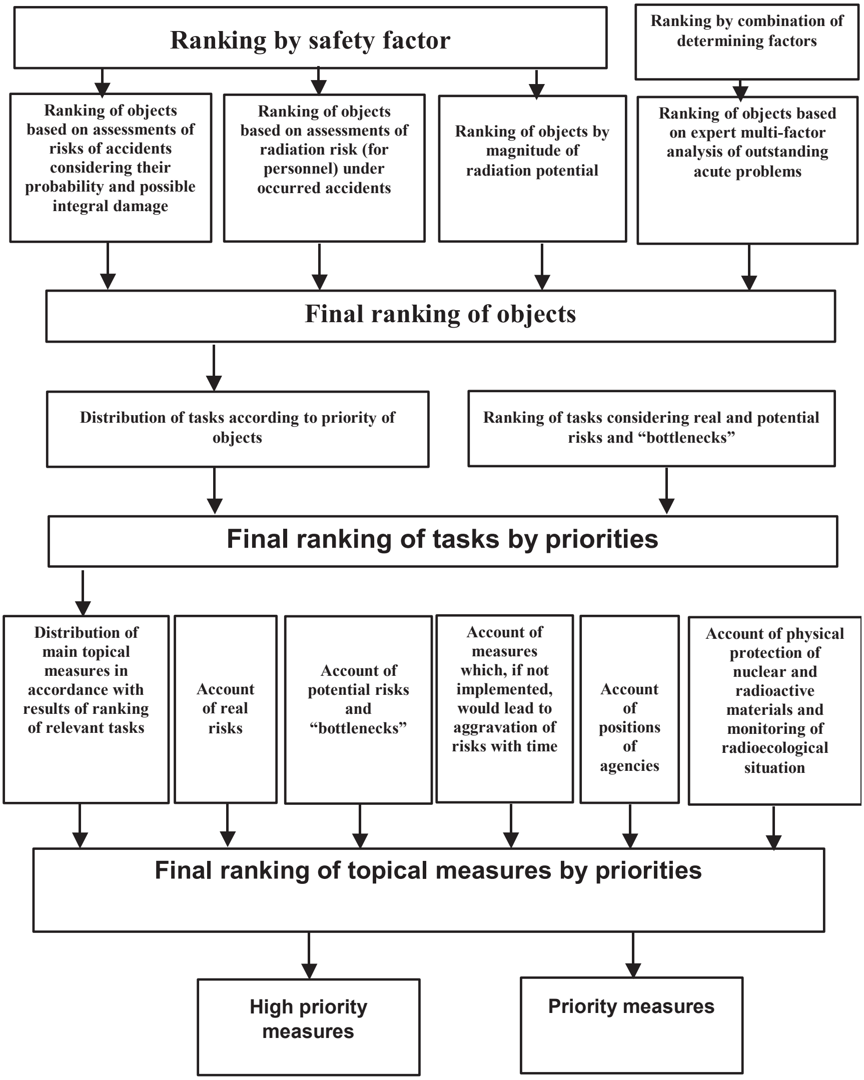

Figure 5. Logic and sequence of stages of the accepted procedure for justification of priorities

The first procedural priority-justification stage was the ***ranking of decommissioning and environmental rehabilitation objects***, i.e. identification of objects to be covered first by a set of required works. While ranking, a comparative analysis was done in four independent areas three of which were related to assessment of their safety. The forth area represented an analysis performed with consideration for the integrity of determining factors at the expert level (Fig. 6).

Attributing the outmost importance to the safety factor was justified as regards nuclear and radiation hazardous objects: all objects of the decommissioning and environmental rehabilitation pertained to that category. The safety criterion was also highlighted as the main factor for justification of priorities in the EBRD Terms of Reference for the SMP development.

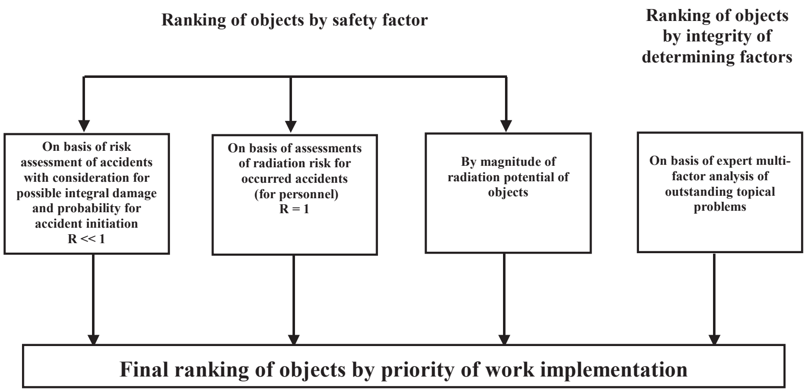

Figure 6. Main areas of analysis while justifying the objects' priority

For these reasons, to obtain more reliable end results, the ranking of objects by the safety factor was done using three different procedures. In the first case safety was assessed by magnitude of the integral damage associated with accident consequences with consideration for the probability of initiating emergency situations. In the second case the raking was done on basis of assessment of radiation risk of already occurred accidents. And lastly, the object safety assessment and their ranking were performed by the magnitude of radiation potential accumulated at the objects. It should be noted that at the object ranking stage only potential risks were considered. Account of real risks as well as of other safety aspects was performed at the subsequent stages of ranking of tasks and measures.

Object ranking by magnitude of integral risk

Ranking of objects within the first area was limited to determining the risk for each object as a sum of probabilities of the most significant emergency events multiplied by the magnitude of integral damage caused by the consequences of such accidents.

Based on systematization of information on the decommissioning and environmental rehabilitation objects, a list of the most significant accidents was drawn up for each type of objects. Accidents with insignificant consequences (for example, spillage of LRW) and low probability events were not considered at all.

The assessment of integral damage caused by implications of individual accidents was compared with some standard levels determined by the International Nuclear Event Scale (INES). The INES scale, supplemented by expert assessments of economic damages, as applied to typical emergencies with decommissioning objects, is illustrated in Table 1.

TABLE 1. Scale of correspondence between severity of implications of possible accidents and predicted property damage in NS decommissioning processes

Consequence severity, levels	Major accident	Serious accident	Accident with off-site risk	Accident without significant off-site risk	Serious incident	Incident	Anomaly
Levels	7	6	5	4	3	2	1
Predicted damage, US$	$>10^{10}$	$>10^{9}$	$>10^{8}$	$>10^{7}$	$>10^{5}$	$>10^{4}$	$>10^{3}$

Note: Level 7 – close correspondence to Chernobyl accident;
Level 5 – NS accident in Chazhma Bay;
Level 4 – sinking of *K-159* NS with SNF, depth >200 m;
Level 2 - sinking of defueled *B-313* NS near a pier in Kamchatka, depth <30 m.

The magnitudes of integral damages of Table 1 can be considered sufficiently confident since real data available for four of seven levels were obtained from experience in elimination of consequences of accident corresponding to those levels.

The probabilities of events were determined with due regard for available statistical data on the analyzed objects. In cases that they were insufficient, the data obtained from other similar objects pertaining to other industry and technology areas were used.

Ranking of objects by magnitude of radiation risk in case of emergency

Ranking of objects within the second area also started from listing of the emergency situaitons for each type of objects. Radiation consequences of accidents expressed in a number of additional fatalities per year per one million of people were determined under the assumption that the event had occurred (R=1). The risks were calculated using the results obtained in Chapter 4 of the SMP Report. In turn, those data were obtained by numerical modeling of emergency situations and, thus, represented calculated values. The results of ranking of objects with consideration for radiation risks are demonstrated in Fig. 8.

Analysis in this area gives more conservative assessment of safety than the results obtained through the analysis under the first area: in this case the probability of emergency events is assumed equal to 1. However under such approach the obtained numerical estimates are more reliable because the analysis excludes estimates of the probabilities of accident occurrence that is always a rather “vulnerable” point.

Priority ranking by magnitude of radiation potential

At first glance, the use of magnitude of radiation potential without consideration for keeping options and technical condition of nuclear and radiation materials, may appear unconvincing; moreover, to a certain degree this magnitude was taken into account in safety assessment procedures by integral risk and radiation risk. However the use of such approach, along with other ones, was justified because direct use of the radiation potential magnitude provided for more conservative safety assessment as compared to the assessments obtained through two first analysis areas.

It would be quite appropriate to draw here an analogy with the generally accepted approach to safety assessment of nuclear power plants where, along with the probabilistic safety analysis, each NPP is calculated for the maximum possible (i.e. beyond the design basis) accident in order to obtain as conservative assessment as possible.

Fig. 7 shows data on radiation potential of decommissioning objects in the North-Western Russia. CMB potential is represented as a summation of potentials of adjacent storage facilities. Such approach was justified due to virtually lacking SNF safety barriers and inevitable involvement of all CMB site and accumulated therein nuclear and radiation materials into any emergency.

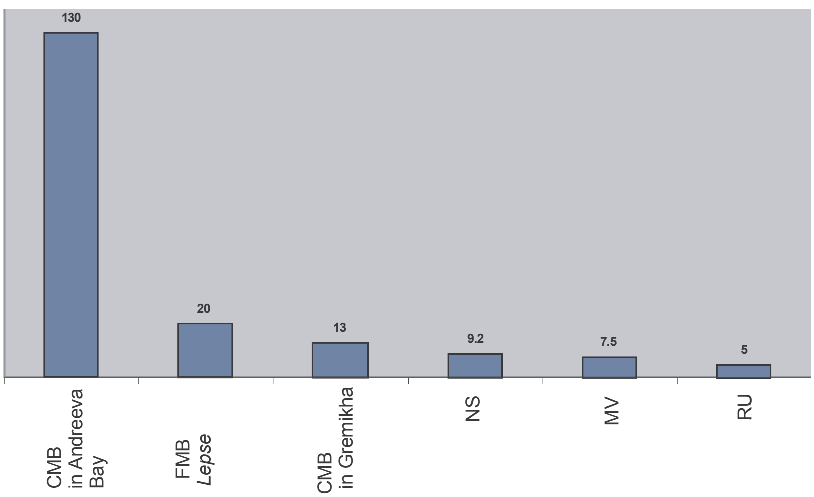

Figure 7. Radiation potential of SNF concentrated at different objects in the North-Western Russia (10^{15}Bq)

At the same time the potential of NS, FSV and RC units was determined by potential of each of similar objects because even in case of neighboring NSs a simultaneous release of radioactivity (and, what is more, of fission products) from the primary circuit under external impacts would be unlikely owing to several SNF safety barriers at each NS. Moreover, NS and MV waterborne storage locations are spread over different piers and bases. Magnitudes of radiation potentials of NS, CMB and RC units (see Fig. 7) concern those specific objects where maximum radiation potential has been accumulated.

It should be emphasized again that CMB situation was considered in all cases under the assumption that the whole of SNF was stored at a single storage facility. The following provisions justified that assumption:

1. Analysis of risks and accounting of safety factor were done under the conservative approach typical of nuclear industry facilities.
2. SNF storage facilities at CMB are located relatively close to each other and have common infrastructure.
3. SNF in storage facilities has less number of safety barriers as compared to SNF in NS reactors. In case of external impacts to one storage facility the neighboring ones will be also involved into the emergency situation.

Ranking of objects on basis of expert multi-factor analysis of outstanding topical issues

To take into account the integrity of main determining factors, ranking of objects by safety factor was supplemented by that performed on basis of a multi-factor analysis. Considering the complexity of taking into consideration a combination of many determining factors in frames of the analytical approach, it was considered reasonable to carry out such analysis at the expert level. Ten problem areas were selected for the expert assessment, and three ways of filling out a table were suggested to the experts to assess a degree of topicality of outstanding problems:

– urgent problems requiring first-priority decisions;
– pressing problems which impede implementation of certain operations at different stages of complex decommissioning or can create problems in near future;
– presence of less-pressing problems or their absence.

In a point of fact, that approach represented a generalized expert assessment of problem urgency with consideration for a rather broad combination of main determining factors.

A special table (see Table 2) was proposed for filling out to a number of leading specialists in this field from different agencies, research institutes and enterprises. Data given in Table 2 represent the result of averaging of individual assessments performed in accordance with the techniques adopted for the expert analysis.

TABLE 2. Ranking of objects by combination of determining factors

Problems to be solved / Objects	State of the input data base	Safety assurance	Physical protection	SNF unloading	SNF management	Dismantlement	Management of toxic waste	Management of RW	Environmental rehabilitation	Priority
NS		red		yellow	yellow		yellow			
NPSS				yellow	yellow	yellow	yellow			yellow
MV		yellow				red	yellow	red		yellow
RC unit		yellow				yellow		yellow		yellow
***Alpha*-class NS**		red		red	red		yellow	yellow		yellow
Andreeva Bay CMB	red	red	yellow	red	red			red	red	red
Gremikha CMB	red	red	yellow	red	red			red	red	red

red - urgent problems characterized by one or several "bottlenecks";

yellow - pressing problems which impede implementation of certain operations at different complex decommissioning stages or can create problems in near future;

white - presence or absence of less-pressing problems.

The fact that the Table contains *Alpha* class NS in the "object list" which was not presented in safety-factor-related ranking is explained by the pressing need to ensure nuclear safety of Spent Removable Sections (SRSs) of such objects.

The results of ranking of objects by all four selected areas are given in Fig. 8.

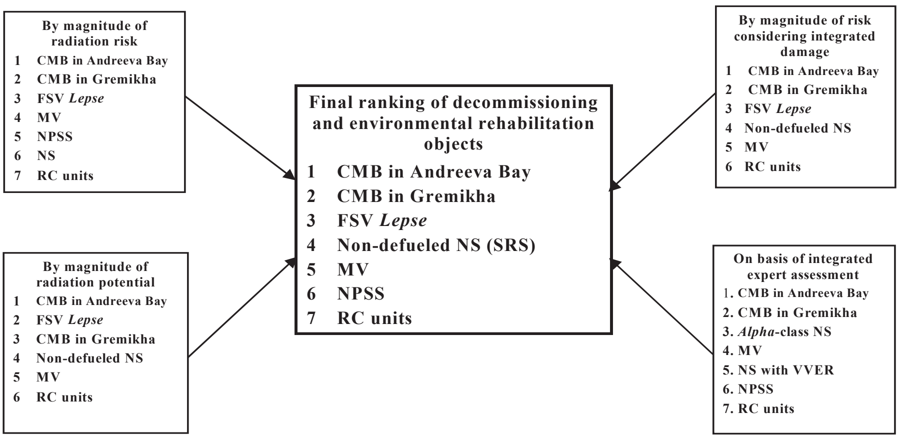

Figure 8. Ranking of objects by safety factor based on integral expert assessment of outstanding problems, and final ranking of objects

Final results of ranking of the decommissioning and environmental rehabilitation objects are also presented in Fig. 8. In doing so, it was taken into account that SRSs of *Alpha*-class NS are/will be stored at CMB in Gremikha.

The results given in Fig. 8 show that the problems related to rehabilitation of the former naval CMBs in Andreeva Bay and Gremikha should be considered as the top-priority challenges. This result may be considered as confident and stable since it is proved by analyses under all four independent areas.

The obtained results cannot be considered as trivial since in Minatom's conceptual documents as well as in other materials developed 2-3 years ago the main priority stated was NS dismantlement, whereas their defueling was set as the most urgent task. Such conclusions were previously made due to lack of information on both technical condition of the objects and radiation situation at CMB at that time.

The ranking of priorities by objects of decommissioning and environmental rehabilitation justified in the performed analysis does not set a fixed sequence of work implementation. The sequence of implementation of specific measures and projects should be determined with due regard for many other considerations and factors. At the same time, the obtained results should be viewed as strategic guidelines of principle to be taken into account when ranking tasks and specific measures in the areas where the selection made on basis of object ranking does not contradict the logic of process operations and other important factors which were not considered in full measure when performing object ranking.

Thus ranking of decommissioning and environmental rehabilitation objects represents an important methodological stage, which fits in the further priority-setting process of tasks and specific measures rather than a stand-alone meaningful item.

In accordance with the adopted list of decommissioning (environmental rehabilitation) objects, a list of tasks which fulfillment assures achievement of the end goal was determined for each object.

Prioritization of execution of individual tasks was justified at the second investigation phase. It would have been quite natural to position those tasks as it came out of the ranking of their related objects. That was exactly done as the first step to justify priorities. Note that the task ranking was done for groups pertaining to different objects. To rank tasks inside such groups the results of expert analysis of risks and “bottlenecks” performed by specialists of Russian Academy of Sciences, Rosatom, Russian Navy, RRC “Kurchatov Institute” and other agencies and research institutions were used.

The results of task ranking performed in such a way are presented below:

- SNF management at CMB in Andreeva Bay.
- SNF management at CMB in Gremikha (SRS at CMB, NSs and RC units).
- Physical protection of CMB in Andreeva Bay.
- Physical protection of CMB in Gremikha.
- Radiation monitoring in the North-Western Russia.
- NS transportation to dismantling centers.
- Management of SNF at FSV *Lepse* and preparing for storage.
- SNF management on both NSs with VVER and NPSSs.
- Ensuring safe working conditions for personnel.
- RW management in Andreeva Bay.
- RW management in Gremikha.
- Making up of RC units and their storage at Temporary Storage Facilities (TSF).
- Construction of long-term storage facilities, RC unit placing to LSF and storage.
- Preparing MV for waterborne storage.
- Rehabilitation of buildings, structures, territories and water areas in Andreeva Bay.
- Rehabilitation of buildings, structures, territories and water areas in Gremikha.
- Management of noxious chemical substances.
- MV dismantlement.
- NPSS dismantlement.
- Management of RW from MV.

At the following work stage the most acute measures were to be selected from the general broad list of measures oriented on solution of the above tasks. In other words,

ranking of measures was to be justified by their topicality with consideration for the integrity of determining factors, such as:

- elimination of "bottlenecks" identified on basis of detailed analysis of process and management cycles with the objects under consideration; and
- elimination of most important nuclear and radioecological risks.

The importance of real risks and potential risks was considered at the quantitative level. Potential risks (both invariable and increasing-over-time risks) were considered apart.

The assessment was performed via modeling of emergency situations and identifying the related implications for environment, personnel and population.

Consideration of "bottlenecks", unresolved problems, actual status of physical protection and radioecological control at objects, attitude of agencies and some other parameters was done at qualitative expert level. The situations were chosen with due regard for severity of their consequences and the probability of realization.

The use of the primary list of measures and its consistent update, taking into account the above factors, allowed drawing up the final list of ranked measures.

It should be stressed that, considering intricate nature of the studied problem and inevitable approximateness of the adopted methodology, the obtained results regarding priority justification should not be treated too rigorously. To a lesser degree this statement is true for ranking of the decommissioning and environmental rehabilitation objects because at their justification stage, along with expert assessments, sufficiently tested analytical approaches were used. As regards ranking of tasks and key measures, here the degree of approximation (i.e., uncertainty of the results) is somewhat higher.

In this connection, the ranking of measures obtained during the analysis should be considered as tentative or "soft" one that is quite natural because the multi-faceted problem of such scope and complexity, as the decommissioning of nuclear fleet and rehabilitation of its supporting infrastructure is, cannot have a straightforward and unambiguous solution. That is why we thought it would be appropriate to avoid numbering of measures on the proposed list. Besides, responding to wishes of the work customer (the EBRD) and of the Russian state coordinator and customer of decommissioning activities (Rosatom) regarding compilation of a stand-alone list of first-priority works, all topical measures have been separated into two blocks: ***high priority measures and priority measures***.

While selecting first-priority projects it would be preferable to use the high-priority measures that, however, would not exclude parallel implementation of projects pertaining to the measures of both blocks.

Implementation of specific projects selected with due regard for the list of top-priority measures justified in the work will objectively contribute to elimination of both real and potential sources of environmental hazard related to the decommissioning objects.

GENERAL POLICY AND STRATEGY FOR FRENCH NAVAL SPENT FUEL MANAGEMENT

J. CHENAIS
CEA, France

Since near 50 years, France has developed a nuclear fleet which can be divided in two generations: a first generation of 6 SSBN "Le Redoutable" class and 6 SSN "Rubis" class, and a second generation of SSBN "Le Triomphant" class (2/4 in active service), an aircraft carrier "Charles de Gaulle" and a new program of SSN "Barracuda".

Insofar as the ships of the second generation are put in active service, the older ones are decommissioned and enter the dismantling process, conducted accordingly to the standardized levels of IAEA. Today 4 SSBN are decommissioned with their spent fuel unloaded.

Nuclear fuel can be divided into two main families:

- the metallic fuels (UZr metallic alloy) used in the first cores of SSBN and the first on shore reactor (PAT),
- the oxide fuel, (UO2 with slightly enriched uranium), the active materials being of the same type of civil nuclear power plants ones.

French strategy differs for each family:

dry storage in CASCAD facility on the CEA site of CADARACHE for metallic fuel, interim storage in pool for oxide fuel, and study of different solutions, in tight connection with the French civil solutions, those already used (reprocessing in COGEMA facilities) and other being studied (long term interim storage, final geological storage).

French nuclear fleet

Since near 50 years, France has developed, in total independence, a nuclear fleet.

This development can be divided into two phases:

- a phase 1: building up the fleet from the design studies and construction of the first on-shore reactor (PAT in the CEA site of CADARACHE) and the first generation of SSNB of "Le Redoutable" class" (6 units) to the SSN fleet of "Rubis" class (6 units) after testing a new compact architecture reactor in a second on shore reactor (CAP);
- a phase 2: putting into active service a new generation of submarines more silent and the aircraft carrier "Charles de Gaulle".

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 35–38.

Two of four SSBN “Le Triomphant” class are in active service, the third one “Le Vigilant” , now in sea trials, will reach them, at the end of this year.

SSBN “Le Triomphant” class and CVN “Charles de Gaulle” are equipped with the same type of reactor, tested in the on shore reactor (CAP removed in RNG).

The new generation of SSN the “Barracuda” program is in design phase, the reactor and its core will be tested in the third on shore reactor RES in construction in the site of CADARACHE.

Dismantling policy

Insofar as the ships of the second generation are being built, the older ones are decommissioned and enter the dismantling process, conducted accordingly to the standardized levels of IAEA i.e.:

- level 1: spent fuel and radioactive liquids are removed,
- level 2: all movable equipment is withdrawn, confined part of the plant is sealed and reduced to a minimum,
- level 3: all radioactive material is removed, decontamination is pursued until no further control is necessary.

Coming back to nuclear submarines, level 1 is easily achieved, as it is not very different from the plant situation during ship overhaul or major refits. In particular, the complete fuel unloading operations may have been performed several times during the service live of the submarine. Consequently, these operations are well known and corresponding tools and infrastructures exist.

To achieve level 2, the reactor compartment is separated from the rest of the ship, sealed and stored on a ground facility located inside Cherbourg Naval Dockyard. The rest of the ship is decontaminated, controlled and sent for scrap like any conventional submarine. The reactor compartment will stay in this intermediate storage facility for roughly 15 years; a duration calculated to allow enough time for short lives corrosion products to disappear, and hence reduce the radioactive dose to workers during the next phase.

After the 15 years period, work will be resumed on the reactor compartment in order to achieve level 3. At this time, all remaining pipes, structures, equipments will be cut into pieces, conditioned and sent to ANDRA for definitive storage.

ANDRA is the French national agency qualified for long term storage of radioactive waste.

Today, 4 SSBN are decommissioned, the first one “Le Redoutable” in complete level 2 dismantling two other ready for the ultim operation of level 2, the cutting of reactor compartment.

Spent fuel policy and strategy

Nuclear propulsion fuels can be divided into two large families:

- The metallic fuels are based on a UZr metallic alloy with highly enriched uranium, and were used in the first cores of SSBNs such as “Le Redoutable”, and for the first on-shore reactor at CADARACHE (PAT).
- The oxide fuels use slightly enriched Uranium. Although the technology is different, it uses components similar to those of cores of nuclear power plants. This fissile material is sintered UO_2 in the form of plates covered with zircaloy. All the fuel elements to be treated in the foreseeable future will be of this type.

The strategy regarding the disposal of the spent fuels differs depending on the family they belong to.

No reference process for the reprocessing of metallic fuels exists at present in the French industrial facilities. Moreover, the fact that the quantities are limited has led to the adoption of the solution of dry storage in the CASCAD facility on the CEA site of CADARACHE, which is designed for lifetime of 50 years.

The fuel elements are stored inside sealed metal pits in a storage bunker. This bunker communicates with the atmosphere through an air intake and a stack. The heat produced by the decay of the radioactive matter in the fuel elements heats up the metal wall of the pits by internal convection, conduction and radiation.

The air around the pits warms up through contact with them, becomes lighter and rises. The hot air accumulates under the roof of the bunker and is evacuated by thermo siphon through the stack.

The irradiated fuel elements are placed dry in containers and sealed before they are transferred to the CASCAD facility. These operations are carried out in a hot cell on the CADARACHE site.

All transfers from the port pools to CADARACHE are made in dedicated large-capacity containers, and pose no particular problems.

The feasibility of reprocessing oxide fuels is established due to the fact that the technology is similar to that of nuclear power plant fuels. Technically speaking, oxide fuels can be reprocessed in the industrial facilities of the COGEMA, on condition that specific means are set up for the cutting, handling and transportation of the fuel.

The other methods explored for civilian fuels have also been examined for military fuels. Among these we can mention:

- interim storage in CASCAD-type facilities, as described above;
- long-term interim storage prior to geological storage in subsurface dry storage facilities or in triple-function “transport-interim storage – final storage” containers;
- final storage in deep geological facility.

At the present time, the oxide fuels are stored in pools. A new pool of the RES program, already mentioned will enter in operational service at the beginning of 2005 in the CEA/site of CADARACHE. The new facility and the existing ones in the ports allow the decisions to be postponed beyond 2015, and allow integrating the strategy for the disposal of military fuels into that for civilian fuels, which is to be debated in the French Parliament in 2006. The economic aspects will obviously be taken into consideration.

Conclusion

Spent fuel represents the highest activity of a reactor; consequently the first priority, when a submarine is decommissioned is to unload this irradiated fuel. Then it is easier to manage two separated problems, the dismantling of submarines on the one hand, the conditioning (and possible reprocessing) of spent fuel on the second hand. This is the French policy.

A general policy and strategy for spent fuel must be complete, that means intermediate and long term solutions have to be researched. For interim storage solutions, dry storage facilities, such as CASCAD facility, are more economic than pool storage.

France owns a major nuclear power plant program for electricity production (more than 70% of total production). Both reactors for civil and defence programs are pressurized water reactors, with similar fissile materials.

So French strategy for dismantling process, disposal of radioactive waste coming from these operations, and for spent fuel of naval reactors is, as close as possible, to the civilian strategy; that means using the same facilities (those of ANDRA for disposal, potentially those of COGEMA for fuel reprocessing and other future facilities for interim storage and long term storage).

This decision puts some constraints on the Navy but has two main advantages:

- First, it is cost-effective;
- Secondly, it proves to public opinion that the Navy has nothing to hide as far as protection of environment is concerned.

MANAGEMENT OF SPENT NUCLEAR FUEL IN FINLAND: POLICY, PAST AND PRESENT PRACTICES, PLANS FOR THE FUTURE

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Abstract

In Finland, about 1700 tU of spent nuclear fuel has arisen from the operation of the four nuclear power units which were commissioned in late 1970's - early 1980's. Initially the spent fuel management policy was based on seeking for international centralised options because of the small size of the nuclear energy program. The amendment of the Nuclear Energy Act of 1994, however, revised the policy and disposal of spent fuel into the domestic bedrock is nowadays the only option.

About 330 tU from the Loviisa NPP has been shipped to the Mayak complex in Russia, but that practice was terminated in 1996 due to the legislative amendment referred to above. Nowadays all spent fuel is stored at the NPP sites until it will be disposed of. Only pool type storage technology is used and the operating experiences are good.

Finland has a determined and advanced spent fuel disposal program, which was started more than 20 years ago. A general authorisation, including designation of the disposal site, has been made by the Government and endorsed by the Parliament. In mid-2004, construction of an underground rock characterisation facility, which is intended to constitute a part of the repository, was commenced. The construction licence application for the encapsulation and disposal facility will be submitted in 2012 and the operating licence around 2020.

Though the Finnish fuel cycle policy is currently based on the once-through option, international developments in the fuel cycle technology are followed and regularly assessed, because the long storage period before permanent disposal leaves also other spent fuel management options open.

Keywords

spent nuclear fuel/fuel cycle policy/spent fuel storage/spent fuel disposal

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 39–44.

1. Spent fuel from the nuclear energy programme

Four nuclear power units are currently in operation in Finland: the Loviisa NPP has two 488 MW(e) VVER units and the Olkiluoto NPP has two 840 MW(e) BWR units. These NPP units have been in operation for 23-27 years. The construction of the fifth reactor, EPR 1600 MW(e) to be located at the Olkiluoto site, is scheduled to be started in early 2005.

Spent nuclear fuel from the NPPs is stored on-site in pool-type interim storages. The total amount of stored spent fuel is about 1350 tU. Besides that, about 330 tU of spent fuel was earlier shipped to Soviet Union/Russia.

At the Finnish NPPs, lifetime extension programs are going on and the current estimates for their operational lifetime falls in the range of 50-60 years. Thus, up to about 2700 tU of spent fuel might further be generated by the existing NPPs. The new NPP unit would add about 2500 tU at most to the spent fuel arisings. The total quantity of spent nuclear fuel to be managed in Finland would then amount to 6500 tU at most.

2. Past policies and practices

The decisions on building the current NPPs in Finland were made in late 1960's - early 1970's. At that time, the prospects for nuclear energy were very promising and spent fuel was regarded as an asset due to the worth of its plutonium and uranium as nuclear fuel. Accordingly, the contract for the supply of the Loviisa NPP included clauses for the return of spent fuel to the supplier of the fresh fuel in Soviet Union. Though no such stipulations included in the supply contracts for the Olkiluoto NPP, it was taken obvious that the operator would later make contract with a French or British reprocessing company.

However, the prospects changed in tmid-1970's. The Western reprocessors elevated substantially their prices and adopted a contractual stipulation on the return of reprocessing wastes to the generator of the spent fuel. This implied that commercial reprocessing services were no longer an attractive option for a country with no fuel cycle industry like Finland. Consequently, the licensee of the Olkiluoto NPP, while followed prospects in reprocessing area, opted for extended interim storage of spent fuel and launched preliminary spent fuel disposal studies.

The national spent fuel management policy was formulated by the Governments Decision in Principle of 1983, stating: *In dealing with spent fuel, international central repositories should be made use of where possible because the total amount of spent fuel arising from the operation of domestic nuclear power plants will remain small. The aim continues to be achievement of contractual arrangements through which the reprocessing waste or spent fuel can be transferred and disposed irrecoverably outside the domestic territory. However, in case of spent fuel for which this kind of contractual arrangements are not achieved, the licensees must provide preparedness for carrying out the final disposal in Finland in a safe and environmentally acceptable way.*

The Government Decision established also a schedule for the development of a spent fuel repository: the disposal site should be selected by the year 2000 and the repository should be operation around 2020.

This policy with the primary and secondary goals remained valid until mid-1990's. The licensee of the Loviisa NPP had contractual arrangements for the return of spent fuel and during 1981-1996, about 330 tU of spent fuel was shipped to the Mayak facilities in Southern Urals. The licensee of the Olkiluoto NPP could not find any satisfactory contractual arrangement and strengthened the spent fuel disposal programme e.g. by starting site investigations. The interim storage capacity for spent fuel at Olkiluoto was also extended by building an on-site pool-type facility. The operating experiences of the Finnish wet interim storages for spent fuel have been good.

A new policy was formulated in 1994 by the amendment of the Nuclear Energy Act, stating (note that by definition nuclear waste includes also spent fuel): *Nuclear waste generated in connection with or as a result of the use of nuclear energy in Finland shall be handled, stored and permanently disposed of in Finland. Nuclear waste generated in connection with or as a result of the use of nuclear energy elsewhere than in Finland, shall not be handled, stored or permanently disposed of in Finland.*

One reason for the policy change was that Finland joined the European Union in 1995 and there were concerns that Finland, having advanced nuclear waste disposal programs, might be compelled to accept nuclear waste from other EU countries. Furthermore, the discussions in Russia concerning the policy for reception of foreign spent fuel and the environmental problems around the Mayak facilities affected the formulation of the Finnish policy.

The new policy led to collaboration between the licensees of the Olkiluoto and Loviisa NPPs and in 1995, they founded a joint company, Posiva Oy, to continue the spent fuel disposal program.

3. Spent fuel disposal program

The Finnish spent fuel disposal has so far progressed in accordance with the target schedule established in the policy decision of 1983. A site screening report was published in 1985 and the site investigations started a couple of years later. Six sites have been subject to deep drillings and other surface based investigations, two of them being the NPP sites Olkiluoto and Loviisa. The final choice, involving e.g. environmental impact assessement (EIA) processes, was done between four sites. Of them, Posiva picked in 1999 the Olkiluoto site as the preferred disposal site.

The first authorisation step pursuant to the Finnish nuclear legislation is Government's Decision in Principle (DiP). In the DiP, the political and local acceptance for the nuclear project is requested and it is also crucial to siting the proposed nuclear facility. Posiva submitted its DiP application for building a spent fuel disposal facility at Olkiluoto in 1999. After STUK's positive safety appraisal, the proposed host municipality approved the application and the Finnish Government made the DiP in late 2000. Finally, the Parliament almost unanimously endorsed the DiP half a year later.

The disposal concept is based on cooling of spent fuel bundles for 30-40 years whereafter they are encapsulated into iron-copper canisters. The canisters would be deposited into a network of tunnels, in crystalline bedrock at the depth of 400-700 meters, and isolated from the rock by a layer of bentonite clay. After operational period, all underground spaces would be backfilled and sealed and the above ground buildings demolished. The disposal concept is illustrated in Fig. 1.

The next licensing step, pursuant to the nuclear legislation, is the construction license. According to the decision by the Ministry of Trade and Industry, the licence application should be submitted in 2012 at the latest. After the Government has granted the construction licence, the encapsulation facility and the first compartments of the repository would be constructed. The operating licence process is scheduled to take place around 2020.

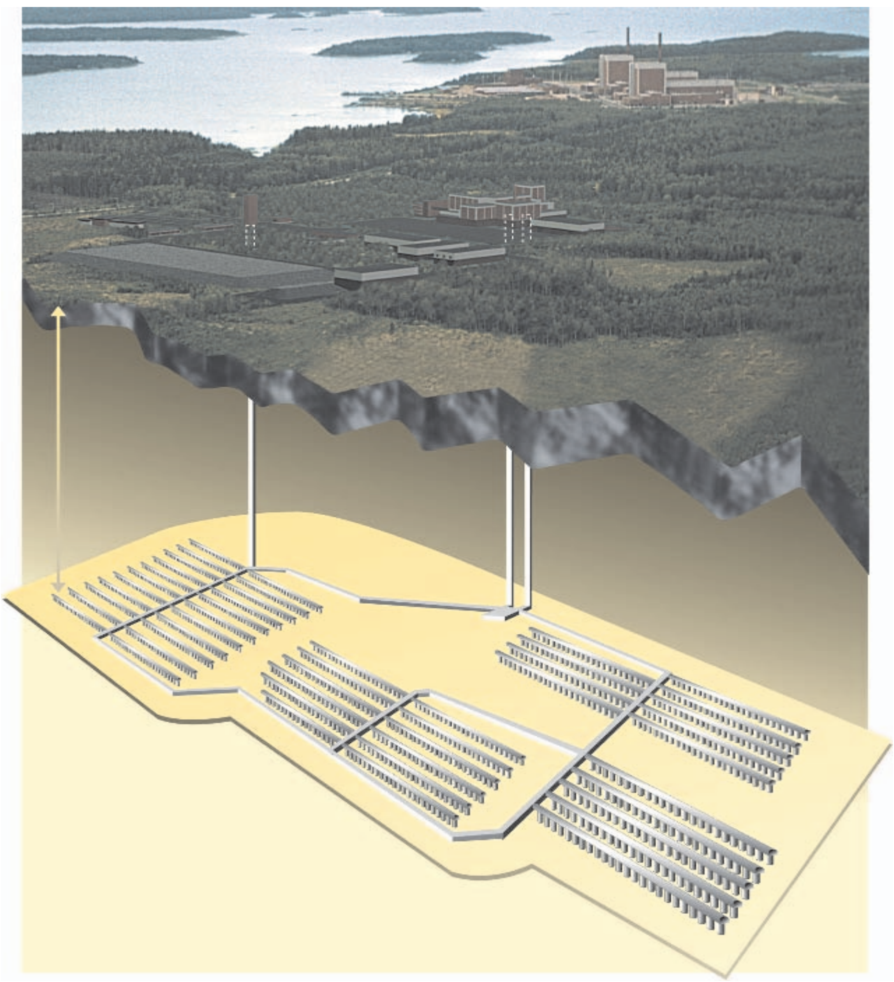

Figure 1. Vision of the encapsulation and disposal facility at the Olkiluto site

Currently Posiva is conducting an extensive research, development and technical design program aiming at gaining preparedness for the submittal of the construction licence application. The program includes site confirmation studies, technical design of the facilities and the engineered barrier system as well as development of safety assessment tools and databases. An underground rock characterisation facility (URCF, see Fig. 2.), the construction of which was started in mid-2004, plays an important role in Posiva's program.

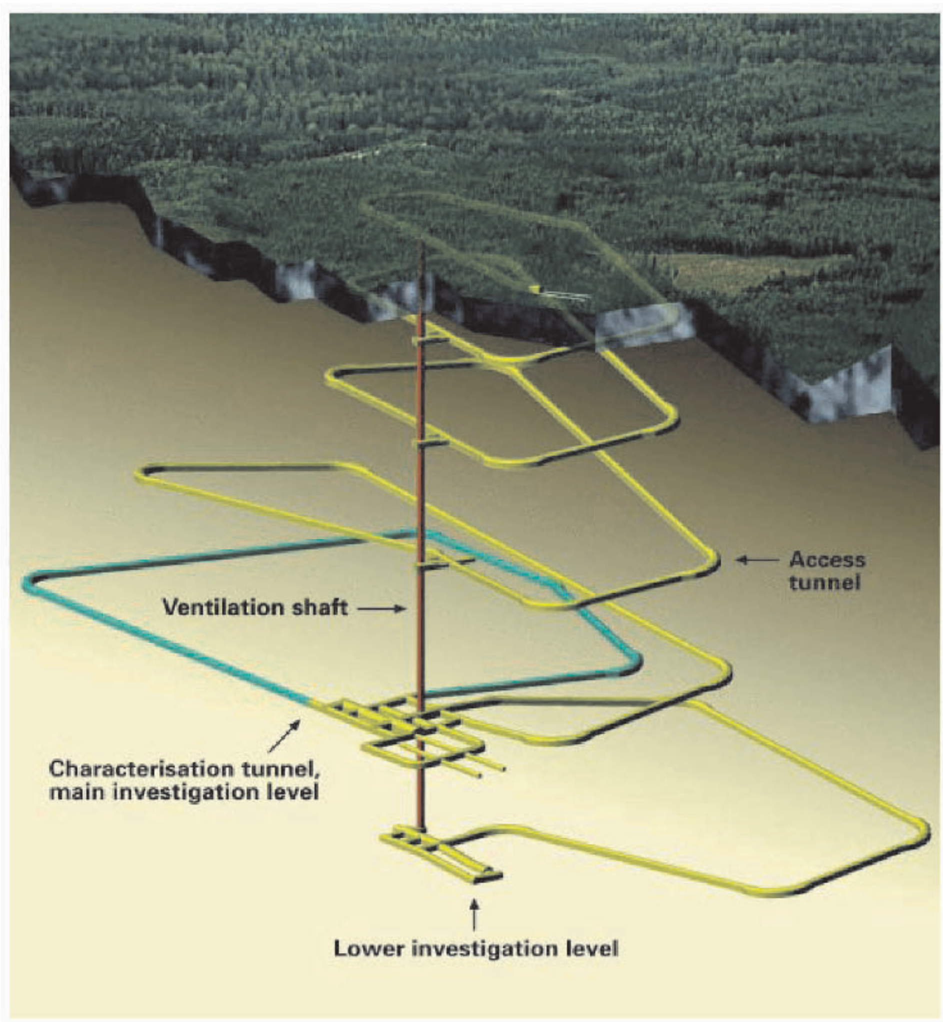

Figure 2. Design of the Underground Rock Characterisation Facility (URCF)

The spent fuel disposal program is subject to regulatory oversight by the Ministry of Trade and Industry and STUK. The main regulatory tools in the current preparatory phase have been mandatory regulations and triennial reviews of implementor's research, development and technical design program. Construction and operation of the URCF, envisaged to constitute a part of the disposal facility, is particularly subject to STUK's inspection and review activities. The implementation of the spent fuel disposal facility and related regulatory control is detailed in Table 1.

TABLE 1. Implementation and regulatory oversight of the spent fuel disposal program

Period	Implementation	Regulation
1983 - 1999	• Technical design • Site characterisation • Research and development	• Government's policy DiP of 1983 • STUK's safety reviews of 1987, 1994 and 1997
1997 - 2001	• EIA program and report • DiP application for a disposal facility at Olkiluoto	• EIA hearings and judgement • STUK's preliminary safety appraisal • Government's DiP and Parliaments endorsement
2000 - 2012	• Site confirmation, incl. URCF • Research, deleopment and design	• Oversight of site confirmation • Triennial reviews of the program
2012 - 2019	• Construction licence application • Construction of the facilities	• Review of licence application • Oversight of construction
2020 -	• Operating licence application • Operation of the facilities	• Review of licence application • Oversight of opearation

4. Prospects for the future

The Finnish spent nuclear fuel management is currently firmly based on the once-through option. Spent fuel is stored in on-site pool-type facilities and enlargement of them is foreseen in early 2010's to cover the required capacity prior to the commencement of disposal operations around 2020. Disposal operations would continue towards the end of century though the first compartments of the repository would be closed and sealed in mid-century.

However, the international developments in the fuel cycle area, such as partitioning and transmutation technology, are followed and regularly assessed in Finland. The long storage period before permanent disposal leaves the various spent fuel management options open. The disposal concept is retrievable, thus recovery of the disposed spent fuel bundles is feasible in case that unforeseen reasons for that emerge in a later phase.

SWEDISH STRATEGY AND EXPERIENCE IN SPENT NUCLEAR FUEL AND RADIOACTIVE WASTE MANAGEMENT

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Introduction

The Swedish nuclear program was initiated already at the end of the 1940:s and had - at that time - both a defence and a civilian side. The original "Swedish Line" for nuclear reactors was to use Swedish natural uranium, existing in low-grade minerals in the middle of Sweden, in heavy water moderated reactors.

The first research reactor, located in a rock cavern in Stockholm at the Royal Technical University, was commissioned in 1954. It operated until 1970 and was eventually dismantled in the 1980-s. The site has been decommissioned to "green field" and the rock cavern is now used for other activities without any radiological restrictions. Several research reactors were also operated in the nuclear national research laboratories in Studsvik. From 1964 to 1974 a heavy water moderated PWR reactor was operated for district heating purposes in a suburb to Stockholm but also generating electricity. It was intended as a demonstration facility. It is now waiting dismantling.

The defence program was terminated after about 10 years.

Today the Swedish programme consists of 11 LWR reactor units at 4 sites generating about half of the Swedish electricity, a fuel fabrication plant, a nuclear research centre and extensive application of nuclear technologies in medicine, research and industry.

In addition, Sweden has a mature strategy for management of all the radioactive waste and spent fuel generated, including processing and facilities for all steps in the management chain except for the conditioning and disposal of the spent fuel and high level waste (Fig. 1).

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 45–56.

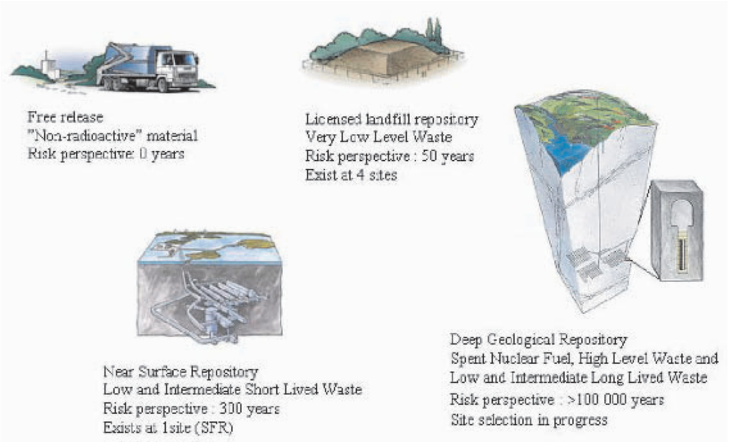

Figure 1. Endpoints for radioactive waste in Sweden

This presentation will give an overview on the development of the strategy, the existing facilities and the remaining challenges with a focus on the waste from the nuclear power plants.

Development of the Swedish strategy

In the very early phase of the Swedish nuclear programme there was no real concern about the spent nuclear fuel and the radioactive waste. The risk associated with ionising radiation was however very well known in Sweden; the initiator of the ICRP (International Commission on Radiological Protection), Prof. Rolf Sievert, was Swedish and very active also in Sweden. Spent nuclear fuel was not considered as waste. It was a resource, which could and should be reused after reprocessing.

There were no specific legal requirements for management of spent nuclear fuel and radioactive waste management in the early nuclear laws and regulations in Sweden.

Also when the relatively large Swedish nuclear power programme was launched in the second half of the 1960s and when the first commercial NPPs were ordered, there were no legal requirements on management of the spent nuclear fuel and radioactive waste.

The early strategy from the reactor owners concerning spent fuel management was to send it abroad for reprocessing and reuse the fissile material in MOX fuel. When the first contract on reprocessing was signed between OKG (the owner of the Oskarshamn NPP) and BNFL, the reactor owner did not have to take back the radioactive waste generated during the reprocessing; only the valuable fissile material should be returned.

The radioactive waste of concern at that time was primarily the high-level waste (HLW) arising from reprocessing. But since this waste was generated abroad, it should also be disposed of abroad. The low- and intermediate-level radioactive waste from the normal operation of the NPPs should, according to the strategy of the reactor owners, be conditioned at the site and placed in interim storage buildings at the site. The authorities (and also the government) accepted this strategy.

In the end of the 1960:s, however, the Swedish politicians got interested in the problems associated with management of the SNF, HLW and other radioactive wastes and in 1972 a Governmental Committee[1] was set up to investigate how the HLW arising from Swedish NPPs should be managed. In 1974 the directive to the Committee was amended to include also proposals to manage the low- and intermediate-level waste originating from the Swedish NPP programme.

Based on the report from the Committee and supplementary discussions and investigations, the government and the industry are since the late 1970:s in agreement on the main strategic approach related to:

1. Management of SNF. Based on a political decision and also for financial reasons, reprocessing is no longer an option in Sweden; the SNF shall be directly disposed of in the Swedish bedrock. Long-lived waste should be disposed of together with the SNF.

2. Low- and intermediate-level short-lived waste. A common disposal site for all low- and intermediate-level waste should be established. It should be located underground, in the bedrock.

3. Very low level waste. At nuclear facilities, mainly NPPs, “landfill-type” repositories can be established for disposal of such waste which decays to level below concern within 50-100 years.

4. Organisation of the waste management. The responsibility for management of the waste stays with the generator of the waste, but a special organisation should be given the task related to all long-term management issues, in particular the R&D work. The authorities should establish special departments for the supervision of waste management issues and the communication with the general public should be strengthened.

5. R&D on waste management. The R&D should be intensified and done in accordance with directives from the nuclear and radiation safety authorities. Every third year an R&D programme should be submitted to the authorities for comments and final endorsement and for approval by the Government.

6. Financing. The NPPs (eventually the electricity consumers) should carry all cost for waste management, disposal and associated research through a special fund that gets its money from a fee on the kWh electricity generated by the NPPs.

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[1] The Committee included representatives of the political parties represented in the Parliament who acted together with technical experts

Implementation of the strategy: Operations Review and Legislation

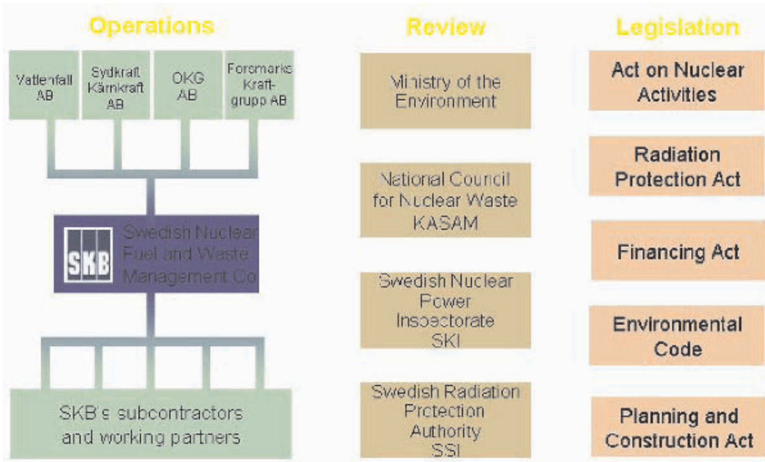

Figure 2. Operations, Review and Legislation

The Government has followed up the strategic decision by establishing the necessary regulatory framework thus giving the authorities the appropriate instrument for supervision and enforcement. The three main acts are the act on Nuclear Activities, the act on Radiation Protection and the act on Financing.

The reactor owners have established the jointly owned stock company Swedish Nuclear Fuel and Waste Management Company, SKB, to dispose of radioactive waste (except for the VLLW which is managed by the reactor owners), manage the SNF outside the NPPs and conduct the necessary R&D work.

Implementation of the strategy: The challenges

In implementing the strategy, there are four basic challenges for society:

The *economical challenge*: To make funds available to cover all costs for research, development, construction and operation for final repositories

The *technical and scientific challenge*: To identify all possible factors to be considered in the safety-, radiation protection- and environmental analyses of the repository.

The *safety challenge*: To ensure that the technical solutions and the methods to develop them have the necessary quality to fulfil the requirements from the industry and the regulators.

The *democratic challenge*: To involve all stakeholders. Here, the involvement of the public is the most difficult one. This is not only a matter of information, it is a matter of communication, education and the decision making process as a whole – basically to develop trust.

Existing facilities and systems

At the NPPs

All the NPPs have their own systems for managing the solid and liquid radioactive waste generated at the site. The very low level waste (VLLW) and the low and intermediate level short lived radioactive waste (L&IL SL) waste is conditioned in accordance with the waste acceptance criteria for the landfill type and the SFR repository respectively. Standard techniques are used processing liquid and solid waste. Cement and bitumen are used as matrix for conditioning.

Figure 3. Landfill repository at Forsmark NPP

Three of the NPPs have licensed landfill-type repositories that can take the major part of the radioactive waste generated during normal operation of the reactor. Since the cost for disposal in the repository is only in the order of 100-200 $/m3, the Swedish society has saved very significant amounts of money with this disposal concept without compromising on safety.

The legislation gives the possibility for clearance (free release) of material when it is demonstrated that the activity on the material is below the clearance level. However, since it often is expensive to demonstrate that a given batch meets the activity levels for clearance it may in many cases be a cheaper (and safer) option to dispose the waste at the landfill repository. Melting of scrap metal is frequently used to ensure compliance with clearance levels.

Transport system

Figure 4. Transport system

All NPPs are located at the Swedish coast, which made a transport system based on sea transport the obvious choice. A special ship of roll-on roll-off type, MS Sigyn, has been built, capable of transporting both the SNF and the radioactive waste. A special terminal vehicle with transport frames are used for the short transport from the NPPs on board the ship. The conditioned waste packages are placed in standardised licensed reusable over-pack during the transport. The use of over-packs is very cost effective since it makes it possible to accept rather high surface doserates on the individual waste packages which significant reduces the disposal volumes.

SFR – Final repository for low and intermediate level short lived radioactive waste

Figure 5. SFR-1 Location at Forsmark

For the disposal of L&IL SL generated at the Swedish NPPs a repository has been established in crystalline rock (as required by the Swedish strategy) 50 m under the seabed outside Forsmark NPP. Following an agreement with the Government and the authorities the repository is also used for radioactive waste from Swedish use of radionuclides in medicine, research and industry that meets the waste acceptance criteria approved by the authorities for the repository.

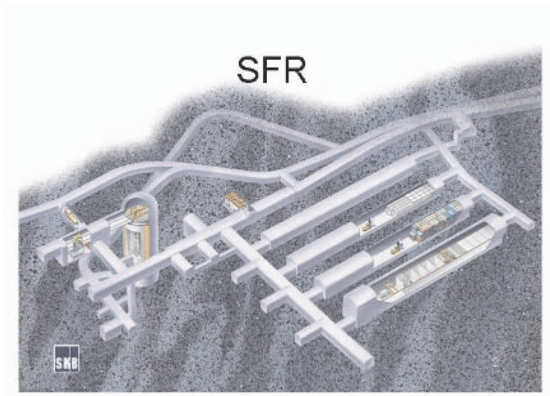

Figure 6. SFR-1 — Layout

The construction work started in 1983 and the repository was taken into operation in 1988. Its capacity is 63 000 m^3. The location under the seabed has a number of advantages, one being the reduced risk for intrusion for example by drilling for water. The repository consists of one silo and four rock caverns with different engineering barriers. The silo has the most advanced system of engineering barriers and is intended for waste packages with the highest activity content. For two of the rock caverns there is no need for engineered barriers to ensure long term safety of the disposed waste.

Annually 1 000 – 2 000 m^3 is disposed of by an operating and maintenance staff of 12 persons. The construction cost was about 100 million USD and operating cost 4 million USD/year. The disposal cost in the different parts of the repository varies between 1 500 and 4 000 USD per m^3.

<u>CLAB – Central interim storage facility for spent nuclear fuel</u>

Figure 7. CLAB

In 1980 the construction started for an underground interim storage for all SNF from the Swedish NPPs. It was taken into operation in 1985. It is a wet storage and is located close to the Oskarshamn NPP.

The first phase consisted of one cavern with a storage capacity of 5 000 tons uranium (with compacted storage), corresponding to 20 000 BWR and 2 500 PWR fuel elements. The construction cost was 250 million USD. Presently a second rock cavern is under commissioning and is expected to be taken into operation by the end of 2004. With the new cavern the capacity will be 8 000 tons.

Annually 100-200 tons of uranium is delivered to CLAB. The running and maintenance cost is about 15 million USD/year and the staff is about 100 persons.

<u>Financing system</u>

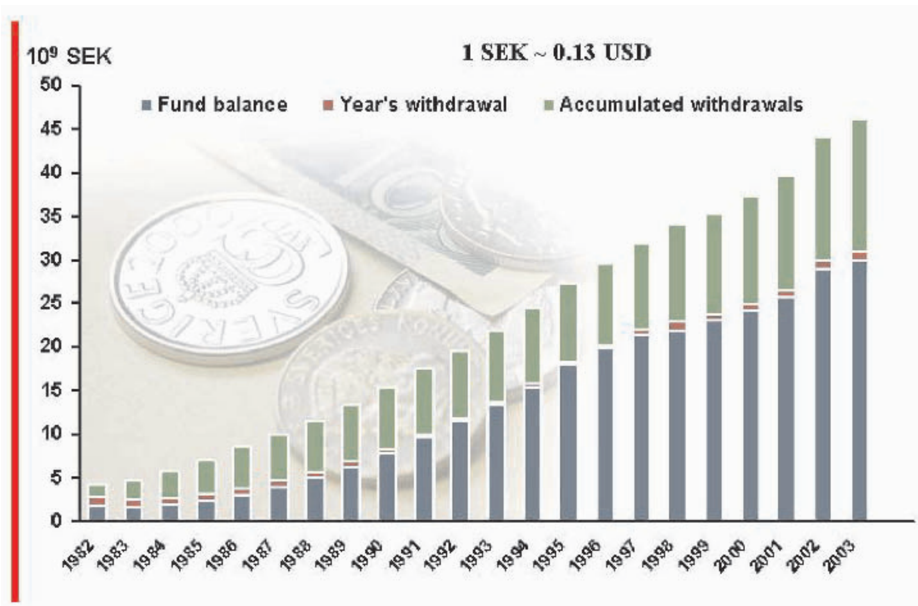

Figure 8. The nuclear waste fund

The financing act stipulates that the reactor owner shall provide money to a nuclear waste fund based on the number of kWh electricity generated at its NPP. The fees are set annually on an individual basis by the Government based on a proposal from SKB and a recommendation from the Nuclear Power Inspectorate. The fund is managed by a special Board and SKB can use money for its activities after approval by the Board.

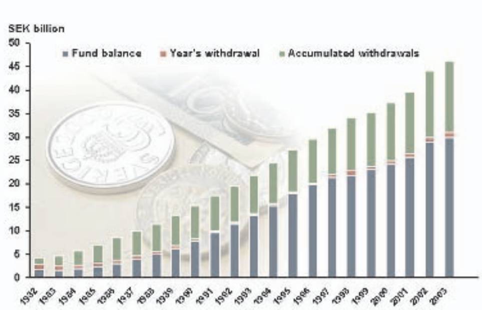

Fig. 9. The nuclear waste fund

At present about 6 billion USD has been transferred to the Fund through fees and financial returns. The total cost for the Swedish programme based on 40 year operation of the remaining reactors is a little more than 9 billion USD. The funds are expected to fully cover this cost.

Ongoing activities

Communication – Public involvement

Figure 10. Public information on board M/S Sigyn

When dealing with waste management issues, especially with the disposal of waste, the most difficult issue is not always to find technical solutions, but to get acceptance for the proposal by the stakeholders; political decision makers, general public, non-governmental organisations etc.. This has for long time been very obvious for the operator and the authorities. Both parties, and especially SKB, have established significant communication programmes to meet the need.

Annually SKB spends approximately 4 million USD on communication activities. This includes written material directed to special target groups, videos, and most important; meetings with people. On a regular basis SKB arrange exhibitions on board Sigyn that is open for the general public and interested groups.

At places where site investigations are taking place, SKB sets up information office to be able to meet the extensive information requirements occurring in such places.

Research and development

According to the law, the waste generators has to conduct the necessary research and development to ensure full understanding of all processes of importance for management of the waste, especially the long term effects of disposal of HLW and SNF. The R&D shall also give all necessary knowledge to design and operate the facilities. Already from the very beginning of the programme, SKB realised that in order to fulfil these requirements, extensive involvement of the international scientific community had to be established. Hundreds of internationally recognised scientists and experts have been and still are involved the SKB R&D programme.

Every three year SKB shall submit an R&D programme for approval by the Government (after endorsement by the authorities). The previous was issued in 2001 and a new programme will be submitted this year.

The main ongoing R&D work is related to encapsulation and disposal of SNF which is further discussed below.

Remaining facilities

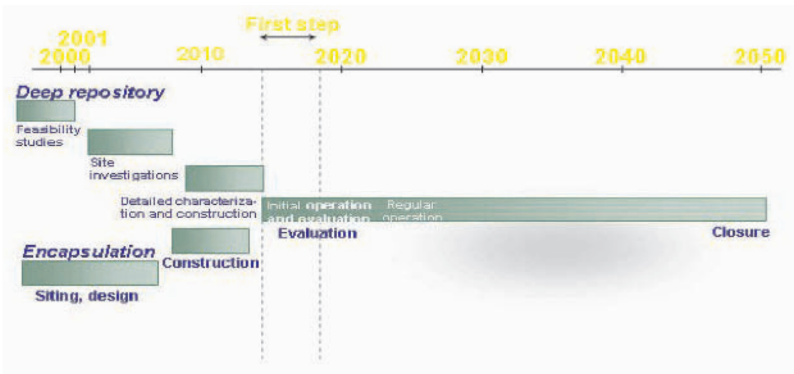

Figure 11. Time plan for remaining activities

The remaining work to be done in order to be able to dispose SNF includes design of an encapsulation facility and a repository and site selection. All three steps are very significant undertakings that have to be well co-ordinated in order to meet the objective to have a repository in operation before the year 2020.

<u>Encapsulation facility for SNF</u>

The program for development of the encapsulation technology comprises of:

– Detailed canister design,
– Manufacturing tests,
– Canister workshop design, sealing and NDT tests,
– Encapsulation process and plant design

One of the most interesting and challenging parts has been the manufacturing and sealing of the copper canisters to be used for encapsulation. A special canister laboratory established for this purpose. Although no decision is made so far on what technology to be used, there is an interesting development of the Friction Stir Welding technique to seal thick copper canisters.

There is today a principle design of an encapsulation facility located adjacent to the CLAB facility. Formal application to establish that facility will be given to the authorities in 2006.

Deep Geological Repository

Before the geological disposal concept was chosen there was a systematic analysis of potential options like sea dumping, sub-seabed disposal, in thick ice sheet, into space, transmutation etc. That analysis has shown that geological disposal is the most suitable solution for Sweden. There is also an international consensus on the acceptability in principle of deep geological repositories for disposal of SNF and high level waste.

In order to further develop all scientific and technical aspects of the disposal of SNF in crystalline rock the Äspö Hard Rock Laboratory was established in the vicinity of Oskarshamn NPP. There is a clear commitment from SKB NOT to use the laboratory for disposal; it will only be used for R&D.

The role of the Äspö HRL is to:

- Provide input for performance assessment
- Develop, test and evaluate methods for investigations, construction and disposal
- Provide experience and training of staff
- Inform of technical and scientific achievement and
- Build confidence

The safety barriers in the Swedish system for the disposed fuel are illustrated in the figure 11.

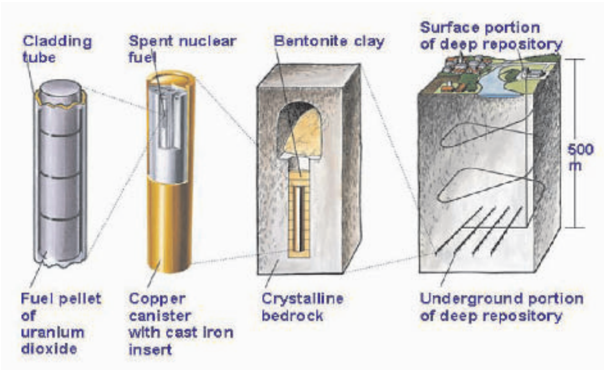

Figure 12. Safety barriers in the Swedish system

The detailed design of the repository is not yet decided, but according to the plans it should be done in 2006 in order to permit an application to be given to the authorities in 2007.

Site selection

The site selection process requires feasibility studies in at least five places and site investigations in at least two places before decision on a site is made. Today the feasibility studies have been concluded and site investigations are going on at two places, one outside Forsmark NPP and one outside Oskarshamn NPP. In addition to the technical team doing the site investigations, SKB has set up information offices at both places to be able to timely and properly respond to information needs in the respective areas. The site investigations will provide the necessary information for the choice of one preferred site before the application will be given to the authorities in 2007.

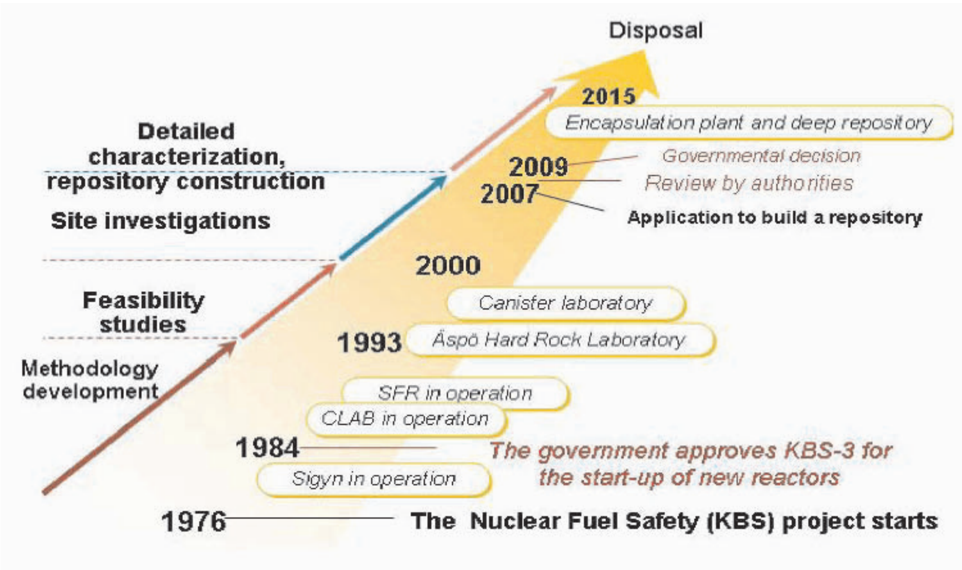

Figure 13. 40 years of development

When eventually the first disposal of the SNF is done, this will mark the end of 40 year of development.

Conclusion

Thanks to a sustainable strategy laid down already in the 1980s, a responsible and knowledgeable industry, international cooperation, competent authorities, a solid finansing system and a clear definition of the roles of the different parties, Sweden has today the privilege of being considered as one of the countries having the most mature back end fuel cycle in the world. An important reason for this good result was that much of the waste management R&D work was done in close co-operation with the international scientific and engineering community. It is natural for all institutions in Sweden dealing with nuclear waste management to continue this international co-operation and share the Swedish experiences with the international community.

Acknowledgement

Thanks to Mr. Bo Nirvin, SKB and Mr. Bo Gustafsson SKB IC who have been most helpful during the preparation of this report and have provided many of the photos and diagrams used.

SPENT FUEL MANAGEMENT IN THE UNITED KINGDOM

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Abstract *Spent fuel management strategy in the UK has developed to meet the needs of the nuclear industry over the last fifty years. The strategy has responded to political and economic influences and advances in technology, engineering and fuel design. This paper describes previous and current facilities which have been constructed to deal with spent nuclear fuel across the UK industry.*

Keywords *Spent fuel, nuclear, storage, strategy, engineering, United Kingdom.*

Introduction

Techniques for the handling and storage of Spent Nuclear Fuel (SNF) in the UK have been developed over a long period. Storage and reprocessing facilities deal with fuel arising from Nuclear Power Plants (NPPs) in the UK and overseas and the UK nuclear submarine fleet.

The Windscale Piles in the UK, built to produce plutonium for defence purposes, represented the first requirement to handle SNF. Fuel cartridges were discharged from the two horizontal reactor piles into a submerged bogie system and transferred to an outdoor storage pool. Although the storage period for this fuel prior to reprocessing was short, the selected engineering arrangements were to influence the approach to fuel storage and handling for the majority of facilities in the UK.

In 1956 Calder Hall, the world's first civil nuclear power plant, came into operation. Calder Hall is a Magnox type reactor using carbon dioxide as coolant in a large graphite core. The term Magnox refers to the fuel pin cladding which is a Magnesium/Aluminium alloy. A further 10 Magnox stations were built. These were followed by 6 Advanced Gas Reactor (AGR) and 1 Pressurised Water Reactor (PWR) using uranium oxide fuel pellets with stainless steel and zirconium cladding respectively. After discharge from

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reactor the majority of UK spent fuel is handled and stored underwater. Only the Magnox station at Wylfa has a dry storage facility.

Degradation of cladding in a water environment means that SNF from the Magnox reactors has been, and will continue to be, reprocessed at Sellafield. AGR fuel can be reprocessed or placed in long term storage.

In addition to indigenous fuel, quantities of SNF have been received at Sellafield from overseas. This includes Magnox fuel from Tokai Mura in Japan and Latina in Italy, and Light Water Reactor (LWR) fuel from Europe and Japan.

Two fast reactors were built at Dounreay in Scotland and they also have fuel storage pools. Some of this fuel has been reprocessed. The UK has 27 nuclear-powered submarines with 16 still in service. The spent submarine fuel is also stored in water filled pools at Sellafield. SNF from the Sizewell B PWR is stored in pools at the reactor site.

Responsibility for commercial reactor fuel storage rests with British Nuclear Fuels plc as the current owner of the Magnox NPPs and Sellafield, and British Energy owner of the AGR NPPs and the Sizewell PWR. The UK Ministry of Defence (MoD) has responsibility for naval fuel.

Criteria for Safe Storage of SNF

For all fuel types the storage regime should satisfy the following criteria:

- Fuel cladding corrosion should be minimised to eliminate, as far as is practicably achievable, radioisotope release to the environment.
- The fuel must be shielded to minimise radiation up-take by plant operators.
- The fission product thermal output of the fuel should be dissipated to prevent excessive fuel temperatures during storage.
- Storage must prevent a critical assembly being formed under any condition.
- Radiation stability of the storage environment should be such that breakdown products can be easily controlled.
- Unauthorised access and movement should be prevented.

Based on these considerations and the experience from operation of the Windscale Piles, storage in deep water filled pools was adopted for the early Magnox fuel. This storage philosophy was also adopted by the British electricity generating utilities for the majority of their commercial nuclear power stations.

Initial Storage at Reactor Sites

Of the 17 gas cooled reactor sites in the UK, 16 were built with pools for SNF storage.

Magnox or AGR fuel elements are stored underwater in metal skips to allow initial cooling and radioactive decay prior to transfer to Sellafield. During this period the fuel cladding material is in contact with the poolwater. For transport the skips are placed in a water filled cuboid transport cask weighing approximately 40 tonnes. This cask provides radiation shielding, cooling and physical protection during the journey by rail to Sellafield.

At the Wylfa Magnox NPP (located on the island of Anglesey in Wales) SNF is stored in a dry vault prior to transport to Sellafield.

Storage at Sellafield

SNF is stored at Sellafield in water filled pools. Water provides radiation shielding and allows remote handling to be carried out from above the water surface plus good visibility.

Magnox fuel was originally stored in open skips in pools with no weather protection. This resulted in chloride contamination of the poolwater by chloride ions entrained in the coastal air. Under these conditions the Magnox cladding materials were susceptible to corrosion but storage was short-term and the fuel was generally reprocessed before penetration of the cladding.

It was recognised that fuel corrosion during prolonged periods of storage could cause operational problems. Consequently water chemistry within the pool was modified to minimise corrosion of the cladding. A reliable long-term storage regime was developed by BNFL that utilised the concept of containerised storage to isolate the fuel from the bulk poolwater in conjunction with optimisation of the poolwater chemistry. Containers which isolate the fuel from the bulk poolwater are now used for all civil fuel types stored at Sellafield.

Fuel Storage Pools at Sellafield

There are several fuel storage pools at Sellafield. The pools are reinforced concrete structures built with an 'above ground' philosophy. Some pool walls are painted and the area above and below the water surface are covered with stainless steel. Pools where casks are received and where fuel containers are opened are generally lined with stainless steel with systems in place to detect any poolwater leakage.

The most recently constructed pools incorporate double containment. SNF is held in small volume metal containers which isolate the water surrounding the fuel from the bulk poolwater which itself provides an additional barrier. As a result it is now not necessary to line the entire pool with stainless steel. Systems are provided to circulate the poolwater through heat exchangers to provide cooling.

Magnox Fuel Storage

Sellafield currently accepts fuel from Magnox power stations in the UK and elsewhere, and Magnox fuel will continue to arrive at Sellafield at least until the closure of Wylfa which is expected to be no later than 2012. Temporary dry storage has been successful at Wylfa NPP, but other stations have continued with pool storage. Dry storage at Sellafield following wet storage at reactor was assessed but was not considered practicable.

Irradiated Magnox and AGR fuel is stored at the power stations in open topped skips and transported to Sellafield in shielded flasks. At Sellafield, in the Fuel Handling Plant (FHP), the fuel is placed into ullaged containers which can be stacked. Based upon an extensive research programme an optimum water chemistry of pH 13 was identified for the containers. Under these conditions storage times of at least five years without cladding penetration are achievable.

Advanced Gas Cooled Reactor Fuel

AGR fuel is transported to Sellafield in skips which are placed inside lidded containers using a dry inlet facility shared with Magnox fuel in the FHP. The container provides criticality control by segregation so boron addition to the water is unnecessary. The design of the lid allows containers to be triple stacked. After a minimum of 180 days cooling the elements are dismantled.

The fuel pins are transferred into slotted cans and the redundant graphite sleeves and additional stainless steel components are stored in drums as Intermediate Level Waste (ILW). This not only converts the fuel into a form suitable for reprocessing but also results in a 3 or 4 fold increase in storage density.

To evaluate the risk of fuel pin cladding corrosion through sensitisation, the condition of AGR fuel elements has been subjected to an extensive monitoring programme to determine the effects of increasing storage time.

From this work a thorough understanding of the pool storage behaviour of AGR fuel has been obtained. This programme identified a corrosion inhibitor. It was found that dosing the water to pH 11.4 using sodium hydroxide, prevented the perforation of AGR fuel pins, which otherwise would have perforated due to the poolwater chloride levels of 2-4 ppm. Condition monitoring has demonstrated that long term storage is feasible.

For AGR storage in the THORP storage pool, which also stores Water Reactor (WR) fuel, it was recognised that, due to considerations of compatibility with Multi Element Bottles (MEB's) used for storing WR fuel, sodium hydroxide dosing could not be used. In this pool ullaged containerised storage has been successfully achieved by the development of catalytic recombiners in the ullage space which prevents the formation of an explosive atmosphere. The containers are filled with high quality demineralised water (-0.1 ppm Cl). It was demonstrated that at these very low chloride levels the lack of hydroxide dosing did not result in fuel pin corrosion.

Water Reactor Fuel Storage

Irradiated uranium dioxide fuel from PWR's and Boiling Water Reactors (BWR's) is currently stored under water at Sellafield prior to reprocessing in THORP.

On discharge from the reactors the fuel is stored for an initial cooling period in a pool at the reactor site. BWR pools are filled with demineralised water, whereas PWR pools are filled with a dilute (typically 0.2M) boric acid solution. The boric acid in PWR pools results from mixing the boronated water in the reactor vessel with the bulk poolwater during refuelling operations. This chemistry results in BWR pools operating at approximately pH 5.8-7 and the PWR pools at pH 4.5-6. Most of these pools operate at, or below, 40°C.

The fuel is transported from the reactor pools to Sellafield in MEBs contained within heavily shielded, high integrity, transport flasks. The MEBs are cylindrical stainless steel vessels containing stainless steel clad "Boral" or boronated stainless steel dividers between the fuel assemblies to prevent criticality. "Boral" consists of boron carbide particles in an aluminium matrix clad with pure aluminium and is widely used as a neutron absorber. MEB's are used to contain mobile contamination from crud or spalling surface layers of the fuel pins.

On arrival at Sellafield, the flask is placed in the pool, opened under water, and the MEB containing the fuel is removed. The flask is then removed from the pool, decontaminated, and returned to service, while the MEB is transferred to a storage frame which supports it vertically during its time in the pool. The fuel storage pools used for the storage of WR fuel contain undosed demineralised water with a purge to maintain low chloride plus sulphate concentrations (<0.5 ppm).

Several benefits accrue by the use of MEB's; including easier fuel handling with less risk of damage to the assemblies and less contamination of the storage pool. This also allows control of the water chemistry around assemblies. Boron inserts allow the close packing of assemblies within the MEB. In some MEB's the 'Boral' is exposed which results in the removal of oxygen, produced by radiolysis of the water, to produce an ullage gas composed mainly of hydrogen. However, the 'Boral' in the remaining MEB's is wholly clad in stainless steel which does not remove oxygen, resulting in an oxygen/hydrogen mixture in the ullage. To prevent explosive mixtures developing the latter MEB's were originally vented to the poolwater. Current practice is to fit sacrificial carbon steel plates which remove free oxygen by oxidation.

Naval Fuel

The UK currently has 27 nuclear-powered submarines. 16 of these are in service, and 11 are "laid-up" (no longer in service). 3 more nuclear-powered submarines (the Astute class) have been ordered, the first of which is currently being built at Barrow-in-Furness, Cumbria. 3 further Astute class submarines are planned, subject to government approval.

During operational life these submarines are subject to periodic maintenance operations which can include refuelling. To undertake these operations, the submarine is taken into dry dock and supported on cradles which provide seismic stability. In addition to the dock gates further protection against flooding is provided by concrete caissons. Before any refuelling operations take place, the primary circuit is decontaminated and connected to shore based services which support the reactor whilst the submarine is being refuelled in dry dock. When the reactor systems of a submarine have been decontaminated, and radiation levels adequately reduced, refuelling may start. A transportable workshop, the reactor "access house", is placed on top of the submarine to allow access to the reactor. The top of the reactor is removed and fuel is taken out one module at a time. All the used fuel handling operations are carried out within heavy shielding to ensure that the refuelling teams are not exposed to high levels of radiation. The fuel is then transferred to the used fuel storage facility to await transfer off-site. Other components from the reactor compartment are removed for servicing in the dockyard workshops. When all the fuel has been removed and the reactor components replaced, new fuel is placed in the reactor, again one module at a time. Once the reactor is fully fuelled it is reassembled, the submarine hull is sealed and the reactor access house removed.

SNF is transferred from the dockyard in a specially designed cask which provides radiation shielding, prevents criticality, absorbs impacts and dissipates heat. This cask is carried on a special rail vehicle to Sellafield.

Initially this fuel was stored in a dedicated facility at Sellafield. As this facility is reaching the end of its economic life MoD undertook a competitive tender process and a contract was let on BNFL on 1st April 1996 for the provision of a Reusable Used Fuel Storage Service (RUFSS). The Invitation to Tender was in the form of a performance specification and potential suppliers were encouraged to consider how best to store the fuel and not to rule out the option of dry storage. From 1st December 2001 the new covered Wet Inlet Facility (WIF) at Sellafield has been available to accept SNF from submarine refits and that stored in the original B27 pool at Sellafield. MoD have a contractual commitment from BNFL that they will store fuel in the WIF for 40 years (ie until 2041) although no difficulties are presently foreseen in extending this period by at least 10 years. The WIF is large enough to accommodate all existing and foreseen SNF. The MoD's reactor programme has directed the development of cores with much extended useful lives thereby considerably reducing the need for storage capacity and the movement of fuel between refit venue and storage facility.

MoD are presently committed to storage in the WIF. When the decision was made to contract for this facility it was not considered economically viable to reprocess naval spent fuel. This is regularly reviewed as the economics of the issue change. The chosen storage technology allows MoD to retain the option of reprocessing. The fuel and its cladding is not degrading, is stored in its designed environment and fuel handling will be no more demanding in 40 years than it is today. If the decision is taken to dispose of the fuel the actual method employed, which could be dry storage, would need to take account of the sensitivity of the design.

Summary

The majority of SNF in the UK is stored in water filled pools. A significant proportion of this fuel has been, or will be, reprocessed. Research has demonstrated that longer term storage of stainless steel or Zircalloy clad fuel is feasible and it is now planned for some fuel from British Energy NPPs to be stored for the foreseeable future.

Naval SNF will be stored in water filled pools with a design life of 40 years which may be extended. No decision has been taken on eventual disposal or reprocessing of this fuel.

There are currently no plans for a disposal facility for spent fuel of any type in the UK.

THE DTI FSU NUCLEAR LEGACY PROGRAMME: SPENT NUCLEAR FUEL MANAGEMENT AT ANDREEVA BAY

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Abstract

Andreeva Bay is located in the Zapadnaya sea inlet at the extreme North-West of the Kola Peninsula (Russian Federation), about 40km from the Norwegian border and 80 km from Murmansk to the south-east.

Spent nuclear fuel (SNF), arising from the operations of the former Soviet Union's Northern Fleet, was initially stored in two large pools within Building 5. However, after serious leaks in the early 1980s the fuel was transferred to an external "drystore" constructed by adapting three existing concrete tanks, previously allocated for the storage of liquid radwaste. This was intended to be a temporary solution to the emergency situation. There are currently approximately 20,000 spent fuel assemblies (SFA) stored within the three tanks. The condition of the dry storage units is poor with inadequate roofs which have allowed water ingress to the tanks. Inspection of the cells has indicated very high activity levels in the interstitial water.

The UK Government's Former Soviet Union (FSU) Nuclear Legacy Programme, managed by the Department of Trade and Industry, has been supporting a project concerned with SNF management at Andreeva Bay since 2002. RWE NUKEM is the Programme Management Consultant to the DTI for this, and other projects, under this Programme. The Programme forms part of the UK's contribution to the G8 Global Partnership Initiative.

The progress of the project is described in this paper. The underlying objective of the project is to identify and implement solutions for existing safety, security and environmental problems of SNF storage at Andreeva Bay, which are acceptable to both the UK and all key Russian stakeholders.

Background

The Andreeva Bay Coastal Technical Base was established in the early 1960's and was used for the refueling of nuclear powered submarine cores and for storing spent nuclear

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fuel from submarines and nuclear powered ice-breakers. The Base was also used for interim storage of the solid and liquid radioactive wastes resulting from nuclear submarine operations and maintenance.

Andreeva Bay is located in the Zapadnaya sea inlet at the extreme North-West of the Kola Peninsula (Russian Federation), about 40km from the Norwegian border and some 80km from Murmansk to the south-east.

Spent nuclear fuel (SNF) was initially stored in two large pools within Building 5, however in the early 1980's, after serious leaks from the pools, the fuel was transferred to an external "drystore" constructed by adapting three existing concrete tanks, previously allocated for the storage of liquid radwaste. There are currently approximately 20,000 spent fuel assemblies (SFA) stored within the three tanks.

The site currently contains very large inventories of radioactive waste. This is principally present as spent fuel, in the dry storage units (Tanks 2A, 2B and 3A, Figures 1 and 2). The potential inventory in the three dry storage units is of the order of 10^{17} Bq. Another very contaminated facility is Building 5, the former pond storage facility for spent fuel.

The dry storage units were designed to store the spent fuel for 6 years. The tanks are currently in poor condition and are no longer proof against rain and snowmelt and from ground water penetration. Water is now present in many of the cells and is in contact with the fuel as the water is contaminated. The activity of the water has been observed to be increasing since 1999 suggesting that there is continuing fuel degradation in the tanks. There are no facilities or equipment on the site to allow improved management of this fuel. One of the dry storage units in particular, Tank 3A, is more susceptible to the penetration of rain water and snow melt as it has no cover other than concrete slabs covered with bitumen. The other tanks have roofs that allow some, but not full, protection.

Figure 1. Dry Storage Units 2,3 (facilities 2A and 2B)

Figure 2. Dry Storage Unit 1 (facility 3A)

Until recently there has been very little infrastructure to support operations at Andreeva Bay. None of the existing facilities had been maintained over the last 30 to 40 years and there were no services (electricity, water, roads, health physics, monitoring, decontamination, waste management) at the site. The old pier remains in very poor condition and adjacent areas are very contaminated. The new pier was never completed and is not in an operational condition.

UK Programme

The Department of Trade and Industry (DTI) on behalf of UK Government Departments manage the UK Nuclear Legacy Programme for the Former Soviet Union. DTI conducted a tendering exercise in 2002 to select a Programme Management Consultancy team for this Programme. RWE NUKEM Ltd was successful and the contract commenced in the summer of 2002. The programme is currently addressing Andreeva Bay spent nuclear fuel management, Submarine Decommissioning in North-West Russia, AMEC projects, an interim SNF store at Mayak and fast Reactor decommissioning in Kazakhstan.

The UK DTI project at Andreeva Bay started in August 2002. The underlying objectives of the project are to identify solutions for existing safety, security and environmental problems of SNF storage at Andreeva Bay, which are acceptable to both the DTI and all key Russian stakeholders, including the relevant regulatory bodies.

The process by which projects are established is for the Federal Agency for Atomic Energy (FAAE) of the Russian Federation (Rosatom) to identify both projects and participating institutes. This Project Identification includes a brief description of the objectives, scope of work and outline costs and timescale. This is reviewed by RWE NUKEM who make recommendations to DTI for acceptance or otherwise. The next stage is for the project to be defined in detail. This phase is funded by the DTI and involves a detailed description of the issues, optioneering studies to determine the optimum way of achieving the objectives and preliminary analyses of risk and environmental impact. A detailed cost estimate, work breakdown structure and project programme is also developed at this stage. Relevant regulatory approvals are also obtained.

Once the Project Definition stage is completed and accepted by DTI, the project moves into design and implementation phases.

The FAAE has nominated three institutes to have responsibility for various areas at Andreeva Bay. These are NIKIET, who are addressing spent fuel management at the site, ICES who are responsible for investigations into Building 5 and SevRAO who are the site operators and are specifically tasked with improving the condition of the spent fuel tanks.

Each of these organisations has identified projects that they wanted to undertake in their specific areas of concern In summary the Tasks address the characterisation of the existing conditions at the site and it’s facilities, the development of options to improve the situation and the development of facilities and systems to allow safe working at the site now, and in the future. These projects are now nearing completion of the project definition phase.

The current Tasks are summarized below and progress for each is described in some detail in the following sections.

Task Number	Lead Organisation	Topic
Task 1	ICES	Characterisation of Building 5
Task 2	NIKIET	SNF Management Options Study
Task 3	SevRAO	Establishment of safe conditions for interim storage and management of spent nuclear fuel in the dry storage tanks
Task 4	SevRAO	Radiation Protection
Task 5	SevRAO	Site Surveys
Task 6	ICES	Integrated database
Task 7	SevRAO	Criticality Monitoring

Task 1: Characterisation of Building 5

This Task is concerned with establishing a comprehensive dataset for Building 5 so that both short-term and long-term management plans for the building can be developed by Russia. The Task has addressed the collection of existing data, establishment of a data base, a preliminary survey of Building 5 and an analysis of the data obtained for consistency and completeness.

Currently the work is focusing on defining the requirements of, and methodology for carrying out, a comprehensive survey to complete the data set on the current condition of Building 5. The best means of carrying out the survey has been established through a formal options exercise. ICES has been developing the methodology and planning the arrangements for the survey in more detail. An outline for this is given below:

Pools

There is a need to examine the bottom of the pools which are 6.5 m deep. There are overhanging beams making access difficult, and there are high radiation fields of up to 0.4Sv/hr. ICES are therefore proceeding on the basis of using a self-propelled robot to carry out the survey. The robot would be equipped with high coverage video equipment, a manipulator for recovery of samples / debris, and equipment for measuring γ, β dose rates.

The pool walls will be examined using of the bridge crane. Equipment would be mounted on a table / platform hanging from the crane, the crane would then be used to traverse the equipment along the pool walls.

Main Hall

It is proposed that the survey of the interior of the main hall will be carried out manually using hand held instruments. This will be supplemented using the bridge crane mounted equipment to examine the hall walls up to crane rail height.

The manual measurements will include radiological measurements and structural measurements on the concrete and its reinforcement.

Roof

It is proposed to survey the roof externally by man access. The survey will include visual, sampling, radiation mapping, and opening up the roof in one place for examination of the fabric.

Foundations

The condition of the foundations will be examined by exposing them at six pre-determined locations. The foundations will be photographed and structural tests carried out. Radiological measurements will also be made.

Basement

Survey of the basement is considered important particularly as it contains two columns which support one end of the pools. Simple access to the basement is not possible at present as access via the hall is closed off by plates, and access via the external door is prevented by earth piled against the door. These measures may indicate very high radiation levels within the basement. These may be associated with the basement being used as a dump for the chains used to support the fuel canisters when the pools were in use.

The proposal is to drill or cut a hole in the door and insert an endoscope and radiation detector into the basement to establish the overall situation within it.

It is currently anticipated that the man-entry surveys will take place in late 2004 whilst the robotic surveys will take place in 2005.

Task 2: SNF Management Options

This task is concerned with conducting an options study to determine the optimum SNF management strategy for the site.

The team put together by NIKIET to conduct the optioneering was very broadly based with up 14 separate organisations involved from a range of technical and regulatory backgrounds. The team members were invited to propose options for SNF management, propose the criteria against which these should be assessed and to identify the information they thought necessary in order to make a judgment between options.

The preliminary list of options accepted for evaluation were:

- Remove all SNF from site and send for reprocessing
- Disposal of SNF
- Remove SNF from the site, reprocess where possible, store or dispose of damaged SNF

The preliminary criteria to evaluate the options were:

- Cost
- Environmental Impact
- Dose
- Timescale
- Risk
- Socio-political events
- Technical flexibility
- Secondary waste generation
- National strategy

The conclusion was that the best option for SNF management was to remove all the fuel from the site for reprocessing at Mayak. There will also be continued consideration of the option of removing the intact fuel, sending it for reprocessing and storage of damaged fuel (prior to reprocessing).

The optioneering has continued with a detailed consideration of how the SNF may be retrieved and transported from the site. This has resulted in a final two options being selected for comprehensive analysis.

Option 1

- removal of the canisters with SFA from the DSU cells;
- SFA repacking into new canisters and temporary storage of the canisters in DSU No. 2 (2A) cells,
- loading of the new canisters to the cells of SNF storage facilities at nuclear service and storage vessels Imandra or Lotta of OAO Murmansk Shipping Company;
- transportation of new canisters by nuclear service and storage vessels to FSUE Atomflot;
- transfer of the new canisters from SNF storage facility at nuclear service and storage vessels to transport containers TK-18 at FSUE Atomflot;
- loading of the containers with SNF to containers cars and their transportation to PO Mayak by railway.

Option 2:

- removal of the canisters with SFA from the DSU cells and their transfer to a shielded facility in a SNF handling complex;
- SFA repacking in a special building of the SNF handling complex;
- canisters loading into transport containers TK-18 (TUK-108/1) in a special building of the SNF handling complex;
- temporary storage of containers TK-18 (TUK 108/1) at the storage pad;
- loading of containers TK-18 (TUK 108/1) to a ship (a special container ship, cargo hold of nuclear service and storage vessel Lotta or to the reequipped technical tanker, design 1150 Amur);
- transportation of containers TK-18 (TUK-108/1) to FSUE Atomflot;
- transfer of containers from a ship to container cars for transportation to PO Mayak by railway.

These options are currently being developed in detail and there will be a decision conference held before the end of the year, at which all key stakeholders will participate, which will evaluate these options and select a preferred option to be taken forward.

Task 3: Establishment of Safe Conditions For Interim Storage and Management Of Spent Nuclear Fuel in The Dry Storage Tanks

This Task is concerned with improving the conditions of the SNF tanks both in the short-term and in the long-term. In the short-term a temporary weatherproof cover is to be developed and constructed over Tank 3A. The long-term option will provide facilities for access to and inspection of the fuel and, potentially, retrieval of the fuel in all three storage Tanks (Tanks 2A, 2B and 3A). This Task is clearly closely linked with that of Task 2.

The design for the temporary Tank 3 A cover has been approved and construction is virtually complete. This is essentially a low-pitched steel roof with facilities for ventilation and filtration (Figure 3).

It has been constructed in a clean area adjacent to the SNF area and will be lifted into position using the existing crane. The cover will be installed and commissioned before the end of 2004.

Work is ongoing to produce the required documentation, a Design Assignment, Declaration of Intent (DON) and Feasibility Study (OBIN) for the design and construction of buildings and facilities for the handling of SNF as well as infrastructure required to support all activities. The OBIN will include an Environmental Impact Assessment for Andreeva Bay and a Technical Safety Assessment. Once this documentation has been approved by all of the relevant authorities the project will proceed to a conceptual, then detailed, design phase leading to construction.

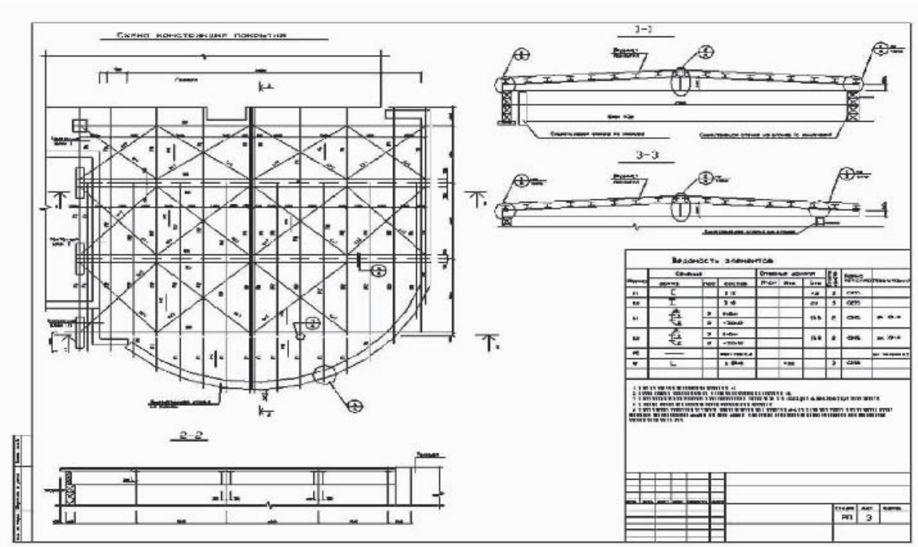

Figure 3. Temporary Cover for Dry Storage Unit 1 (facility 3A)

Task 4: Radiological Protection

The main aim of Task 4 is the establishment of an overall Radiological Management System (RMS) for the Andreeva Bay site including the provision of appropriate radiological protection equipment and facilities.

Progress to date is summarized below:

- Mobile sanitary passes, for 10 persons, have been installed in the SNF area and near B5,
- a vehicle decontamination pad is being built near the SNF area,
- a ventilation system is being installed in the Norwegian village so that the laboratory facilities there can be used,
- materials/equipment for secondary SRW and LRW management are being purchased
- an engineering survey of B50 with a view to evaluating whether this building is capable of long term use (with renovation) as a radiation protection control station.

Future work will address the establishment of an environmental monitoring system and a permanent, and larger, sanitary pass station. SevRAO are also producing a radiation and safety management plan which will cover the systems of work at the site.

Task 5: Site Surveys

The Andreeva Bay site requires characterisation of its geology, hydrogeology and contaminative state for:

a) construction of new facilities both for infrastructure and improving SNF management;

b) establishment of groundwater and surface water management systems (particularly with respect to the SNF area);

c) protection of the health of the site operators from uncontained contamination;

d) protection of the environment in the long-term with decisions on the future of the site and its final state.

These require measurements of the site's topography, its geological structure, the geotechnical properties of that geology in locations around proposed new facilities and those whose long-term stability requires checking, spatial distribution of contamination, etc. This work has been supported to date by Norwegian funding.

Some additional work has been identified; further boreholes around the DSU's and an engineering survey of the pier which will be supported by the UK.

Task 6: Integrated Database for Andreeva Bay

This Task to develop an integrated database has arisen due to the expanding programme of work that is being undertaken at Andreeva Bay and the need to manage the data and information being generated as part of this work. The initial objective of the database is to improve the coordination between tasks (including those funded by all Donors) by providing easy access to technical information, objectives and progress of tasks, completed and planned work, available and required machinery and equipment.

This Task is in its early stages and is identifying the user requirements and required functionality for the database.

Task 7: Criticality Monitoring

The FAAE have proposed the establishment of a criticality monitoring system for the DSU's. Whilst this has been accepted in principle by the UK, the actual specification of the equipment has yet to be defined. This in turns requires an understanding of the potential criticality events, and the physical phenomena which will result from these. Once this is determined the Task will proceed to determining the optimum monitoring system.

Conclusion

The UK has been funding projects at Andreeva Bay since 2002. Substantial progress has been made by the Russian Federation in characterizing the condition of the Site and its facilities; improving the radiation protection infrastructure and systems; improving the condition of DSU 1 (Tank 3A) and developing detailed options for the retrieval and transport of SNF from the site.

IMPLEMENTATION OF THE CONCEPT AND THE PROGRAM OF COMPLEX DECOMMISSIONING OF NUCLEAR SUBMARINES, NUCLEAR-POWERED SURFACE SHIPS AND MAINTENANCE VESSELS AND REHABILITATION OF RADIATION-HAZARDOUS FACILITIES: MAIN RESULTS AND UNRESOLVED PROBLEMS

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1. General Status of the Problem

1.1. INCREASE IN NUMBER OF RETIRED NUCLEAR SUBMARINES PENDING COMPLEX DECOMMISSIONING

Starting in the late 1950s, large-scale activities were deployed in the former USSR on development of the oceanic nuclear naval fleet comprising: general-purpose and strategic-missile Nuclear Submarines (NSs) – total built about 250 subs - and Nuclear-Powered Surface Vessels (NPSS) – total built 5 vessels. To support the nuclear fleet activities, appropriate supporting infrastructure was established: 4 naval Coastal Maintenance Bases (CMB) and more than 30 nuclear Maintenance Vessels (MVs).

Due to both expiration of service life of the above vessels and observation by the Russian Federation (RF) of the relevant international obligations, in the latter half of the 1980s active process of withdrawal from military service of general-purpose NSs, strategic-missile NSs and MVs began.

Complex decommissioning of NS differs considerably from that of non-nuclear vessels and armaments due to presence of nuclear Reactor Installation (RI) containing considerable amount of radioactive substances (up to 1 million Ci in activity) accumulated in Spent Nuclear Fuel (SNF) and RI metal structures.

Thus in addition to standard activities accompanying complex decommissioning of non-nuclear vessels – demilitarization (arms dismantling), dismantlement of equipment, cutting of main constructions into scrap metal, etc., – one has to do with some special operations, such as:

- supporting works in NS Reactor Compartment (RC) to prepare NS for

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 75–97.

defueling;

– SNF unloading from NS reactors, preparing for interim storage and shipment for reprocessing;
– RC cutting from NS hull and preparing for long-term storage; and
– SNF reprocessing and management of Solid and Liquid Radioactive Waste (SRW and LRW) generated during complex decommissioning of NSs.

The available then infrastructure of enterprises of RF Rossudostroenie, RF Navy and Rosatom (related to SNF shipment, storage and reprocessing) - mostly specialized in nuclear vessel construction, repair and support-of-running activities - was unprepared for large-scale environmentally-safe complex decommissioning of NSs complying with the paces of their withdrawal from military service. Those circumstances aggravated further by serious economic recession due to economic reforms initiated in 1990s, resulted in rapid gathering of many taken-out-of-operation NSs in their basing centers (see Fig. 1).

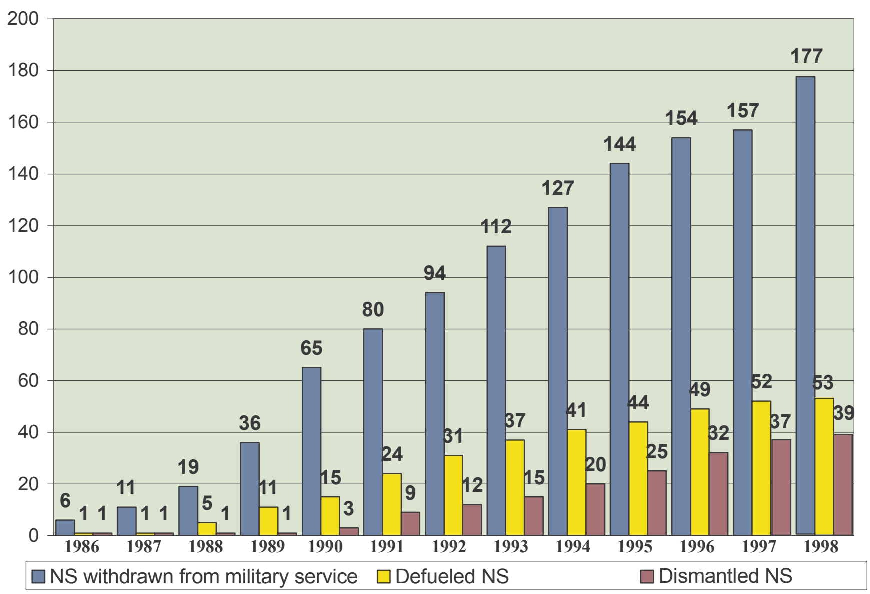

Figure 1. Dynamics of NS taking out of service, defueling and dismantlement before 1998

As seen from Fig. 1, most of NSs pending decommissioning were non-defueled.

Despite a number of decrees of the RF Government and ordinances of the RF President put into action in 1991-1996, the situation did not improve.

1.2. POTENTIAL RADIATION HAZARD

During the period under consideration the environmental situation at NS basing centers considerably aggravated due to gathering of non-defueled retired NSs with steadily worsening condition of hulls that created a risk of their non-controlled sinking.

Technical condition of nuclear- and radiation-hazardous objects at naval CMBs worsened too.

By the end of 1998 138 NSs were stored afloat in “pending decommissioning” condition including 124 non-defueled NSs.

By that time the majority of infrastructure facilities at 4 naval CMBs – two CMBs in the Northwest Russia (Fig. 2) and two CMBs in the Pacific Russia (Fig. 3) – constructed mostly in the 1960s became inoperative. Virtually no works on repair of CMB equipment, buildings and constructions were conducted. Only individual anti-wreck measures were taken.

As the result, storage facilities for SNF, SRW and LRW became nearly full, and since the early 1990s three of four CMBs (in Andreeva Bay, Gremikha and Krasheninnikov Bay) have not been practically used according to their purpose (by way of example see Fig. 4).

To accelerate solution of the challenges related to defueling and dismantlement of NSs withdrawn from military service and environmental rehabilitation of radiation-hazardous facilities of RF Navy, the RF Government (Decree #518 of 28.05.98) transferred the functions of state customer-coordinator of NS complex decommissioning activities from RF Ministry of Defense (MoD) to RF Ministry for Atomic Energy (Minatom, presently Rosatom).

Figure 2. Location of CMBs in the Northwest Russia

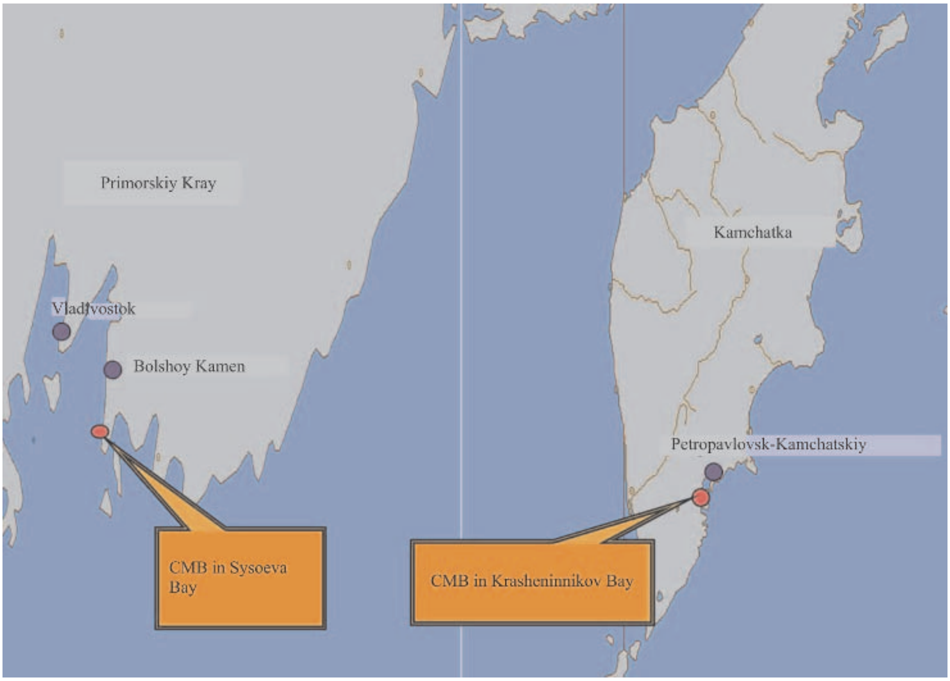

Figure 3. Location of CMBs in the Pacific Russia

Figure 4. Storage Facility for SNF and RW at CMB in Andreeva Bay

1.3. MAIN FUNCTIONS OF ROSATOM

In pursuance of the RF Government Decree #518, Rosatom acting as the state customer-coordinator of works for purposes of:

- minimizing (excluding) the probability of nuclear and radiation incidents at the retired NSs, NPSSs and MVs and former naval CMBs;
- fulfilling international obligations of the Russian Federation on nonproliferation of nuclear materials and withstanding international terrorism; and
- removing alien functions from RF MoD

supports:

- complex decommissioning of retired NSs, NPSSs and MVs including:
- measures on ensuring all-type safety at NSs taken out of military service;
- reconstruction and development of new equipment to perform NS defueling;
- construction and running of buildings and equipment for safe interim storage of SNF at NS defueling centers before removal for reprocessing;
- removal from regions and reprocessing of SNF of retired NSs at PA “Mayak”;
- making up of RC units and cutting of nose and stern compartments of retired NSs;
- safe storage of made up RC units;
- unloading of SNF and Radioactive Waste (RW) from storages of MVs subject to complex decommissioning;
- sealing, temporary afloat storage and subsequent complex decommissioning of MVs;
- collection and processing of RW generated during complex decommissioning of NSs; and
- environmental rehabilitation of CMB facilities temporarily housing SNF, SRW and LRW including:
- collection, processing, conditioning and subsequent storage of all-type RW accumulated at CMBs;
- development and implementation of SNF transport and management cycles including SNF removal from CMBs and forwarding for reprocessing; and
- complex of works on environmental rehabilitation of CMB buildings, constructions and sites.

1.4. PRINCIPAL LEGAL AND STANDARD DOCUMENTS

Implementation of works related to complex decommissioning of NSs is supported by the following main legal and managerial documents:

- orders and decrees of the RF President;
- Decree #158 of RF Government of 28.05.1998 "On Measures on Speeding up Complex Decommissioning of Nuclear Submarines and Nuclear-Powered Surface Vessels Withdrawn from Military Service and Environmental Rehabilitation of Naval Radiation-Hazardous Facilities";
- "Concept of Complex Decommissioning of Nuclear Submarines" approved by Minister of RF Minatom and prescribed as the guide to action by RF Government Order # IK-P7-02738 of 17.02.2001;
- Decree of RF Government #220-r of 9.02.2000 on establishing Federal State Unitary Enterprise (FSUE) "SevRAO" and FSUE "DalRAO" on basis of former naval CMBs;
- Concepts of environmental rehabilitation of CMBs in the Russian Northwest and the Pacific regions agreed with all interested federal bodies and approved by RF Rosatom's Leader;
- Receipt-transfer schedules for NSs and MVs approved by RF Minister of Defense, Rosatom's Leader and Director General of Rossudostroenie;
- "Order of Transferring to Work Executors of NSs, NPSS, Diesel Subs, Surface Ships and Maintenance Vessels Withdrawn from Service in the Russian Navy and of MoD facilities Used for Temporary Storage of SNF, SRW and LRW"; and
- annual Joint Decisions of Rosatom, RF MoD and Rossudostroenie "On Measures on Defueling NSs Subject to Complex Decommissioning" appended by defueling schedules and details on use of defueling equipment and SNF removal to PA "Mayak" for reprocessing.

The main provisions of the above Concepts comply with the internationally-recognized principles and agreements related to the use of atomic energy and are aimed at attaining the following main common goal: upon completion of the life cycle of the objects in question their ultimate condition must not be a source of further radiation and environmental hazard.

Thus gradual diminishing of radiation and environmental risks should be the main objective at all phases of NS complex decommissioning and CMB rehabilitation.

The above concepts were developed on basis of the following fundamental provisions and objectives:

- unconditional ensurance of nuclear and environmental safety on basis of the existing legislation at all stages of complex decommissioning of NSs, NPSSs and MVs being taken out of Navy service and environmental rehabilitation of CMBs including NSs and NPSSs under waterborne storage pending decommissioning with ensurance of their explosionproofness, fire safety and floodability;

- implementation of the “closed” cycle of managing SNF from decommissioned NS, NPSS and rehabilitated Navy radiation-hazardous objects at the existing processing lines of PA “Mayak” for SNF receipt and reprocessing and the possibility (if necessary) of SNF temporary storage in dry containers before it is shipped for reprocessing;
- optimal use of the available infrastructure facilities for the sake of complex decommissioning of NSs, NPSSs, MVs and environmental rehabilitation of CMBs;
- implementation of the “delayed” decommissioning of radiation-hazardous equipment of ship Nuclear Power Installations (NPIs) and disposal of the equipment, which cannot be reused after a long-term hold up (down to a state acceptable for RC cutting, i.e. about 70 years after reactor shutdown) in the form of RCs specially-prepared for long-term storage;
- maximum possible use, according to the established procedure, of free space inside RCs to place SRW generated during preparative-to-NS-defueling operations or in the course of NPI repair/updating and temporarily stored at the enterprise-executor of works on complex decommissioning of NSs (NPSS);
- openness and accessibility of information for the local public as regards ongoing or planned works related to complex decommissioning of NSs, NPSSs and MVs and environmental rehabilitation of CMBs as well as measures on ensuring nuclear, radiation and environmental safety, results of performed peer reviews on engineering and technological solutions in support of complex decommissioning of NSs and NPSSs and environmental rehabilitation of CMBs and status of objects and facilities save for information items constituting a state and commercial secret;
- observance by all work executors of the principles of nuclear technology nonproliferation and ensurance of national safety of the Russian Federation.

To attain the above objectives of the Concepts based on the priorities of protecting life and health of the present and future human generations as well as the environment against noxious impacts of radiation sources, nuclear materials and radioactive substances, observance of the following main international principles during work organization and direct handling of radiation sources, nuclear materials and radioactive substances is mandatory:

- decrease - down to acceptable level - noxious impacts of radiation sources, nuclear materials and radioactive substances on human health and environment at present and in future. Excess burden on future generations should be avoided;
- providing acceptable level of human health protection against noxious

effects of radiation sources, nuclear materials and radioactive substances according to the principles of optimization, standardization and justification of activities;

- due regard for potential consequences for human health and environment beyond the RF frontiers;
- predictable implications for the health of future generations must not exceed the relevant levels acceptable at the present time;
- establishment of appropriate state legal structure providing for distribution of obligations between the authorities performing public administration and state regulation of safety issues;
- generation of RW subject to disposal should be kept at the as-low-as-reasonably-achievable level, and their characteristics should comply with the standards and rules in force in the Russian Federation in the area of the use of atomic energy and RW management.

2. Main Results of Rosatom activities during 1999-2003

After 1999, when RF Minatom began acting as “the state customer - work coordinator”, the following main activities ensuring environmental safety in the area of complex decommissioning of NSs were conducted:

- Safe storage of NSs, NPSSs and RCs of NSs;
- Safe transfer of retired NSs, NPSSs and MVs to enterprises-executors;
- Defueling of reactors of NS and NPSS, management of damaged NSs and MVs;
- Safe management and reprocessing of SNF;
- Making up of one-compartment (three-compartment) RC units of NSs;
- Construction of long-term storage facilities for RCs;
- RW collection and processing;
- Collection and disposal of noxious and toxic waste;
- Complex decommissioning of MVs; and
- Rehabilitation of contaminated objects at CMBs.

2.1. COMPLEX DECOMMISSIONING OF NSs AND NPSSs

2.1.1. Ensuring Safe Unloading and Management of SNF

As of the beginning of 1998, defueling of NS reactors in Russia was supported by only three naval Floating Service Vessels (FSVs).

Storage facilities for temporary storage of SNF unloaded from NS reactors were either full or damaged.

SNF was removed for reprocessing by only one special train of 4 railcars. Only 7 SNF-removal runs of special train were possible per year at best.

Taking into account the greatest environmental and radiation hazard issuing from non-defueled NSs and with due regard for the fact that about 25% of retired non-defueled NSs lost most of their reserve buoyancy due to corrosion of basic structures and depressurization of driving ballast tanks, Rosatom concentrated most of efforts on immediate defueling of retired NSs and safe management of unloaded SNF.

To ensure safe SNF unloading:

- renewal of three FSVs, design 326M, was performed, and the relevant documentation authorizing prolongation of their service life was drawn up;
- routine repairs of three operating FSVs, design 2020, was made along with repairs of the available sets of OK-300PB reloading equipment and fitting out of the latter with new units to support defueling of reactors of the first- and second-generation NSs;
- two new sets of OK-300PBU reloading equipment were made;
- NS defueling flowsheets using MVs of the Murmansk Shipping Company were implemented; and
- two on-shore defueling complexes were constructed and commissioned at FSUE “Zvezdochka” ShipYard (SY) and FSUE Far East Plant (FEP) “Zvezda” (Fig. 5).

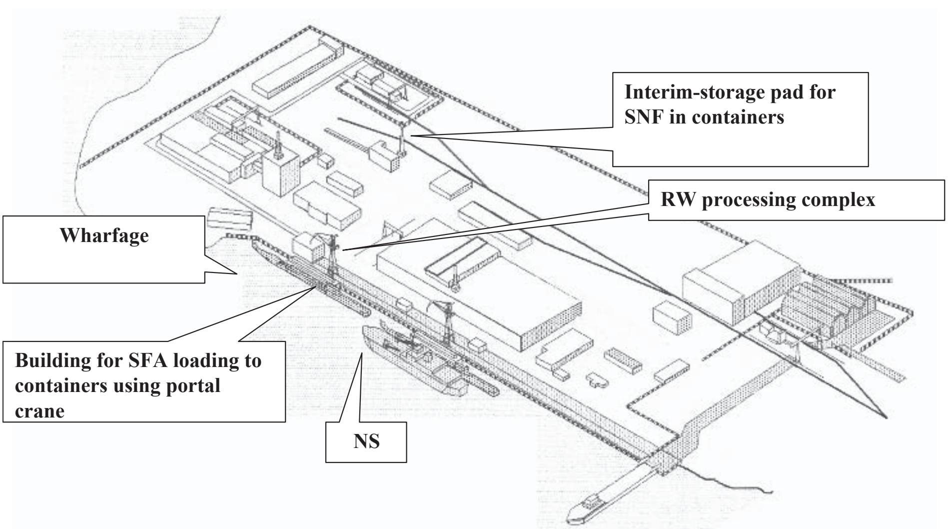

Figure 5. “On-shore defueling facilities”

To ensure safe temporary storage of SNF, the “dry” container-storage technology was implemented (Fig. 6). For this purpose:

- 48 dual purpose metal-concrete casks of TUK-108/1-type were developed, manufactured and prepared for SNF storage and transportation;
- a lot of 25 casks, TUK-108/1-type, was being constructed funded by the US CTR Program;
- construction of 2 pads for temporary storage of SNF in containers (up to 60 pieces) was completed at FEP “Zvezda” and “Zvezdoshka” SY using funds of the US CTR Program;
- a transshipment pad for SNF in containers was commissioned at FSUE “Atomflot” using funds of AMEC Program (USA, Norway);
- a design of Building #301 reconstruction (PA “Mayak”) was developed using funds of the US CTR Program to establish a buffer storage for 154 containers with SNF; and
- temporary-storage pad for containers with SNF at “DalRAO” was enlarged (up to 35 storage places).

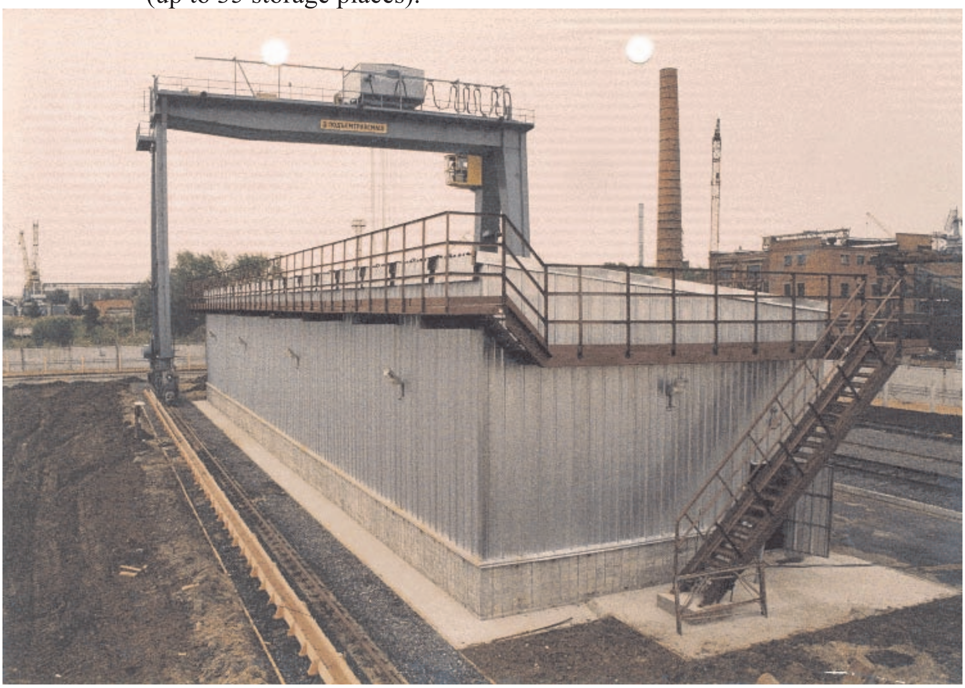

Figure 6. Interim-storage pad for containers with SNF at “On-shore Defueling Facilities”

To accelerate paces of SNF removal to PA “Mayak”, two trains of special railcars for SNF container shipment were constructed and commissioned using funds of Norway and the USA.

Thanks to the implemented measures safe defueling of about 20 NSs per year and safe management of their SNF became possible.

2.1.2. Reactor Compartment Unit Making up

It is obvious that the paces of retired NS defueling should be coordinated with the possibilities of carrying out the subsequent complex decommissioning phases.

Based on the performed R&D and feasibility studies, “The Concept of Nuclear Submarine Complex Decommissioning” admits the so-called “delayed” decommissioning of RI equipment and RC themselves. Despite removal of about 90% of accumulated radionuclides after NS defueling, residual activity of RI equipment and RC basic constructions presents serious radiation hazard. Consequently, operations are necessary on preparing RC of retired NSs for protracted storage until attaining allowable conditions for their ultimate dismantlement, recycling and subsequent disposal of low-active and medium-active equipment. RC cutting out of NS hull allows immediate safe dismantlement and recycling of NS nose and stern parts clearing thereby the shipyards’ wharfage for defueling of the next-in-turn NS.

The temporary NS dismantling technology applied during the considered period at shipyards consisted in RI cutting in the form of a three RC unit (RC plus two adjacent compartment to ensure buoyancy) and sealing for subsequent temporary waterborne storage (Figs 7, 8 and 9).

Figure 7. Three-compartment RC unit of NS dismantled on slip

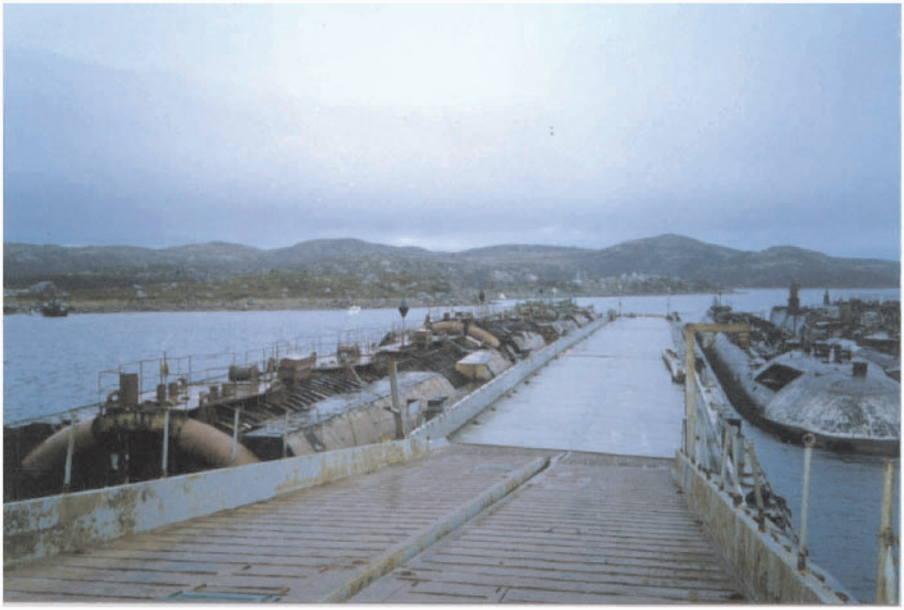

Figure 8. Three-compartment RC units of dismantled NSs afloat in Saida Bay

Figure 9. Three-compartment RC units of dismantled NSs afloat in Razboinik Bay

Over the period under consideration a variety of measures was performed - mainly, thanks to the international assistance - on upgrading industrial infrastructure at shipyards and ensuring environmental safety of the whole industrial cycle (equipment with high-capacity cutting mechanisms, establishment of special cable-cutting areas, and so forth) that made it possible to increase the paces of both RC unit making up and NS end part cutting.

Advanced technology of NS decommissioning provides for making up of one-compartment RC unit followed by its long-term safe storage at special on-shore pads (Fig. 10).

Figure 10. One-compartment RC units on slip-way

However in the period in question implementation of the advanced technology was hindered by lack of on-shore one-compartment RC storage centers and appropriate transport facilities.

Design documentation was developed for construction of on-shore facilities for long-term storage of one-compartment RC units: in the Northwest region – in Saida Bay (Murmansk region) and in the Pacific region – in Razboinik Bay (Primorskiy kray).

An agreement was signed between the Russian Federation and the Federal Republic of Germany under the G8 Global Partnership Cooperation Program on assisting Russia in construction of an on-shore storage facility for RC in Saida Bay (the construction works have started recently).

During the period under consideration similar-type on-shore storage facility in Primorskiy kray was being constructed using funds of the RF State Budget.

2.1.3. Management of Damaged NSs

When resolving the issues of environmentally safe complex decommissioning of retired NSs, two afloat-stored non-defueled NSs with damaged NPIs needed special consideration. None of them could have been safely dismantled using standard technologies, and thus development and implementation of very special procedures was necessary. According to the results of performed investigations, the wisest way of managing the NSs in question would be installing them onto a "solid basement" and next confining within a special storage facility that would improve the radiation situation in the surrounding area up to normal level and exclude radionuclide release to the environment.

It was expected that defueling and dismantlement of the damaged NSs would be possible after long-term (100-200 years) hold up at the storage facility, i.e. after radionuclide decay down to levels allowing necessary dismantling works with acceptable radiation burden for involved personnel, no environmental damage and risk for the nearby population.

2.1.4. Nuclear-Powered Surface Ships

By the turn of 2003 two Nuclear-Powered Surface Ships (NPSSs) were withdrawn from military service: one NPSS in the Northwest region and the other one in the Pacific region.

To prepare NPSSs for complex decommissioning, Rosatom supported development of:

- alternative NPSS complex decommissioning concepts;
- basic technologies of NPSS dismantlement and their RC management;
- engineering solutions on managing SRW generated when defueling retired NPSSs;
- transport and management flowsheets for SNF of NPSSs; and
- feasibility studies for NPSS complex decommissioning.

Since 2004 design and engineering documentation for complex decommissioning of NPSS, including NPSS defueling, has been developed.

2.1.5. Prospects

Based on analysis of the above problems one can state that the actual industrial and transport infrastructure allows supporting complex decommissioning (i.e. SNF unloading and removal for reprocessing, RC making up and preparing for long-term storage, cutting of end parts, etc.) of 18-20 NSs per year.

However with due regard for real budget funding of the entire program of complex decommissioning of NSs, MVs and environmental rehabilitation of former naval CMBs, optimal NS decommissioning paces determined by nuclear, radiation and environmental safety, are estimated today at 13-15 NSs per year at the most.

Such paces would allow completing the majority of related works only by 2010 (Fig. 11).

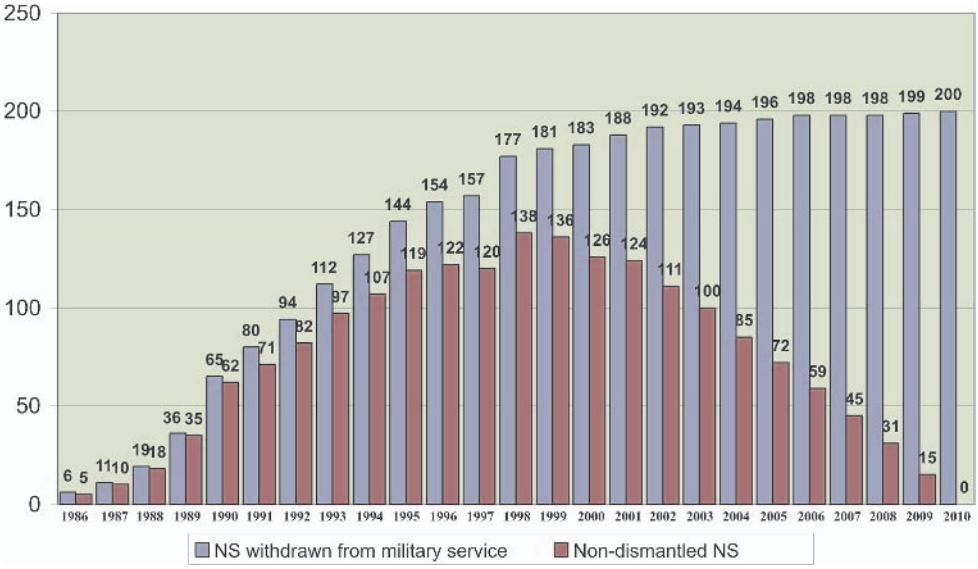

Figure 11. Dynamics of NS taking out of military service and dismantlement

2.2. RADIOACTIVE WASTE MANAGEMENT

2.2.1. General Status

Data on SNF and RW amounts accumulated at the objects of complex decommissioning and environmental rehabilitation, as of the mid-1998, are summarized in Table 1. Obviously, such a situation required urgent measures aimed at improving the whole RW management process.

TABLE 1. Summary data on SNF and RW in 1998

No	Object name	Object number	Core number at objects	SRW amount at objects, m3	LRW amount at objects, m3	Integral activity at objects, Ci
1	Afloat stored defueled NSs	14	-	891	210	~ 0.8·106
2	Defueled RC units	37	-	2331	-	~2.0·106
3	Non-defueled NSs including 2 RC with spent removable units	126	236	7938	12500	~44.0·106
4	MVs	41	13	-	3600	~2.0·106
5	CMBs (Northwest Russia)	2	86	4600	3200	~13.0·106
6	CMBs (Pacific Russia)	2	18	15480	3100	~ 2.5·106

2.2.2. Main Activities on Upgrading RW Management

To ensure environmental safety and reduce LRW amount:

- stationary LRW processing facilities were commissioned at FEP “Zvezda” and "Zvezdochka" SY funded by CTR Program (USA);
- updating of a LRW processing facility at “Atomflot” enterprise – funded by Russia, Norway and the USA – was nearing completion;
- three mobile LRW-processing facilities were made and put into operation in Murmansk region, Kamchatka region and Primorskiy kray;
- LRW processing floating complex (processing barge “Landysh ") provided by the Japanese Government was commissioned (Fig. 12).

Figure 12. LRW processing barge "Landysh"

After putting the above facilities into operation all LRW produced during nuclear vessel complex decommissioning was processed and conditioned. The integral amount of previously accumulated LRW began decreasing gradually.

To manage SRW:

– SRW processing facilities were commissioned and put into operation at FEP "Zvezda" and "Zvezdochka" SY funded by CTR Program (USA);
– operational center on SRW conditioning was constructed and commissioned at SY #10 of RF Ministry of Defense funded by AMEC Program (USA, Norway);
– necessary standard and engineering documentation was developed, and a flowsheet for SRW (of NS decommissioning origin) placing into made up RC units was implemented.

But, though implementation of the above measures allowed preventing further accumulation of SRW of NS-decommissioning origin, today the issues of SRW management still require top-priority consideration.

To transfer low-active and medium-active SRW accumulated at SYs and former naval CMBs to safe condition, urgent establishment of regional centers on processing, deep conditioning and long-term storage of conditioned RW is necessary.

2.3. REHABILITATION OF RW AND SNF STORAGE FACILITIES

2.3.1. Condition of Coastal Maintenance Bases

By 2004 four CMBs (2 in Murmansk region - in Andreeva Bay and Gremikha and 2 in the Far East Russia - in Sysoeva Bay, Primorskiy kray, and in Krasheninnikov Bays, Kamchatka region) were withdrawn from military service.

When developing environmental rehabilitation plans for the former CMBs to be implemented by "SevRAO" and "DalRAO", the following circumstances were taken into consideration:

- facilities in Andreeva Bay (Fig. 13) and Krasheninnikov Bay would not be used in future according to their initial design. Only works related to removal of SNF, SRW and LRW from their sites and rehabilitation (elimination or mothballing) of buildings and constructions and restoration (decontamination) of territories would be carried out;
- at the former CMB in Gremikha, in addition to works on environmental rehabilitation, both restoration and reconstruction of infrastructure would be necessary to support unloading and subsequent temporary storage of Liquid Metal Coolant (LMC) cores of NSs, designs #705 and #705K;
- the former CMB in Sysoeva Bay would be used in the foreseeable future for receiving and processing SRW and managing SNF of retired NSs in the Pacific region.

Figure 13. SNF storage facilities at CMB in Andreeva Bay

2.3.2. Top-priority Works at CMBs

By 2004 a variety of works on environmental rehabilitation of radiation-hazardous facilities at “SevRAO” and “DalRAO” was performed. Taking into consideration real condition of buildings and constructions at the former CMBs, first of all, works on restoration of communications and establishment of appropriate infrastructure to ensure safe working conditions for personnel were conducted.

Individual practical activities on diminishing environmental hazard of the formed CMBs were also carried out, such as:

- radiation survey of CMB sites;
- commissioning of interim storage pad (4500 m^3) for low-active container storage of SRW;
- fragmentation of bulky SRW (180 m^3 in volume); and
- gathering from open pads of 200 m^3 of SRW, their packaging and placing into storage facilities.

As the result of implementation of the above measures, radioecological situation at the former CMBs slightly improved.

It is worthy of notice that the SNF amount presently stored at damaged storage facilities of former naval CMBs is quite comparable to that to be unloaded from reactors of retired NSs. Such a situation requires acceleration of the paces of SNF-removal from CMB storage facilities, SRW and LRW processing and conditioning.

Urgent necessity of such works is also due to the fact that storage facilities of former CMBs (especially those designed to house SNF and SRW) have virtually no localizing safety barriers as distinct from retired NSs (RI vessel structures, NS/RC strong hull).

SNF and RW storage facilities at former naval CMBs are explicit sources of environmental hazard being the most vulnerable objects to potential natural and man-caused (including terrorist) threats.

2.3.3. Plans of Future Works

In addition to the above-listed top-priority works on restoration of infrastructure ensuring occupational and environmental safety, the plans for former naval CMB remediation also provide for:

– performing Integrated Engineering and Radiation Survey (IERS) of buildings and constructions for purposes of subsequent selection of an optimal and safe rehabilitation option;
– developing and implementing transport and management flowsheet for SNF stored at CMB's facilities including safe removal of SNF from their sites;
– designing and making necessary equipment and carrying out practical works on SRW conditioning and preparing to long-term storage;
– performing ultimate phase of environmental rehabilitation of CMB buildings and constructions and restoration of their sites.

To date deployment of large-scale works on environmental rehabilitation of CMBs is hindered by generation during such activities of considerable extra-amounts of LRW and SRW.

Taking into account both SRW amount already stored at CMB sites and newly-generated RW (in the course of SNF management, rehabilitation of buildings and facilities and restoration of territories), the option on establishing regional centers for processing, deep conditioning and preparing low-active and medium-active SRW to subsequent disposal at the site of "SevRAO" Branch #1 (Andreeva Bay) and the main site of "DalRAO" (Sysoeva Bay) appears to be the optimal solution.

Establishment of such centers would allow not only accelerating environmental rehabilitation of the former CMBs but also receiving RW accumulated at the sites of SYs concerned with NS decommissioning and supporting ultimate decommissioning of nuclear Maintenance Vessels (MVs).

2.4. COMPLEX DECOMMISSIONING OF MVs

By 2004 41 Maintenance Vessels (MVs) were taken out of service for subsequent complex decommissioning at the Russian Northern Fleet and the Pacific Fleet (FSV design 326, special tankers (TNT-type), Floating Control Dosimetry Vessels (PKDStype), etc.). Works on defueling of damaged storages of FSVs PM-80 and PM-32 were completed, and type technology of MV complex decommissioning was developed.

As expected, during MV decommissioning activities considerable SRW amount would be inevitably generated. Under lacking SRW safe-management infrastructure (i.e. regional centers for SRW processing, conditioning and storage) and for purposes of reducing environmental risks, the following 3-phase sequence of MV complex decommissioning appears to be the most appropriate:

1. MV sealing to ensure safe temporary waterborne storage at "SevRAO" or "DalRAO".
2. Temporary afloat storage of MVs before commissioning of regional centers on SRW management and temporary storage facilities for RC of former NSs.
3. MV cutting and SRW transfer to regional centers for processing, deep conditioning and subsequent storage. LRW processing. Transfer of non-processible storage blocks with SNF to Long-term Storage Facility for RC.

These proposals have been already approved by special decisions of Rosatom and Russian Navy.

2.5. YIELDED RESULTS

Complex Decommissioning of NSs (Table 2)

Since 1999 annual average paces of NS defueling and dismantlement including RC unit making up have increased by about 3.5-4 times as compared to the previous period.

TABLE 2. Current situation as of September 2004

NS taken out of service	194
Made up RC units of dismantled NSs	103
Defueled NSs	129
NPSS taken out of service	2 (non-defueled)
MV taken out of service	44
CMB to be subject of environmental rehabilitation	4
NS to be dismantled	91
NS reactors to be defueled	65

RW Management

To date implementation of SRW and LRW management procedures allows not only preventing their further accumulation but also diminishing gradually RW amount accumulated previously at shipyards in the course of NS repair and decommissioning operations.

Rehabilitation of Former Naval CMBs

At present works on restoration of infrastructure and improvement of RW management are performed at former naval CMBs. Programs and methods of IERS of buildings and constructions, transport and management flowsheets for SNF and RW stored at the CMB facilities with due regard for safety and decrease in radioecological hazard are being developed.

Complex Decommissioning of MVs

Upon agreement with interested agencies, a decision has been taken on step-by-step decommissioning of MVs allowing for the probability of hazardous radiation incidents to be considerably decreased.

Damaged storages of FSVs PM-32, PM-80 and MV "Siverka" have been defueled. Complex decommissioning of 4 MVs has started.

3. Conclusions and Proposals

1. Implementation of a variety of program activities on complex decommissioning of NSs, NPSSs and MVs and environmental rehabilitation of radiation-hazardous facilities has ensured:
   - increase in NS defueling and dismantling paces (including making up of three-RC units) by 3.5-4 times as compared to 1985-1998;
   - transfer to advanced procedure of NS complex decommissioning with making up of one-compartment RC units;
   - prevention of RW accumulation at SYs concerned with NS complex decommissioning and gradual reducing of previously accumulated RW amount;
   - works on: partial restoration of infrastructure at former naval CMBs; increase in radiation safety level; radioecological survey of buildings, constructions and territories;
   - defueling of damaged storages of MVs and onset of works on complex decommissioning of such MVs;
   - preparing necessary scope of design, engineering and project documentation supporting complex decommissioning of NPSSs.
2. Some problems still remain unresolved hindering implementation of the program of NS complex decommissioning and environmental rehabilitation of

former naval CMBs. The most important unresolved problems are due to lack of:

- railway communication between FEP "Zvezda" and Smolianinovo station (Primorskiy kray) needed to support SNF removal for reprocessing;
- up-to-date high-capacity regional centers on processing, deep conditioning and long-term storage of SRW in the Northwest Russia and the Far East Russia;
- ultimate decision concerning selection of optimal options on managing SNF and RW accumulated at former naval CMBs including damaged SNF and that of "*Alpha*"-class NS;
- rehabilitation of CMB buildings and constructions;
- commissioning of on-shore long-term storage facilities for RCs;
- transport facilities providing for safe haulage of NSs from their basing centers to enterprises-executors of complex decommissioning activities;
- comprehensive and reliable information on quantity, type and condition of SNF and RW at storages of former naval CMBs and on condition of their buildings and constructions;
- up-to-date technologies and equipment on safe management of toxic waste generated during complex decommissioning of NSs and NPSSs; and
- regional-scale radiation and environmental monitoring system.

3. The following activities are proposed as priority ones allowing resolving the above problems:

- reconstruction of railway section from Smolianinovo station to FEP 'Zvezda" giving a way for passage of special train of TK-VG-18 (TK-VG-18/A)-type railcars with SNF;
- performing IERS of territories, buildings and constructions at former naval CMBs including the adjacent water areas. Collection and analysis of information on amount, types and condition of SNF and RW;
- development and implementation of projects on optimal and safe management of SNF and RW stored at CMB storage facilities;
- development of projects and establishment of regional centers on processing, deep conditioning and long-term storage of SRW;
- completing construction of on-shore facilities for long-term storage of RCs; and
- development and implementation of projects on establishing regional-scale radiation and environmental monitoring system.

A PERSPECTIVE ON U.S. SPENT NUCLEAR FUEL POLICY

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1. Introduction

The management of spent nuclear fuel (SNF) for U.S. submarines is intimately intertwined with the overall spent fuel policy of the U.S. This paper summarizes the current SNF policy for the U.S. from both a historical perspective and the perspective of the key issues that are shaping the policy and its future direction. SNF policy focuses on what is known as the 'backend' of the nuclear fuel cycle – all those components of the cycle after the fuel is removed from a reactor. It also can and does impact the 'front end' of the fuel cycle (the components necessary to produce nuclear fuel for a reactor).

There are, in essence, three possible basic SNF policies: 'full recycle' where SNF is processed to recover nuclear fuel materials not used in the reactor to make new fuel; 'once-through' where the SNF is not processed and the fuel is stored essentially 'as-is' in interim storage or final storage or disposal (geologic repositories, ocean bed repositories, outer space, etc.); and hybrid policies which combine features of both full recycle and 'once-through'.

Even in the 'full recycle' mode, there are still nuclear waste products that must be disposed of. Early reprocessing technologies using the PUREX process were quite successful at separating unfissioned plutonium and the remaining uranium from the spent fuel that could be used in the production of new fuel for reactors. However, these earlier technologies also left substantial quantities of what is known in the U.S. as 'high level' nuclear waste which must be disposed of. Most of this waste has come from past reprocessing of government nuclear fuel used in production, naval, research, and test reactors. Thelegacy of these wastes is the subject to a massive environmental restoration program within the U.S. Department of Energy. Only a small quantity of commercial fuel in the U.S. was ever reprocessed (from 1966 to 1972). Most commercial spent fuel is stored in tact in storage pools or dry cask storage at the reactor sites around the U.S.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 99–106.

This, too, is an issue since the limits of pool storage are being reached and dry cask storage is expensive. There are about 47,000 metric tomes heavy metal (MTHM) of commercial spent fuel and 2500 MTHM of government owned spent fuel in storage, and about 4700 MHTM of high level waste in the U.S. (ref: 6.10 – values adjusted to current date)/

The U.S. government is responsible for disposition of the SNF and high level waste. Because of the waste issues discussed above and concerns about nuclear proliferation, spent fuel policy is a politically charged issue. It is also inherently tied to the future success or failure of nuclear power as a future energy option for the U.S.

2. Drivers In Shaping Spent Fuel Policy

In order to understand U.S. SNF policy, one must first understand the key drivers that shape the policy:

1.1. DEMAND FOR NUCLEAR FUEL

Sustained growth in the demand for nuclear fuel provides incentives for encouraging a 'recycle' spent fuel policy. This demand is driven by the overall energy demand for the U.S. and, in turn, by the availability of the alternative sources of energy and their respective costs to meet that demand. In the 1950's through the early 1970's, energy was plentiful but the promise of nuclear power was new. Consequently, spent nuclear fuel was considered a valuable resource. Only about 5% of the available energy is burned in a typical commercial reactor. For the initial generation of nuclear engineers and scientists at the time it made sense that one would want to recover the uranium and plutonium from reprocessing of the spent fuel. In the 1980s, though U.S. policy did not specifically preclude spent fuel reprocessing, uranium was plentiful, and due to the extra costs and proliferation concerns (discussed later), reprocessing was not necessary. Currently, it is recognized that fossil fuel sources have a limited life and their contribution to global warming is a world concern. Consequently, nuclear power is being reconsidered seriously as an energy option for the future. In the same light, the economical supplies of uranium ore are in some circles only considered sufficient for about 50 years supply at moderate nuclear growth rates (ref: 6.14). Consequently, if nuclear power is to be a serious contributor in the future, spent fuel recycle will have to be used.

1.2. NON-PROLIFERATION CONCERNS

Risks of nuclear weapons proliferation tend to discourage spent fuel re-cycle. Spent fuel, by itself, is not considered a serious near-term proliferation risk due to the complex reprocessing technologies required to separate nuclear weapons grade materials from the spent fuel. The fuel requires remote equipment to handle it (a commercial PWR fuel assembly weighs about 600 kg and ten years after it is removed from the reactor still gives off radiation of about 20,000 rem/hr at a distance of 1 meter – a dose lethal to humans in a fraction of an hour). From a simplistic perspective, reprocessing technology separates weapons usable materials from the spent fuel and thereby makes them potentially available for miss-use. Though the realities of this at less than a national scale make the risks of this remote, the theft of separated weapons usable material from a re-cycling facility is not beyond question and therefore a difficult area for policy makers to deal with. Consequently, in the current day, any spent fuel re-cycle option will have to have significant anti-proliferation barriers built into the system. U.S. non-proliferation policy "does not encourage the civilian use of plutonium and, accordingly, does not itself engage in reprocessing" (ref: 6.2).

1.3. AMOUNT, FORM, AND CONTENT OF NUCLEAR FUEL WASTE CONSTITUENTS

Spent fuel recycling can reduce the amount of high level nuclear waste that must be disposed of in a geologic repository. In fact, recycling could significantly lengthen the time before a given geologic repository's capacity is filled. In addition, recycling can make nuclear waste disposal a simpler engineering problem to solve. Spent fuel by itself poses a serious radioactive waste challenge in that its constituents are hazardous for thousands of years. In fact, it would take over 300,000 years for a spent fuel element to decay radioactively to the same level as the natural uranium it came from. Recycling offers the potential of significantly reducing this amount (potentially to 1000 years) (ref: 6.10). Consequently, the incentive of having a more manageable nuclear waste solution with spent fuel recycling is a positive driver for a full recycle policy.

1.4. PERCEPTION OF SPENT NUCLEAR FUEL SAFETY

The perception of nuclear facility safety has always been an important driver for any facet of nuclear energy policy and generally discourages new facilities due to vocal local opposition. Safety perceptions tend to promote the status quo and am attitude of not desiring a nuclear facility in your neighborhood – colloquially known as 'NIMBY' (Not In My Back Yard). For example, insofar as spent fuel policy is concerned, perhaps the biggest safety concern areas are transportation issues and the 'safety' of the high level waste as it is stored in final disposal. The current 'status quo' has spent fuel and high level waste being stored close to the facilities where they were generated and the geological repository not open. Consequently, local vocal public pressure (and thus political pressure) will push against the repository opening and the subsequent spent fuel transportation from all parts of the country to fill the repository. This would also tend to be true if full-recycle were used. Challenges to the location and safety of reprocessing facilities would be brought up. The status quo of making no changes would be the typical political fall-back policy position given no other urgent national need.

1.5. COSTS AND ECONOMICS

There is not much that needs to be said here. Whatever policy is chosen, the cost of the policy is always an issue. Relative costs between policies are often difficult to calculate due to the complex interrelationships involved. For example, the cost of a full recycle policy must be measured comparing the costs of building and operating reprocessing facilities and new reactors with the alternative of building and operating new repositories. Complex trade-off studies are often required. At the present time, reprocessing costs make full-recycle clearly more expensive.

3. History of Spent Fuel Policy

The U.S. national policy on nuclear issues has evolved though laws and regulations that established federal agencies responsible for the regulation and promotion of nuclear energy. Initially, the Atomic Energy Act of 1954 laid the foundation by ending the U.S. government monopoly on control of nuclear technical information and made the growth of a private commercial nuclear industry an urgent national goal (ref: 6.11) The de facto spent fuel policy of the U.S. in the beginning years of nuclear power through about 1977 was for full recycle. Reprocessing was initially necessary for production of weapons grade plutonium. Spent nuclear fuel was also considered a valuable commodity for its potential energy value. All of today's power plants were designed and ordered during this time frame when it was understood that the spent fuel would go to reprocessing plants. A demonstration reprocessing plant for commercial spent fuel was built and operated at West Valley, New York from 1966 – 1972, when it was shutdown to make plant improvements. When these were later found to be un-economical, the plant operator decided to cease operations (ref: 6.12). General Electric also built a small plant in the early 1970's but never placed it in commission when initial testing of the plant indicated it would not be capable of reliable operation (ref 6.13). In the mid-1970's, the once supposed scarcity of uranium resources that was the fundamental drive for reprocessing never materialized. Renewed fears of proliferation in the Carter administration and the increased costs of reprocessing (as documented by West Valley), essentially killed reprocessing as a policy option. Though the Carter administration made 'once through' the spent fuel policy of the U.S. in 1977 and cancelled another commercial reprocessing plant venture at Barnwell (stranding half a billion dollars in private investment), Ronald Reagan rescinded this policy in 1981(ref: 6.13). However, a glut in the supply of uranium and the economics of starting up recycling plants effectively left the U.S. with an unofficial 'once through' policy.

In 1982, the Nuclear Waste Policy Act was passed that provided for the siting, construction, and operation of a deep geological repository for the disposal of spent fuel and high level waste and that USDOE would be responsible for developing it. The Act further provided that DOE was to take title to utility spent fuel after 15 years (January 1998) and that the utilities would pay 1 mill per kw-hr of power generated to a nuclear waste fund to fund the repository (ref: 6.2). The EPA was given responsibility for establishing radiation standards for the repository and the NRC was given responsibility for establishing licensing requirements for the repository to meet those standards.

In 1987, Congress passed the Nuclear Waste Policy Amendments Act. The Act provided that DOE restrict studies for a repository to Yucca Mountain, Nevada. The Act also authorized DOE construct an interim storage site for SNF, but the facility could not be located in Nevada and funds for its construction could not be authorized until Yucca Mountain was licensed for construction (for fear that the interim site would become a permanent repository). The act unwittingly made interim storage a de facto adjunct to

U.S. spent fuel policy since it forbade DOE from taking title to SNF until a permanent repository was approved for construction (ref: 6.2). In fact, private interim storage facilities have been proposed. A utility consortium signed an agreement with a Utah Indian tribe in December 1996 to build a 40,000 MT storage facility in the desert west of Salt Lake City. Proceedings to license the facility are still on-going (ref: 6.3).

In 1993, President Bill Clinton formally reinstated the national policy against reprocessing (ref: 6.13). In the mean time, USDOE was proceeding in its studies of Yucca Mountain. In 1994, exploratory work at Yucca Mountain began with initiation of excavation of the exploratory studies facility (ESF) (ref 6.3). In 1998, DOE completed a viability assessment, followed by a draft environmental impact statement (EIS) in July 1999. DOE then completed a preliminary site suitability evaluation in August 2001 that found Yucca Mountain could meet EPA and NRC requirements. In February 2002, DOE recommended the Yucca Mountain site to the President (submitting the final EIS and other supporting documents). The President recommended it to Congress the next day. The approval resolution was passed by Congress and signed by the President over the veto of the State of Nevada on July 23, 2002 (ref: 6.3). Current plans call for DOE submitting its license application to the NRC in December 2004 with plans for receiving its license to receive fuel in 2010. The State of Nevada will vigorously fight the opening of the Yucca Mountain repository by all means possible.

It may be recalled that 1998 was the expected date in the Nuclear Waste Policy Act of 1982 for the DOE to begin taking title to utility spent fuel. The DOE could not obviously meet this date and utilities have had to spend significant amounts of money expanding their SNF storage capability by building higher density storage racks and building dry cask storage areas. Consequently, since 1998, 64 lawsuits have been filed by utilities against DOE to recover the money that they have spent on these storages (ref: 6.7).

Yucca Mountain, even if licensed to receive spent fuel in 2010, only has the capacity to absorb the output of commercial U.S. spent fuel production and existing storage through about 2015.[1] Consequently, President Bush in his National Energy Policy issued in May 2001 (ref: 6.1) recommended:

– "in the context of developing advanced fuel cycles and next generation technologies for nuclear energy, the United States should reexamine its policies to allow for research, development, and deployment of fuel conditioning methods (such as pyroprocessing) that reduce waste streams and enhance proliferation resistance. In doing so, the United states will continue to discourage the accumulation of separated plutonium."
– "The United states should also consider technologies, in collaboration with international partners with highly developed fuel cycles and a record of close

[1] Yucca Mountain is statutorily limited to receiving 70,000 MTHM of SNF or HLW – of which 63,000 MTHM would be for the commercial sector – the balance for DOE and Defense spent fuel and HLW.

cooperation, to develop reprocessing and fuel treatment technologies that are cleaner, more efficient, less waste intensive, and more proliferation resistant."

Based on this policy, DOE has initiated new programs that could lead to nuclear fuel cycles that significantly reduce the amount and radio toxicity of spent fuel high level waste. If implemented in practice, this would result in a 'hybrid' spent fuel policy, using both deep geologic disposal and full recycle. This policy could possibly extend the lifespan of Yucca Mountain by many years. This will be discussed in more detail in the next section.

In summary, the U.S. has had two basic spent fuel policies

– the 'full recycle' policy initiated in the 1950s that lasted through 1977
– the 'once-through' policy from 1977 through to the current day[2]

4. The Current Situation

As stated in the previous section, new recycle and advanced reactor technologies, though not precluding the need for a deep geologic repository, offer the prospects of significantly reducing these technical challenges. These new technologies can significantly reduce the amount, radiotoxicity, and heat generation of SNF nuclear waste to the point where Yucca Mountain's life span could be significantly expanded. Currently, there are about 47000 MTHM of commercial SNF in storage and at current spent fuel production rates, the statutory limit of 63000 MTHM will be reached by 2015. In the wake of September 11th, it is more desirable to have SNF stored in central, underground location rather than distributed around the country (ref: 6.10). Additionally, the time frame where the SNF nuclear waste is considered toxic (usually taken as the toxicity of natural uranium) may be reduced from 300,000 years to 1000 years may be realizable. Thirdly, the new technologies can also significantly enhance the supply of nuclear fuel where the need for new deposits of uranium ore is minimized. With these prospects, the Bush administration in its National Energy Policy of May 2001 recommended the reconsideration of recycle and advanced reactor technologies that could fulfill such prospects. These recommendations are being implemented in the U.S. DOE's Advanced Fuel Cycle Initiative (AFCI) and Generation IV reactor development programs.

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[2] It should be noted that U.S. submarine spent fuel policy historically continued as 'full recycle' past 1977 until the end of the 1989 when economics essentially was the key driver for reverting to a 'once-through' policy.

The technical issues for AFCI and GEN IV programs may be broken into several intermediate and several long term objectives (ref: 6.10). The intermediate-term objectives are:

– Reducing high-level waste volumes
– Increasing the capacity of the planned geologic repository
– Reducing the technical need for a second repository
– Reducing the long-term inventories of plutonium in spent fuel, and
– Enabling recovery of the energy contained in the spent fuel

The long term objectives are:

– Reducing the toxicity of spent nuclear fuel
– Reducing the long term heat generation of spent nuclear fuel
– Providing a sustainable fuel source for nuclear energy
– Supporting the future operation of the Generation IV nuclear energy systems.

The AFCI program is divided into two phases to address these objectives. The AFCI program also requires substantial international cooperation. There is much for the U.S. to learn from the long running and successful recycle programs and research in France, Britain, Japan, and Russia. If the AFCI programs are successful, the amount of high level waste could be significantly reduced with a commensurate reduction of cost for the first repository and possible elimination of the need for a second.

5. Conclusion

U. S. spent fuel policy initially was a 'full recycle' policy – taking advantage of the promise of the peaceful benefits of nuclear power. Proliferation concerns and economic realities hit in the 1970's causing the policy shift to a once through policy which is the current policy of today. The energy realities of the current time are hitting home once again – ranging from concerns of fossil fuel impacts on global warming, the growing demand for energy (especially in the developing world), and concerns about running out of fossil fuels in the next 50 years. Driven by these realities, the U.S. is once again taking a renewed look at spent fuel recycle policies for the U.S. A new research and development initiative has been launched called the Advanced Fuel Cycle Initiative and GEN IV programs whose promise, if fulfilled, could lead to a nuclear renaissance in the U.S., fulfilling the original promise that nuclear power started with.

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ACTUAL PROBLEMS IN IMPLEMENTATION OF THE INTERNATIONAL AGREEMENTS UNDER THE G8 GLOBAL PARTNERSHIP COOPERATION PROGRAM AT FEP "ZVEZDA"

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The largest in the Far East Russia Far East Plant "Zvezda" (FEP "Zvezda") is the only enterprise involved into complex decommissioning of Nuclear Submarines (NS) in the region wherein are ensured both a high level of nuclear, radiation and environmental safety and adequate physical protection of nuclear materials.

So far 28 out of 35 retired NSs of the Pacific Fleet have been dismantled at FEP "Zvezda" (i.e. 80%).

Dismantlement and cutting of one NS, design 667, at FEP "Zvezda" takes presently only 6 months thanks to the establishment of up-to-date infrastructure facilities, which construction lasted for almost 7 years. The infrastructure comprises the following three unique complexes:

1. Facilities for NS hull cutting and dismantlement-product processing;
2. Liquid radioactive waste processing barge and sectors for low-level radioactive waste conditioning and storage; and
3. Spent Nuclear Fuel (SNF) on-shore defueling facilities.

These complexes are the only Far East Russia's facilities, which successfully underwent Governmental Environmental Impact Assessment and were accepted to operation by the State Acceptance Commissions.

That unique infrastructure was established under two international agreements which preceded the G8 Global Partnership Cooperation Program - the Agreement of June 17, 1992 between the Government of the Russian Federation and the Government of the United States of America on the Cooperative Threat Reduction Program (CTR Program) and the Agreement of October 13, 1993 between the Government of Japan and the Government of the Russian Federation Concerning Cooperation for the Elimination of Nuclear Weapons Reduced in the Russian Federation and on Nonproliferation. The infrastructure cost made up 1.7 billion rubles. Thanks to those facilities 11 NSs were defueled and dismantled at FEP "Zvezda" during 1999–2002.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 107–109.

Presently the Program for strategic-missile NS dismantlement in the Far East Russia is nearing completion, one more NS being dismantled in 2003. Nevertheless, in 2003-2004 the US Department of Defense continued funding of the delivered equipment servicing and has allocated funds for extension of the temporary storage pad for containers with SNF and the establishment of the second stage of physical protection facilities at the SNF defueling complex.

In the context of further development of the 1993-year "Agreement between the Government of Japan and the Government of the Russian Federation Concerning Cooperation for the Elimination of Nuclear Weapons Reduced in the Russian Federation and on nonproliferation and the Establishment of a Committee on this Cooperation", in July 1999 (i.e. 5 years ago) by a special resolution of the Board of Directors of the above Committee, a decision was taken on finding three projects including:

1. Complex decommissioning of general-purpose NS #671RTM; and
2. Reconstruction of a railway section.

The resolution was approved by a special Decree #1271 of the Russian Federation Government of February 8, 2000.

Unfortunately, only one project – complex decommissioning of general-purpose NS #671RTM, serial number 304 – is being presently implemented. It should be also pointed out that the negotiations on signing the contract for work execution were preceded by long-lasting (over 1.5 years from November 2001 till June 2003) coordination of an interim Executive Agreement between the Ministry for Foreign Affaires of Japan and the Russian Ministry for Atomic Energy (RF Minatom) and the Russian Ministry for Foreign Affairs. After signing of the Executive Agreement in June 2003 the negotiations began and finally on December 5, 2003 the Contract for dismantlement of NS, serial number 304, was signed. Thus the period from taking the decision on funding the general-purpose NS dismantlement to signing of the relevant contract lasted for 4 years and 6 months, whereas NS dismantlement itself including development of the relevant project documentation took only 9 months and has been already completed. Thus under the US-Russia CTR Program three NSs have been dismantled per year on average, whereas dismantlement of only one NS under the Russia-Japan Agreement took 5 years.

No executive agreement on railway section reconstruction has been signed yet between RF Minatom and the Ministry for Foreign Affairs of Japan. Moreover, in 2004 the Japan Side renounced the project funding. In 2004 RF Minatom passed to the Ministry for Foreign Affairs of Japan a list of 6 projects proposed for implementation under the Intergovernmental Agreement of October 13, 1993 and the G8 Global Partnership Cooperation Program. However only one project of this list – the dismantlement of one NS, design 671, serial number 614 at FEP "Zvezda" – is considered by the Japan Side as the priority project.

Despite the positive cooperative experience collected so far in the course of dismantlement of NS, design 671 RTM, serial number 304, the Japan Side demanded the development of a new Feasibility Study (FS) for dismantlement of NS #614.

According to the Japan Side's time schedule, the FS is to be completed by November 2004; signing of the Executive Agreement for the Contract preparing is scheduled for December 2004, and the Contract itself for dismantlement of NS #614 shall be signed in March 2005. However with consideration for the experience of negotiations with the Ministry for Foreign Affairs of Japan collected so far, signing of this Contract in the first half of 2005 would be a rather good result.

In 2004 Australia joined the G8 Global Partnership Cooperation Program and pledged $7 million for complex decommissioning of NSs in the Far East Russia. Unfortunately, for lack of the relevant intergovernmental agreements between the Russian Federation and Australia, the above funds were transferred to the account of Technical Secretariat of the Russian-Japanese Committee. Consequently, they are to be spent only on agreement with the Ministry for Foreign Affairs of Japan, and so far no decision on the Australian pledge has been taken yet.

At the same time in the Northwest Russia only in 2003 contracts for dismantlement of six NSs were signed.

From the above the following conclusions may be reached:

1. Implementation of the G8 Global Partnership Cooperation Program in the Far East Russia has not been deployed yet; and
2. So far the available capacities of the complex-decommissioning infrastructure to be involved into the G8 Global Partnership Cooperation Program have been insufficiently used.

In our opinion, to enhance the international cooperation efficiency in the complex decommissioning of nuclear submarines taken out of service in the Russian Far East, signing of an additional executive agreement for dismantlement of several NSs and of contracts for dismantlement of 4 general-purpose NSs per year would be appropriate.

THE ARCTIC MILITARY ENVIRONMENTAL COOPERATION (AMEC) PROGRAM'S ROLE IN THE MANAGEMENT OF SPENT FUEL FROM DECOMMISSIONED NUCLEAR SUBMARINES

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Background – What We Have Accomplished So Far

The Arctic Military Environmental Cooperation (AMEC) Program was established in 1996 when the AMEC Declaration was signed by the Ministers of Defense of Russia, Norway and the U.S. Secretary of Defense. AMEC provides a forum for Russia, Norway, and the U.S. to collaborate in addressing military-related environmental concerns in the arctic region. Of special concern were the large quantities of unsecured spent nuclear fuel from decommissioned submarines that threatened the fragile arctic environment in the Murmansk region.

As a first step in developing a program plan, each country designated teams: for Russia, the Ecological Security Directorate from the Ministry of Defense coordinates with the Russian Navy; in Norway the Ministry of Defence works with the Defense Research Establishment and in the U.S. the Department of Defense coordinates with the Navy. Each team consists of a Principal for overall leadership, a Steering Group Co-Chair for the day to day leadership and Project Officers and Technical Experts as appropriate. Within the U.S., the Navy works with Project Officers from the Department of Energy and the Environmental Protection Agency and coordinates closely with the Department of State.

During the first year a program plan was developed to address both radiological and non-radiological problems in Northwest Russia. Radiological projects were planned in five program areas: spent nuclear fuel management; liquid waste treatment; solid radioactive waste processing; radiation monitoring and personnel safety. Priority consideration was given to the management of spent nuclear fuel. Projects are focused on technology demonstrations.

Spent Nuclear Fuel Management

Five projects were initiated to manage spent nuclear fuel: the development of a dual-purpose transport and storage cask; a cask trans-shipment facility; a cask de-watering and fuel drying system; a centralized radio-ecological monitoring system for the cask

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trans-shipment facility and spent nuclear fuel handling, transfer and auxiliary equipment technologies.

Figure 1. SNF CASK

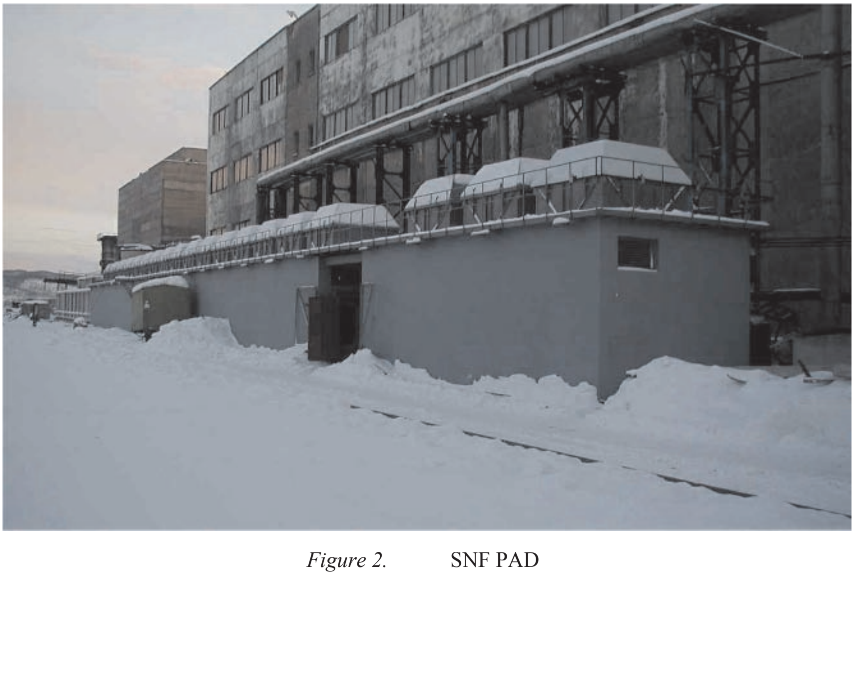

Figure 2. SNF PAD

SPENT NUCLEAR FUEL CASK

The only cask available at the time was designed for transport only. A cask was needed that could be used for transport and long term storage – up to fifty years. The design and fabrication of the AMEC prototype cask was completed in 1999. The cask is metal-concrete and can hold up to 49 undamaged fuel assemblies. When loaded the cask weighs 40 metric tons - ten to twelve cask can hold the fuel from one nuclear submarine. Testing demonstrated that the cask can be handled with existing equipment. The cask was assigned the design classification TUK-MBK-VMF and the prototype cask was designated TUK-108/1. Serial production started in 2002 and when serially produced the cask costs 80% less than the container used previously.

SPENT NUCLEAR FUEL TRANS-SHIPMENT FACILITY

A trans-shipment facility was needed at FGUP Atomflot in Murmansk , since it is the only site in the Kola Peninsula for rail transport of the SNF to Mayak. Considerable amount of time was lost either in waiting for the arrival of the service ship that transports the spent fuel from the de-fueled submarines or the special railcars for the transport to Mayak - there was no facility to store the casks temporarily. Construction of the trans-shipment facility was started in 2000 and completed in 2003. The facility consists of a reinforced concrete foundation plate that can hold nineteen TUK-108 casks vertically with walls to protect workers in nearby facilities and it has a 50-year service life. The pad is designed to use existing handling equipment to transfer the spent nuclear fuel from the service ship to the railcar and can accommodate both the TK-18 transport cask and the TUK-108 transport/storage pad. Completion of the facility was delayed by almost two years due to jurisdictional disputes between Gosatomnadzor (GAN), the Russian civilian regulatory authority, and MOD GAN, the military regulatory authority, regarding the relative roles and responsibilities for transport and handling of the SNF using both military and civilian equipment as well problems identifying and obtaining all required Russian clearances and licenses to operate the storage pad. The pad is still awaiting final documentation before going into full production.

CASK DE-WATERING AND FUEL DRYING

This project was initiated in 2002 to extend the service life of the TUK-108 casks by removing the residual water and conditioning the fuel - current procedures allow up to 3.2 liters of water in the cask before sealing. The residual water increases the risk of corrosion and hydrolytic gas production when stored for long periods. Preliminary designs were developed but the project was stopped in 2003 when the Cooperative Threat Reduction Program announced plans to build a large dehydration facility at Mayak. The design of this facility is completed but funding for the actual construction of the facility was cancelled.

CENTRALIZED RADIO-ECOLOGICAL MONITORING SYSTEM

Radio-ecological monitoring at the trans-shipment facility is accomplished with a centralized radiological surveillance system, the Picasso Environmental Monitoring system developed by the Institute for Energy Technology (IFE), Halden, Norway. The Russian Institute for Nuclear Safety (IBRAE) modified the system for use at Russian facilities and integrated Russian manufactured terrestrial and underwater gamma detectors, smart controllers and radio-modems for off-site transmission of data. The centralized server to process the real-time activity of the site is integrated into the FGUP Atomflot central control room. The system recently successfully completed its operational test phase and is now fully operational.

SPENT NUCLEAR FUEL HANDLING, TRANSFER AND AUXILIARY EQUIPMENT TECHNOLOGIES

The goal of this project is to identify specific auxiliary equipment that would increase the cost effectiveness and safety of spent nuclear fuel handling and transport. A Concept Level Proposals has been completed and approved by the AMEC Principals, but funding has not been identified for this project.

SUMMARY

AMEC's spent nuclear fuel management projects have achieved the following results: storage and transport of SNF has been improved through the use of the AMEC developed cask and the transshipment pad at RTP Atomflot in Murmansk RF. The cask is Russia's first "dual use" – transport and storage cask – while the pad reduces the SNF de-fueling from nuclear submarines from 3 months to 3 weeks according to Russian Officials. The cask is now in serial production by the Cooperative Threat Reduction program to be used for the spent fuel from dismantled ballistic submarines, and Russia is using the cask for the SNF from the nuclear submarines that they are dismantling. The pad is awaiting final documentation before it can go into full operation. Security of the SNF transshipment pad is further enhanced through remote monitoring for both radiological and ecological conditions and the personal safety of radioactive waste workers has been improved through training and US and Norwegian supplied dosimeters.

Where We Are Now — Ongoing Projects

Several decommissioned Russian nuclear submarines were reported to have sunk pier-side while awaiting dismantlement and many have suffered serious loss of buoyancy. Polystyrene has been injected into the ballast tanks of about 17 decommissioned submarines to keep them from sinking. The sinking of the K-159 nuclear submarine while being towed to the dismantlement site underscored the importance of safe transport of these submarines. Two projects are currently ongoing to address the issues of buoyancy of decommissioned submarines and the safe transport to the dismantlement site.

BUOYANCY OF DECOMMISSIONED NUCLEAR SUBMARINES

This project was proposed in 2003. It is the first project for which the United Kingdom has taken the project lead – they joined AMEC in June 2003. The tasks for this project include the injection, removal and reprocessing of the Polystyrene. The project definition stage is almost complete and it seems to the donors that the most pressing problem is the removal of the Polystyrene – a variety of methods are currently being investigated.

SAFE TRANSPORT OF DECOMMISSIONED NUCLEAR SUBMARINES

The goal of this project, which was started in 2003, is to develop the means for safe transport of the submarines to their dismantlement sites to avoid accidents such as the sinking of the K-159 which sank while under tow. During the project definition phase over 28 different options have been evaluated, this has now been reduced to 4 options.

Figure 3.

Priority consideration has been given to the completion of this project within the next year.

DISMANTLEMENT OF A NUCLEAR SUBMARINE AT A RUSSIAN NAVAL SHIPYARD

The goal of this project is to dismantle a nuclear submarine at Shipyard 10 in Polyarnyi and to demonstrate new techniques that will reduce the hazardous waste produced during the dismantlement process. An assessment of the existing infrastructure at the shipyard is planned for the end of September 2004.

Where We Are Going – Future Strategic Direction

As AMEC was completing its initial set of projects and new projects were considered, two events helped shape AMEC's future focus. In the aftermath of September 11, 2001 – the terrorist attack on the world trade center – the view of risks and threats changed substantially in acknowledgement of changes in terrorist target selections and methods. Risks posed by decommissioned Russian submarines grew from being strictly an environmental issue to include diversion and theft of fissile and highly radioactive materials. Accidents at sites where large amounts of poorly maintained and protected SNF and radioactive waste accumulated were viewed as more likely because such sites could become terrorist targets. In June of 2002, the G8 announced a new initiative "The Global Partnership Against the Spread of Weapons and Material of Mass Destruction". The G8 countries pledged $20B over the next ten years to fund nonproliferation, disarmament counter-terrorism and nuclear safety issues. Among the top priority concerns are the dismantlement of nuclear submarines activities in the former Soviet Union.

As a consequence of the new view of risks and threats posed by decommissioned Russian nuclear submarines AMEC is developing a new Strategic Program Plan that reaffirms the goals of the AMEC Declaration and supports emerging global issues. The objectives of the plan are to carry out collaborative research and demonstration of technology to address environmental security, non-proliferation and threat reduction in addition to conventional environmental issues. The plan focuses AMEC activities on the following areas of cooperation:

- nuclear security issues in support of the G8 Global Partnership priorities;
- nuclear submarine dismantlement and issues associated with decommissioned military nuclear-powered vessels;
- management of hazardous waste generated as a result of military activities;
- environmental sustainability, safety and security.

In the period to 2010, it is anticipated that the emphasis will be toward the program areas of nuclear security, nuclear submarine dismantlement and the management of hazardous waste. The nuclear submarine dismantlement program is expected to be well advanced by 2010, and both technology gaps and infrastructure will have been largely addressed by this time. The areas of cooperation over the period to 2015, are expected to focus on environmental sustainability, safety and security of military activities.

REMAINING ISSUES include the following:

Legal Issues**:** The partner countries have used their own bi-lateral legal agreements to implement projects with the appropriate legal coverage. However, this arrangement has difficulties that have been overcome in the past but has had an impact on cost and schedule. A common legal agreement between the parties is needed to standardize the terms and conditions under which the projects are implemented.

Expansion. The AMEC partner countries have agreed that other Nations can participate in AMEC at the project level by providing resources and technical expertise, a concept known as “AMEC Plus”. This participation can be extended to a more formal level by inclusion of the nation as a member country or party to the AMEC Declaration. The AMEC partner countries also support the concept of expanding the geographical area to include Russia’s Far East. However, this would require the establishment of a separate administrative structure so that the current resources of the partner countries would not be diluted.

Consistent with the timeframes of the G8 Global Partnership Initiative, it is anticipated that new projects would not be started after 2013, thus leaving several years to complete remaining projects by 2015.

CONCEPT OF COMPLEX DECOMMISSIONING OF CIVIL NUCLEAR POWERED SURFACE SHIPS AND MAINTENANCE VESSELS

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1. Actual condition of civil nuclear fleet

Both the Russian and the world practice of navigation at freezing seas have confirmed major importance of icebreaker fleet for safe and efficient navigation in ice conditions. Construction of Russian Nuclear IceBreaker (NIB) fleet was of crucial importance for large-scale exploration of the Arctic region and establishment of regular navigation at the Russian Arctic seas. The collected so far positive experience of running as well as feasibility calculations and the related assessments have shown that stable and efficient running of the Russian "Northern Seaway" is only possible if the Arctic NIB fleet is used.

Being in federal ownership of the Russian Federation, Russian nuclear icebreaker fleet is under asset management of "Murmansk Shipping Company" (MSC) Public Corporation.

To date the Russian civil nuclear fleet comprises: 8 nuclear icebreakers and one transport nuclear-powered ship – lighter-aboard container ship "Sevmorput" with one-reactor Steam Producing Installation (SPI). In addition, some Maintenance Vessels (MVs) are used for maintenance purposes: -as floating storages of fresh and spent nuclear fuel; -in supporting operations on reloading of both nuclear fuel and ion-exchange filters; -receiving, storage and processing of Radioactive Waste (RW); -dose control activities, etc.

By now practical solution of the tasks related to complex decommissioning of civil nuclear ships has become rather topical. So far of 8 NIBs of the Russian nuclear fleet:

– "*Lenin*" NIB has been withdrawn from service due to poor technical condition;
– "*Sibir*" NIB has been stored afloat since 1992 due to the need of complex overhaul;

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 119–129.

and

– SPI of "*Arctica*" NIB has expired its design service life (100 000 hours) by March 2000.

Because of expired SPI resource (100 000 hours), still operating Russian nuclear NIBs should be taken out of service as follows:

– Routine-run NIBs with 2-reactor SPIs:

"*Russia*"	in 2004
"Sovetskiy Soyuz" ("Soviet Union")	in 2006
"*Jamal*"	in 2009;

– Restricted-draft NIBs with one-reactor SPIs:

"*Taimyr*"	in 2004
"*Vaigach*"	in 2005.

SPI resource of "*Sevmorput*" lighter-aboard container ship will be expired by 2010. Some MVs are also to be taken out of operation and dismantled in the near future (first "*Lepse*" FSV with Spent Nuclear Fuel (SNF) storage and "*Volodarskiy*" FSV with Solid Radioactive Waste (SRW) storage).

Accelerated paces of NIB taking out of service cause the risk of disrupting stable navigation through the "Northern Seaway" in the near future: for normal running one needs 5 NIBs including 3 routine-run NIBs and 2 restricted-draft NIBs (in addition, because of the need of repair, servicing and reactor fuel reloading, one more NIB has to be permanently out of operation).

Thus, since 2000 Russia has faced a shortage of NIBs potentially leading to reduction of the Arctic navigation season and even to its complete closing firstly during winter-spring period and next in summer months.

"*50 Years of Victory*" NIB – completing the "*Arctica*"-type NIB series – has been constructed at Baltic shipyard since 1989: its commissioning is expected in 2006 (at the best) if adequate financing is provided.

7-8 years of work are necessary to design and construct a lead new-generation NIB (all-purpose two-drift NIB) for replacing NIBs of "Arctica" and "Taimyr" types.

The collected so far running experience of "*Lenin*" (107 000 SPI running hours, service life 30 years) and "*Arctica*' (145 000 SPI running hours, service life 25 years) NIBs, actual condition of NIB systems and equipment, preliminary investigations by the NIB designers and examinations of main NIB equipment have allowed the following conclusion: there is a possibility for postponing the icebreaker "shortage" via prolongation of SPI recourse from 100 000 to 150 000 running hours and the relevant extension of NIB service life. For this purpose the following documents have been developed: -a Program of NIB SPI resource prolongation up to 150 000 running hours and service life extension up to 30 years and -the related Target Program "Nuclear Icebreakers" providing for implementation of a variety of technically and economically justified measures on extending service life of the actual NIB fleet for a period of reliable and safe operation of nuclear SPI – the most important NIB element with a predetermined resource.

If SPI resource were prolonged from 100 000 to 150 000 running hours (in case of "*Arctica*" NIB from 145 000 to 175 000 running hours), NIB service life could be extended by 6–8 years; in case of SPI resource prolongation up to 175 000 running hours – by 10-12 years. Thus implementation of the above Programs would allow extending service life of the actual NIBs until 2010-2015.

If the measures provided by the "Nuclear Icebreakers" Target Program are fulfilled successfully and opportunely, "*Arctica*" NIB (expiration of "prolonged" SPI resource - 175 000 running hours - in 2006) will be the first actually running NIB to be withdrawn from service.

The very first civil nuclear vessel "*Lenin*" NIB constructed in 1959 was taken out of operation in 1991 and is presently stored afloat at Federal State Unitary Enterprise (FSUE) "Atomflot". When preparing the icebreaker to protracted waterborne storage, its in-dock overhaul was performed, which included: -dismantlement of propeller shafts and rudder propeller and -sealing of deadwoods, outboard fittings and ice trunks ##3, 4, 5 and 6. The structures and equipment necessary for safe protracted waterborne storage of the icebreaker were thoroughly examined. According to the results of "*Lenin*" NIB examination (1991), the resource of its equipment and structures (including main survival and fire safety systems) was recognized as sufficient for safe waterborne storage over the next 10–15-years.

Prior to placing in dock, "*Lenin*" NIB reactors were defueled, and shielding assembly was unloaded from Reactor #1. Electromotors of the primary coolant pump were made dead, automatic power supply was cut off and sealed, drives of control rod groups and actuators of emergency protection system were dismantled and removed from both reactors. The primary circuit was filled with bi-distilled water under a pressure of 15 kgf/cm^2 (reactors #1) and 9 kgf/cm^2 (reactor #2). The third-circuit system was filled with atmospheric-pressure water.

SRW was removed from the NIB's storage and transferred to "Atomflot" for processing and disposal. To improve the radiation situation at the icebreaker, all reactor compartment rooms were decontaminated.

During waterborne storage the following systems and equipment of "*Lenin*" NIB are operable being in "stand-by" condition:

- utility systems;
- reactor compartment ventilation system;
- survival (fire safety, ballast, drainage, high-pressure air, low-pressure air and discharge) systems;
- reserve diesel generator;
- emergency diesel generators;

- auxiliary boiler installation;
- radiation control system; and
- thermotechnical control gears.

During the early waterborne-storage period permanent crew of retired "*Lenin*" NIB consisted of 47 people; then it was reduced to 25 people.

Though "*Sibir*" NIB has not been formally taken out of operation yet, due to poor technical condition of steam generators and the pressurizer equipment no decision on their resource prolongation can be taken without major repair. "*Sibir*" NIB is presently stored afloat on the shipyard berth. The NIB has been already defueled and is maintained by a minimal crew of 42 people.

On the assumption of average statistical 6 000 SPI running hours per year, presently-operating NIBs are to be taken out of service as follows: "*Sovetskiy Soyuz*" in 2014 and "*Jamal*" in 2017; the restricted-draft NIBs with average statistical ~ 8 000 SPI running hours ("*Taimyr*" and "*Vaigach*") are to be withdrawn from service in 2010 and 2011, respectively.

2. Main factors influencing the choice of the concept of complex decommissioning of nuclear-powered surface ships (NPSS)

To date the lack in Russia of a specialized ship-cutting enterprise capable of performing the whole dismantling and cutting cycle including subsequent placing of Reactor Compartment (RC) units for protracted storage is the major factor hindering practical implementation of NPSS complex decommissioning process and determining - to a considerable degree - the decommissioning concept.

The whole cycle of works related to complex decommissioning of civil nuclear vessels needs to be based on agreed and coordinated inter-agency decisions concerning: industrial capacities to be used, reactor compartment transport and management flowsheets (including RC placing for protracted storage) and management of produced radioactive waste. The choice of basic technology and procedures of NPSS withdrawal from service are considerably complicated by lack of unambiguous decisions on protracted storage of RC units containing reactor equipment. At present protracted waterborne storage of retired NPSS at "Atomflot" water area (similar to that of "*Lenin*" NIB) seems to be an inevitable decommissioning phase.

Among the main challenges to deal with, one needs to:

- Determine industrial and technological basis for implementation of works on cutting of hulls of NPSS and MVs and making up of RC units containing reactor equipment;
- Develop - coordinated at the inter-agency level - initial technical requirements on a multi-purpose storage facility for RCs units of former NPSS and SRW (including high-active elements of MVs);
- Establish initial technical requirements on transport and management procedures of

RC unit (containing reactor equipment) haulage and placing for protracted (permanent) storage;

– Estimate the amount of expected radioactive waste during NPSS complex decommissioning;
– Estimate the overall dimensions and masses of RC units containing reactor equipment; and
– Select an optimal option on removing uranium-zirconium SNF from storages of "*Lotta*" and "*Imandra*" FSVs and damaged SNF from storage of "*Lepse*" FSV.

To perform complex decommissioning of NPSS and MV, development and putting into action of a special Target Program is necessary. Such program should comprise: -a list of civil nuclear vessels to be taken out of service with indication of their expected withdrawal dates and -a list of activities to be implemented under the program.

The objectives put by are listed below:

– Determine necessary scope of work, the contents of R&D to support complex decommissioning of Russian nuclear civil fleet and identify participants of works;
– Determine the scope of the branch's standard basis necessary to perform complex decommissioning of NPSS and MV;
– Identify the scope of standard documentation for enterprises/storage facilities concerned with complex decommissioning;
– Determine the contents of works on developing industrial and technological basis and identify participants of works; and
– Determine the contents and dates of preparatory works on NPSS and MV withdrawal from service and identify participants of works.

When addressing industrial and technological tasks, one needs to:

Determine an enterprise (shipbuilding plant, shipyard, FSUE "Atoflot") that could carry out works on complex decommissioning of NPSS and MV;

Perform preliminary analysis of industrial and technology potential of the selected enterprise with estimate of work terms and costs; and

Develop proposals on developing industrial capacities and enterprise procurement with additional equipment.

To solve the tasks related to storage of RC units comprising reactor equipment and SRW, one needs:

– Developing agreed at the inter-agency level initial technical requirements for multi-purpose storage facility of cut out RCs and SRW (including high-active elements of MVs);
– Determining participants of works on storage facility designing and construction.

When dealing with RC transportation and management, one needs:

– Developing initial technical requirements to transport facilities and equipment for protracted (persistent) storage of cut out RC units of former NPSS and SRW (including high-active elements of MVs); and

- Determining participants of works on development of RC unit transport and management cycle, designing and construction of the relevant equipment.

Adequate solution of the problem of SNF storage at "*Lepse*" FSV and completion of updating of physical protection system at all NPSS and MV, especially on the water area side, should be also considered among the top-priority tasks.

3. Comparative assessment of basic technological options of complex decommissioning of civil nuclear vessels

Due to lack of specific decisions on NPSS dismantlement and cutting at specialized enterprises, their protracted waterborne storage at "Atomflot" is presently considered as an inevitable phase of complex decommissioning.

So far several options of NIB complex decommissioning using industrial capacities of "Atomflot" - that previously performed only servicing and repair of Russian civil ships - have been developed.

The following alternatives of managing RC equipment of dismantled NPSS are under consideration:

Option #1 – partial dismantlement of equipment and constructions and NPSS preparing to protracted (10-15 years) waterborne storage;

Option #2 – dismantlement and separated unloading of main structures from RC and SPI equipment from Metal-Water Shielding Tank (MWST), the latter being taken out separately after dismantlement of reactor equipment;

Option #3 – dismantlement and unloading of SPI equipment from RC together with MWST; and

Option #4 – ship cutting within a floating dock and making up of a RC unit complying with the requirements for protracted waterborne storage.

OPTION #1

NIB preparing to protracted (10-15 years) waterborne storage is conventionally considered as a separated option because the condition of NIB prepared to waterborne storage is the "initial" one for subsequent works under options 2-4. In the present-day situation conducting of works under Option #1 seems inevitable.

The performed analysis has demonstrated that complex decommissioning of "*Lenin*" and "*Arctica*" NIBs using actual capacities of "Atomflot" would be hardly possible without important capital investments. Radical solution of this challenge, as applied to civil and naval NPSS, requires construction in the Northwest Russia of a specialized ship-cutting enterprise capable of supporting the whole cycle of works related to RC management. At the first decommissioning phase one of presently operating shipbuilding enterprises specialized in construction of nuclear vessels could be temporarily used for such purposes.

Prior to take an ultimate decision, it would be wise to content oneself with works under Option #1 providing for protracted storage of NIBs at "Atomflot" water area.

All works on NIB preparing to protracted waterborne storage, including in-dock operations, could be performed at "Atomflot". During that period "Atomflot"'s personnel could also execute some dismantling and pre-cutting operations. However in such a case the enterprise's wharfage would need major reconstruction.

During waterborne-storage period every NIB should be provided with power supply, steam, water, compressed air and telephone communication from the coast. Factual data on the scope of work performed at the retired "*Lenin*" NIB and its actual condition are addressed in para 1 of this paper.

On board of every NIB prepared to protracted (10-15 years) waterborne storage appropriate conditions should be created for permanent staying of a minimum-necessary crew. For this purpose a variety of restoration works should be performed to ensure self-contained protracted afloat storage. Some works should be performed in dock conditions (e.g., dismantlement of rudder propeller complex, sealing of hull undersluices (save for sewage system sea inlets) and protection painting of the underwater parts of ships). In the course of such works a part of NIB equipment should be dismantled and unloaded provided that appropriate trim, stiffness and floodability of partly dismantled NIB are kept. All works should be performed in compliance with the NIB designers' documentation. Technical proposals on NIB haulage by sea to long-term storage center/dismantling & cutting enterprise should be also developed.

In any case a huge scope of works is to be placed on "Atomflot" to be performed in several stages providing for repeated re-mooring of retired NIBs to different berths of the enterprise depending on the type of individual operations and with consideration for routine works on civil nuclear vessel repair and servicing performed by "Atomflot".

An option of "cold" protracted waterborne storage of NIB allowing reducing the related expenses could be also considered. In such a case, in addition to NIB defueling, one should also remove sorbents from filters of SPI primary and third circuits, discharge and seal all SPI process circuits including LRW tanks (circuit waters and sanitary room waters). After that no crew would be necessary on NIB board. Inventory and dismantlement of easily removable equipment and fittings potentially re-usable (as spare pieces) at NIBs still in operation would be also appropriate. Openings caused by such dismantling operations should be sealed or welded.

RC rooms should be decontaminated, and the radiation parameters be measured and mapped. RC power supply should be cut off, and all RC rooms be locked and sealed.

During waterborne storage of retired NIBs adequate physical protection (to exclude unauthorized actions including theft, especially of radioactive constructions) and regular control over outboard water inflow (to prevent the risk of buoyancy loss and NIB sinking on the berth) must be ensured.

OPTIONS #2 AND #3

Unloading of SPI equipment from RC is performed either on specially equipped on-shore near-berth pad or on special floating facilities followed by making up of RC unit to be next towed and placed for long-term storage.

As such floating tanks, purposefully reequipped lighters (type "DM", carrying capacity 1100 t) serving in the lighter sea carriage could be used. The lighter has been certified in the Russian Sea Navigation Register as a barge with double boards and case-shape hull.

If using "DM"-type lighters for purposes of SPI equipment transportation and placing for long-term storage, such lighters should be adequately reequipped and provided with adequate biological shielding. Three compartments could be made up in the lighter cargo hold via installation of transverse watertight bulkheads, the nose compartments and the stern compartments being separated from the central compartment by cofferdams.

OPTION #4

Thanks to "Atomflot" procurement with a floating dock, design 'V960', reequipped in compliance with the requirements on in-dock repair of NIBs and MVs, a possibility emerges on applying a fundamentally new NIB complex decommissioning technology proposed by NIB designer ("Iceberg" Central Design Bureau), i.e. Option #4.

The dock-reequipping designer (the Western Design Office) has already performed the relevant estimates for 'V960'design floating dock reequipping.

The following process operations should be provided:

– NIB hull cutting into three parts, the section lines being beyond RC (along cofferdams); RC cutting is to be performed across the width;
– mounting of additional constructions to improve buoyancy and perform haulage of the cut out parts of NIB hull; sealing of the cut out parts of NIB hull.

Installation of the following "additional constructions" is planned:

– pontoons arranged on the end nose and the end stern bulkheads of RC (in case of "*Lenin*" NIB such a pontoon has: length – 12 m, width – 25 m, height – 8 m); and
– transverse watertight bulkheads installed one by one at the nose part and the stern part of the hull (according to estimates, such works would require ~18% of total working hours for in-dock operations).

Sealing of all openings and installation of additional equipment to perform haulage of individual parts of NIB hull are also envisaged.

The cut out and extra-equipped parts of NIB hull are to be removed from the dock in a condition allowing their sea towing: -RC - to storage center; -nose/stern compartments - to scrap metal enterprise.

According to calculations performed with reference to "*Lenin*" NIB, the displacement of NIB parts after cutting would make up :

- RC 3700 t,
- Nose part 4400 t, and
- Stern part 4000 t.

RC overall dimensions would be: length – 30 m, width – 28 m, height – 29 m.

However management of RC units of such huge dimensions would pose major problems related to not only their transport and management cycle but also to construction of special storage facility. As mentioned above, such problems should be resolved at the inter-agency level taking into consideration the forthcoming large-scale complex decommissioning of naval NPSS with similar dimensions of RC units.

If necessary, when placing a RC unit for long-term storage, its height could be diminished through cutting off some elements of the biological shielding system. The cut out metal could be used when sealing RC or watertight bulkheads. Prior to RC sealing, the cut out elements of composite biological shielding system (with known radioactive contamination and unused as structural elements) could be placed into reactor room.

To reduce the amount of produced RW, contaminated equipment and elements of contaminated constructions could be placed into reactor room and other rooms of the cut out RC unit.

For every above-considered option of NPSS complex decommissioning the following main indices were compared: labor expenditures (including radiation-hazardous operations), number of involved operational personnel, work cost (in relative units), collective exposure dose for personnel and amount of generated SRW (Table 1).

TABLE 1. Comparative estimate of main indices of NPSS complex decommissioning under different options

Option	1	2	3	4
Labor-intensiveness:				
thousand norm-hours	374.5	537.1	517.7	534.3
relative units	1.0	1.6	1.5	1.52
including radiation-hazardous operations:				
thousand norm-hours	69.6	138	118	83.1
% of the total work scope	19	26	23	16
relative units	1.0	2.2	1.82	1.28
Number of involved operational personnel:				
people	175	250	241	249
relative units	1.0	1.43	1.38	1.42
Work cost:				
relative units	1.0	2.0	1.82	1.54
Collective dose:				
rem	30	360	120	70
SRW amount				
integral volume, m^3,	275	585	465	307
containers 1.5 m^3 in volume, pieces	184	390	310	205
relative units	1.0	2.13	1.7	1.12

As follows from the above table data, Option #2 and Option #4 differ only slightly from each other when considering labor-intensiveness, which exceeds that of Option #1 by 1.5 – 1.6 times. The number of involved operational personnel under Options #2 and #4 varies within 241-250 people. Note that under Option #2 and Option #3 works related to re-equipment of lighters were not taken into account.

The proportion of efforts on reactor equipment dismantlement and RC unit making up (category of "radiation-hazardous operations") for Options 2, 3 and 4 are estimated as 26, 23 and 16% of the total efforts, respectively.

The distinctions in work cost under Options 2 – 4 at virtually the same level of efforts are due to different scope of radiation-hazardous operations and differences in expenses related to RW management. The results of work-cost calculations given in Table 1 were obtained using average weighted values of the cost of estimated norm-hour with reference to the following industrial programs of "Atomflot": -"shipbuilding", "machine-building" and "other works".

Conclusions

1. Practical solution of the challenges related to complex decommissioning of NPSS has become rather topical in recent years. So far "*Lenin*" NIB and "*Sibir*" NIB (being formally still in service) have been stored afloat; standard service life and the resource of main equipment and SPI systems of "*Arctica*" NIB expired in 2000.

   Due to expiration of SPI standard resource (100 000 hours) all presently operating NIBs are to be taken out of service from 2004 to 2009.

Implementation of SPI resource prolongation measures up to 150 000 hours and their service life extension up to 30 years would allow postponing the onset of NIB withdrawal from service until 2010 – 2015.

2. The only option of RC equipment management has been implemented so far providing for preparing of the taken-out-of-service NIB to protracted (10 – 15 years) waterborne storage on the berth of "Atomflot". The whole scope of preparatory works, including in-dock overhaul, is performed at "Atomflot".
3. Appropriate complex decommissioning of civil nuclear ships including processing and disposal of contaminated equipment and materials would be only possible if a special federal program were developed and implemented with the participation of interested agencies. Such a program should provide for construction of a specialized up-to-date ship-cutting enterprise to conduct works in compliance with the present-day radioecological safety requirements.
4. Of a variety of complex decommissioning options considered in the paper, the most promising options are: -NIB in-dock cutting and making up of RC unit complying with the requirements on its transfer for long-term storage and – making up of the "nose part" and the "stern part" of NIB prepared to transfer at scrap-metal enterprises for subsequent cutting.
5. After "Atomflot" procurement with a floating dock, design 'V960', a new opportunity has emerged on considerable extension of the scope of work on complex decommissioning of NPSS at "Atomflot" down to making up of floating RC unit. However such works would require important capital investments for further development of the enterprise including extension of work sections, storage facilities, sanitary and utility areas, etc.

UNLOADING, STORAGE AND SUBSEQUENT MANAGEMENT OF SPENT NUCLEAR FUEL OF LIQUID-METAL-COOLANT REACTORS: ACTUAL STATUS AND PROBLEMS

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Abstract

The paper addresses the actual status and the main problems of unloading, storage and subsequent management of Spent Nuclear Fuel (SNF) of Russian Nuclear Submarines (NSs) with lead-bismuth Liquid-Metal Coolant (LMC) in the primary circuit. On this basis some topical tasks related to analysis of the sources of potential radiation releases and risk assessment are formulated and discussed.

1. Introduction

During 1962 - 1997 a number of NSs with lead-bismuth LMC in the primary circuit were in operation in the Russian Navy. Reactors of such NSs developed under scientific management of the Russian Research Center "Institute for Physics and Power Engineering" (RRC IPPE, below IPPE), Obninsk, fall into the category of intermediate-neutron reactors [1]. Fuel composition of LMC reactors comprises intermetallic compound UBe_{13} with ^{235}U enrichment up to 90 % dispersed over beryllium matrix. Some characteristics of such-type NSs are demonstrated in Table 1.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 131–144.

TABLE 1. Some characteristics of NS with LMC

Period of operation	1962 - 1997
Design	645, 705, 705K
Number of reactors on board	2 (645), 1 (705,705K)
Coolant	Pb + Bi eutectic alloy
Neutron spectrum	intermediate
Fuel composition	UBe13
U enrichment	~ 90%
Fuel reloading mode	as a single Spent Removable Unit (SRU)
Actual status	withdrawn from service
Location	Gremikha, Northwest Russia
Expected storage time	from ~3 years to ~5 years
Real storage time	from ~10 years to ~35 years

A specials feature of NS with LMC consists in the following: using a special transport and management devices cores of such NSs are loaded into/unloaded from reactors as a single removable unit comprising the core with inserted rods of the Control and Protection System (CPS), side reflector (designs 645 and 705K) and an upper plug of biological shielding structure [2]. During nuclear fuel unloading from reactors of NS, design 705, the side reflector is not taken out.

2. NS, Design #645

The very first NS with LMC (design #645) was equipped with two-reactor Power Reactor Installation (PRI). After the port-side-reactor accident during the second fuel lifetime (1968), the NS was kept afloat for some period. Then, after filling of free reactor cavities and the whole Reactor Compartment (RC) with preservative agents, the NS was dumped in the Kara Sea close to the Novaya Zemlia (New Land) archipelago at 50-m depth (1981). Some characteristics of NS, design 645, are given in Table 2.

TABLE 2. Some characteristics of NS, design 645

Period of operation	1962 - 1968
First fuel reloading Power output	1967 ~100%
Port side accident Power output Coolant condition	1968 ~10% frozen
Fuel status	non-unloaded
Work on conservation	performed in the late 1970s
Location	dumped in the Kara Sea (~50 m)

3. NSs of Designs 705 and 705K

The remainder nuclear submarines with LMC (designs #705 and #705K known in the West Countries as *Alpha*-class NSs) were equipped with only one reactor. Altogether 4 NSs, design 705 (##900, 905, 910 and 915), and 3 NSs, design 705K (##105, 106 and 107) were built; note that NS #105 functioned in different periods with two Steam Producing Installations (SPI): SPI #120 and SPI #125. So far all NSs with LMC, designs #705 and #705 K have been withdrawn from service, four of them being defueled, whereas two NSs still house fuel in reactors with "frozen" coolant. Two RCs cut out of former NSs (#900 and #105) also house nuclear fuel, their coolant in reactors being also "frozen". The remaining non-defueled NSs with LMC are to be defueled in the future, save for NS # 900.

4. NSs of Design 705

The NS #900 put into operation in 1970 was the very first vessel of designs #705 and #705K. In 1972 because of failure of NS primary circuit's auxiliary pipelines and impossibility of their repair, the NS was taken out of service after expiration of only 10% of fuel lifetime in the reactor. The RC was cut out of NS, and free cavities of the primary circuit were filled with preservative agents on furfurol basis. A bitumen-layer of about 1000 mm was laid over the whole surface of the upper deck of RC including the reactor upper head. Under such condition the RC is to be stored with nuclear fuel for long within the waterborne storage center. Some characteristics of NS, design 705, are demonstrated in Tables 3, 4 and 5.

TABLE 3. Some characteristics of NS #900

Put into operation	1970
Withdrawn from service	1974
Reason	non-nuclear accident
Coolant condition	frozen
Power output	~10%
Fuel status Fuel storage area	non-unloaded afloat inside RC (Saida Bay)
Fuel element condition	intact (according to radiation monitoring data)

TABLE 4. Some characteristics of NS #905 and #915

Put into operation	1978 (#905); 1981 (#915)
Withdrawn from service Reason	1986 (#905); 1989 (#915) motorized resource expiration
Power output	~70% (#905); ~77% (#915)
Coolant condition	frozen
Fuel status Fuel element condition Fuel storage area	unloaded 1989 (#905); 1990 (#015) intact (according to radiation monitoring data) land-based storage (Gremikha)
RC storage area	waterborne storage (Saida Bay)

TABLE 5. Some characteristics of NS #910

Put into operation	1979
Withdrawn from service Reason	1990 motorized resource expiration
Coolant condition	frozen
Power output	~80%
Fuel status	non-unloaded due to hazardous radiation situation on reactor upper head (152Eu and 154Eu in the chamber of mechanisms of CPS rods)
Fuel storage area	waterborne storage in basing area
Fuel element condition	intact (according to radiation monitoring data)

Reactor of NS #910 was "frozen", all absorber rods being completely inserted. In such condition the alloy in shrouds of shim rods and automatic control rods was "frozen". Cables were dismantled from all CPS actuating mechanisms. Operational time of the reactor was about 80% of its lifetime.

A distinctive peculiarity of the NS PRI consists in an increased gamma-radiation level on the reactor upper plate due to a provoked release of europium isotopes (^{152}Eu and ^{154}Eu) forming the absorbing composition of CPS rods into the chamber of actuating mechanisms of three rods at the reactor upper plate level. High gamma levels in the area of CPS actuating mechanisms give no way for their dismantlement and performing other pre-defueling works on the reactor upper head. It is worthy of notice that the RCs of this NS and of other NSs have no surface contamination, whereas the condition of fuel element claddings of the reactor core before SPI "freezing" was "normal" according to the data of instrumental radiation control.

5. NS of Design 705K

NS #105 with SPI Units #120 and #125

SPI Unit #120. SPI Unit #120 of NS #105 was used at the first phase of the nuclear submarine running. In 1982 after the accident caused by ingress of the primary circuit coolant into RC, the RC comprising SPI unit was cut out of the NS vessel. Assembled with additional "nose" and "stern" buoyancy tanks, the RC was launched and since then has been kept afloat.

Separated from the primary circuit with stop valves, the reactor was "frozen" and represents a monolithic "lead-bismuth ingot" pierced with the channels of reactor CPS. All absorber rods were completely inserted, made immobile via electric cable dismantlement, the shim rods and the automatic control rods being in the "frozen" alloy. The operational time of the reactor was ~50%. Since the RC cutting, no inspection of equipment and PRI system (to be put into action when reactor heating before core unloading) has been performed. Before reactor heating up coiled pipelines of the heating system are to be pressurized, subdivided into smaller-size sections and re-arranged to provide for controlled and smooth heating of the "frozen" reactor. It is unlikely that the reactor temperature control sensors are operable; consequently, thermo-sensors of a specially designed tensor-meter measuring device are to be used instead of one of scram rods. Though the appropriate fittings to perform such potentially nuclear- and radiation-hazardous operation were developed and fabricated previously, they were never used before. Keeping these fittings in proper condition at the Navy storehouses seems to be very important.

Thus an important task of pre-defueling operations consists in inspecting both the equipment and systems to be put into action when heating SPI up. If the reactor coiled pipelines keep their operational characteristics, there will be no special problems hindering the reactor heating up and, consequently, unloading of the reactor core of SPI unit #120. According to the data of instrumental radiation control, the condition of the core fuel element claddings before "freezing" was considered as "normal" one. Some characteristics of NS #105 with SPI units #120 and #125 and of NSs #106 and #107 are demonstrated in Tables 6, 7 and 8.

TABLE 6. Some characteristics of NS #105 with SPI unit #120

Put into operation	1977
Withdrawn from service Reason	1982 loss-of-coolant accident
Coolant condition	"frozen"
Power output	~50%
Fuel status Fuel storage area	non-unloaded afloat inside RC (Saida Bay)
Fuel element condition	intact (according to radiation monitoring data)

Unit #125. SPI Unit #125 was used at NS #3105 at the second phase of its running. In 1997 the NS reactor was shutdown, coolant drained, cut out of the primary circuit and "frozen". All absorber rods were completely inserted and made immobile via electric cable dismantlement, scram rods and automatic control rods being in the "frozen" alloy. The operational time of the reactor was about 15% of its lifetime.

Comment. The NS was taken out of operation for the reasons not associated with technical condition of its equipment and systems. The SPI inspection of 2001 revealed that the reactor heating system had kept its operational characteristics, the remainder equipment being also in order. However the standard temperature control system was put out of action with no authorization, all temperature sensors of the reactor being operable. The control systems give no way either of using the standard system of equipment cooling for purposes of cooling of individual reactor elements. Thus at the current phase of pre-defueling operations at SPI units #125 and #120 (with dismantled pump equipment) a decision has been taken on draining water out of lead-water shielding tanks and applying high-temperature sensors for neutron flux control purposes. According to the data of instrumental radiation control, the condition of fuel element claddings in the core prior to its "freezing" was estimated as "normal".

TABLE 7. Some characteristics of NS #105 with SPI Unit #125

Put into operation	1992
Withdrawn from service	1996
Power output	~15%
Coolant condition	“frozen”
Fuel status Fuel element condition	non-unloaded intact (according to radiation monitoring data)
Fuel storage area	waterborne storage in basing area

TABLE 8. Some characteristics of NS #106 and #107

Put into operation	1978 (#106); 1981 (#107)
Withdrawn from service Reason	1990 (both NSs) motorized resource expiration
Power output	~96% (#106); ~87% (#107)
Coolant condition	“frozen”
Fuel status Fuel element condition Fuel storage area	unloaded 1991 (#106); 1992 (#107) intact (according to radiation monitoring data) land-based storage in Gremikha
RC storage area	waterborne storage in Saida bay

6. Actual status of LMC reactor taken out of operation

NS PRI’s were mainly withdrawn from service due to expiration of either motorized resource of basic equipment or power resource of the reactor core. The procedure was realized as follows [3]: first, such NS was transferred to a dry dock of the specialized Coastal Maintenance Base (CMB) for core unloading. Next, at shipyard slip NS hull was cut (all NS compartments, save for RC, were to be recycled). Then RC was cleared from equipment located outside the biological shielding structure, supplemented with the nose and the stern buoyancy tanks and after sealing was filled with nitrogen of a minor excess pressure. Later on such RC was launched and transferred to a long-term waterborne storage center. To store RCs on a solid ground, special supports welded from the light hull outer side were provided. In some RCs free-of-dismantled-equipment space was filled with solid radioactive waste generated during NS cutting and repair operations at the shipyard site. It is in such condition that defueled RCs of four LMC NSs have been stored. In addition, two more non-defueled RCs are stored afloat including those of NS #900 and NS #105 (SPI unit #120). The latest defueling operations were performed in 1992. Since then the specialized reloading CMB has not been used for this purpose. Protracted storage under “non-standard” condition and special (“frozen”) state of the primary circuit coolant are distinctive features of storage of such NSs.

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Reactors of NS #105 with SPI units #120 and #125 and, wherever possible, of NS #910 are to be defueled according to the drawn-up time schedule. Presently after inaction for almost 10 years the CMB faces many problems including:

- breakdown of base boiler installation – the outer source of steam for reactor heating prior to SRU unloading;
- need for cranage repair (2 bridge cranes and 1 portal crane);
- breakdown of the main physical control system; and
- defects of building constructions of the long-term cooldown facility (cracks in walls of the above-surface part of the building).

The unloaded SRUs with SNF are stored within special facilities at Gremikha CMB (Kola Peninsula). The SRUs are placed into special steel containers filled with non-radioactive molten (next solidified) Pb-Bi eutectic. The decay heat in every core is presently below 2 kW. To date land-based storage facilities also house two first-lifetime SRUs of NS #645 unloaded from the reactors in 1967. Their storage conditions are similar to those of SRUs of NSs designs #705 and #705K. So far no replacement of SNF by "fresh" nuclear fuel has been performed at NSs of designs #705 and #705K, the operational time of the reactor cores varying within 10÷100 % of the lifetime.

7. Radiation potential of LMC reactors

Operation of NS PRIs at "energy" levels of power was accompanied by radioactivity accumulation in their cores, adjacent constructions and coolant requiring a special attention when managing SNF [2, 4]. The amount of accumulated long-lived activity at every NS depends on their reactor power output. Radionuclide composition of SNF and calculated estimates of radioactivity levels of its principal components in PRI of NS design 705K (100% core operation lifetime) are demonstrated in Tables 9-15, provided that the core operated at a rated power. Beryllium was used in the core and in reflector as neutron moderator. As the result of two successive reactions: ${}_4Be^9 + n \rightarrow {}_2He^4 + {}_3Li^6$ and ${}_3Li^6 + n \rightarrow {}_2He^4 + {}_1H^3$, under the action of neutrons tritium was generated in the reactor. In addition, tritium generated within CPS absorber rods according to the reaction: ${}_5B^{10} + n \rightarrow 2\ {}_2He^4 + {}_1H^3$ as well as under ternary fission of uranium (Table 13).

TABLE 9. Radionuclide composition of SNF

Fission products in fuel	^{137}Cs; ^{90}Sr, etc.
Actinides in fuel	$^{238,239,240,241}Pu$; $^{241,\ 242m}Am$, etc.
Control absorber rods	^{152}Eu and ^{154}Eu
Fuel and reflector	Tritium
Coolant	$^{207,\ 208}Bi$; ^{205}Pb; ^{210m}Bi; etc.
Materials of equipment and steel structures	^{60}Co; ^{59}Ni; ^{63}Ni; etc.

TABLE 10. Radioactivity of coolant of the primary circuit due to long-lived nuclides, Bq/kg

Nuclide	$T_{1/2}$	Hold-up time after reactor shutdown, year				
		5	10	20	50	100
^{41}Ca	1.35E05 y	7.00E00	7.00E00	7.00E00	7.00E00	7.00E00
^{54}Mn	312.3d	1.69E03	2.73E01	7.12E-03	1.26E-13	-
^{60}Co	5.27 y	1.41E05	7.34E04	1.98E04	3.88E02	5.54E-01
^{58}Co	70.8 d	9.86E-03	1.84E-10	-	-	-
^{63}Ni	100.1 y	2.30E04	2.22E04	2.07E04	1.68E04	1.19E04
^{108m}Ag	127 y	2.80E04	2.73E04	2.58E04	2.19E04	1.67E04
^{113m}Cd	13.6 y	3.18E05	2.47E05	1.48E05	3.20E04	2.50E03
^{119m}Sn	250 d	6.24E03	8.34E01	1.49E-02	-	-
^{204}Tl	3.78 y	9.55E04	3.82E04	6.13E03	2.53E01	2.65E-03
^{205}Pb	1.51E07 y	2.56E03	2.56E03	2.56E03	2.56E03	2.56E03
^{207}Bi	30.2 y	3.16E05	2.82E05	2.24E05	1.12E05	3.54E04
^{208}Bi	3.65E05 y	8.39E04	8.39E04	8.39E04	8.39E04	8.39E04
^{210m}Bi	3.6E06 y	1.76E04	1.76E04	1.76E04	1.76E04	1.76E04
^{210}Po	138.4 d	2.57E06	2.77E02	3.21E-06	-	-
A_{α}, Bq/kg		2.58E06	1.79E04	1.76E04	1.76E04	1.76E04
A_{β}, Bq/kg		1.01E06	7.76E05	5.31E05	2.70E05	1.53E05

TABLE 11. Radioactivity of actinoids, Bq

Nuclide	$T_{1/2}$, year	Hold-up time after reactor shutdown, year				
		5	10	20	50	100
^{232}U	72	3.03E09	2.88E09	2.62E09	1.96E09	1.21E09
^{234}U	2.48E05	3.29E11	3.29E11	3.29E11	3.29E11	3.29E11
^{235}U	7.04E08	9.88E09	9.88E09	9.88E09	9.88E09	9.88E09
^{236}U	2.34E07	2.77E10	2.77E10	2.77E10	2.77E10	2.77E10
^{238}U	4.47E09	2.06E08	2.06E08	2.06E08	2.06E08	2.06E08
^{237}Np	2.14E06	2.43E10	2.43E10	2.43E10	2.43E10	2.43E10
^{236}Pu	2.85	8.89E09	2.64E09	2.34E08	1.62E05	8.74E-01
^{238}Pu	87.75	5.37E13	5.12E13	4.73E13	3.74E13	2.52E13
^{239}Pu	24380	3.40E12	3.40E12	3.40E12	3.40E12	3.39E12
^{240}Pu	6537	8.21E11	8.20E11	8.19E11	8.17E11	8.12E11
^{241}Pu	14.54	1.35E14	1.06E14	6.60E13	1.58E13	1.47E12
^{242}Pu	3.76E05	2.90E08	2.90E08	2.90E08	2.90E08	2.90E08
^{244}Pu	8.2E07	8.70E-01	8.70E-01	8.70E-01	8.70E-01	8.70E-01
^{241}Am	432.8	1.75E12	2.73E12	4.04E12	5.45E12	5.44E12
^{242m}Am	152	2.08E10	2.03E10	1.94E10	1.69E10	1.35E10
^{243}Am	7400	5.02E08	5.02E08	5.02E08	5.02E08	5.02E08
^{243}Cm	28.5	2.01E09	1.78E09	1.40E09	6.74E08	2.00E08
^{244}Cm	18.1	7.68E09	6.35E09	4.33E09	1.38E09	2.05E08
^{245}Cm	8532	5.72E05	5.72E05	5.71E05	5.70E05	5.68E05
^{246}Cm	4820	9.37E03	9.37E03	9.35E03	9.31E03	9.25E03
^{247}Cm	1.56E07	2.55E-02	2.55E-02	2.55E-02	2.55E-02	2.55E-02
^{248}Cm	3.39E05	1.60E-02	1.60E-02	1.60E-02	1.60E-02	1.60E-02
Sum, Bq		1.95E14	1.65E14	1.22E14	6.33E13	3.67E13

In CPS absorber rods in reactors of NS, designs 705 and 705K, the composition including boron and europium was used resulting in rather high levels of accumulated radioactivity of Eu^{152} and Eu^{154} isotopes. It is worthy of notice that radioactivity of europium in absorber rods during 1÷20-year hold-up time contributed at the most to decay heat in reactor core exceeding that of fission products by about 2 times.

TABLE 12. Radioactivity of europium in CPS rods, Bq

Nuclide	$T_{1/2}$, year	Hold-up time after reactor shutdown, year				
		5	10	20	50	100
^{152}Eu	13.2	4.74E15	3.65E15	2.16E15	4.48E14	3.24E13
^{154}Eu	8.5	1.98E15	1.32E15	5.82E14	5.04E13	8.48E11
Sum, Bq		6.72E15	4.97E15	2.74E15	4.98E14	3.32E13

TABLE 13. Radioactivity of tritium, Bq

Components	Hold-up time after reactor shutdown, year				
	5	10	20	50	100
Core	1.85E15	1.40E15	8.00E14	1.45E14	8.50E12
Side reflector	2.80E15	2.10E15	1.20E15	2.20E14	1.30E13
CPS rods	5.50E13	4.10E13	2.40E13	4.40E12	2.60E11
Sum, Bq	4.70E15	3.54E15	2.02E15	3.70E14	2.18E13

TABLE 14. Radioactivity of long-lived fission products, Bq

Nuclide	$T_{1/2}$	Hold-up time after reactor shutdown, year				
		5	10	20	50	100
^{79}Se	6.5E04 y	1.47E10	1.47E10	1.47E10	1.47E10	1.47E10
^{85}Kr	10.74 y	2.62E14	1.90E14	9.95E13	1.43E13	5.68E11
^{87}Rb	4.88E10 y	9.13E05	9.13E05	9.13E05	9.13E05	9.13E05
^{90}Sr	28.5 y	2.53E15	2.24E15	1.76E15	8.46E14	2.51E14
^{90}Y	61.4 h	2.53E15	2.24E15	1.76E15	8.46E14	2.51E14
^{93}Zr	1.53E06 y	7.11E10	7.11E10	7.11E10	7.11E10	7.11E10
^{93m}Nb	13.6 y	3.20E10	3.70E10	4.50E10	6.00E10	6.60E10
^{94}Nb	2.03E04 y	8.52E05	8.52E05	8.52E05	8.52E05	8.52E05
^{99}Tc	2.15E05 y	4.64E11	4.64E11	4.64E11	4.64E11	4.64E11
^{106}Ru	368.2 d	1.02E13	3.29E11	3.43E08	3.9E-01	-
^{106}Rh	29.9 s	1.02E13	3.29E11	3.43E08	3.9E-01	-
^{107}Pd	6.5E04 y	5.39E08	5.39E08	5.39E08	5.39E08	5.39E08
^{108m}Ag	127 y	2.58E05	2.51E05	2.38E05	2.02E05	1.54E05
^{108}Ag	2.41 min	2.19E04	2.13E04	2.02E04	1.71E04	1.30E04
^{113m}Cd	13.6 y	1.52E11	1.17E11	7.05E10	1.52E10	1.18E09
^{121m}Sn	55 y	3.29E09	3.09E09	2.72E09	1.87E09	9.94E08
^{126}Sn	1.0E05 y	1.28E10	1.28E10	1.28E10	1.28E10	1.28E10
^{126m1}Sb	19 min	1.28E10	1.28E10	1.28E10	1.28E10	1.28E10
^{126}Sb	12.4 d	1.79E09	1.79E09	1.79E09	1.79E09	1.79E09
^{129}I	1.57E07 y	8.67E08	8.67E08	8.67E08	8.67E08	8.67E08
^{134}Cs	2.062 y	1.51E14	2.80E13	9.63E11	3.91E07	1.84E00
^{135}Cs	2.3E06 y	4.62E10	4.62E10	4.62E10	4.62E10	4.62E10
^{137}Cs	30.174 y	2.58E15	2.30E15	1.83E15	9.18E14	2.91E14
^{137m}Ba	153.5 s	2.43E15	2.17E15	1.73E15	8.66E14	2.74E14
^{144}Ce	284.3 d	3.30E13	3.87E11	5.37E07	1.4E-04	-
^{144m}Pr	7.2 min	4.30E11	5.04E09	6.99E05	1.8E-06	-
^{144}Pr	17.28 min	3.30E13	3.87E11	5.37E07	1.4E-04	-

^{147}Pm	2.623 y	5.93E14	1.58E14	1.12E13	4.05E09	7.27E03
^{151}Sm	87 y	4.39E13	4.22E13	3.90E13	3.07E13	2.06E13
^{152}Eu	12.4 y	2.68E12	2.03E12	1.16E12	2.18E11	1.34E10
^{154}Eu	8.5 y	4.29E13	2.85E13	1.26E13	1.09E12	1.84E10
^{155}Eu	4 96 y	2.75E13	1.37E13	3.38E12	5.09E10	4.63E07
Sum, Bq		1.13E16	9.41E15	7.25E15	3.52E15	1.09E15

TABLE 15. Radioactivity of steel of fuel element claddings, lattices and barrel of SRU, Bq

Nuclide	$T_{1/2}$, year	Hold-up time after reactor shutdown, year				
		5	10	20	50	100
^{59}Ni	7.5E04	3.82E11	3.82E11	3.82E11	3.82E11	3.82E11
^{63}Ni	100.1	3.40E13	3.30E13	3.05E13	2.50E13	1.80E13
^{60}Co	5.27	5.50E14	2.90E14	8.50E13	2.00E12	2.70E09
^{55}Fe	2.72	1.10E15	3.00E14	2.30E13	1.05E10	3.00E04
Sum, Bq		1.68E15	6.23E14	1.39E14	2.74E13	1.84E13

Analysis of the data of Tables 9÷15 shows that the integral activity of long-lived radionuclides in nuclear fuel, constructions and coolant during the hold-up time of 5÷50 years after reactor shutdown can reach $2.2\cdot10^{16}\div3.7\cdot10^{15}$ Bq ($6\cdot10^{5}\div10^{5}$ Ci), respectively. Note that among actinoids accumulated in the core over one fuel lifetime the amount of ^{238}Pu ($T_{1/2}$=87.75 years) makes up about 80 g. This means the presence in the shutdown reactor, in compliance with ^{238}Pu half-life, of a permanent neutron source of ~ $4\cdot10^{9}$ neutron/s intensity due to (α,n) reaction on beryllium nuclei of fuel composition. With due regard for multiplication in subcritical reactor, e.g., at 10-% subcriticality, the integral intensity of neutron radiation in the reactor over a rather protracted period following its shutdown would make up ~ $4\cdot10^{10}$ neutron/s. Actual storage conditions of both unloaded and non-unloaded SNF are characterized in Table 16.

TABLE 16. Actual storage conditions of unloaded and non-unloaded SNF

Land-based storage of 6 SRU, including:	2 SRU, design #645 and 1 SRU, designs ##106, 107, #905 and 915
Waterborne storage within RCs (2):	#105 (SPI #120) and #900
Waterborne storage in NS (2):	#105 (SPI #125) and #910
Total:	10 SRUs

8. SNF unloading problems

Putting into operation of transport reactor installations with LMC required the development and making of appropriate fuel-reloading facilities differing from those used in case of other-type reactors. Because, in addition to reloading equipment itself, such fuel-reloading facilities must have also included special hoisting machines and a dry dock for NS installation on a solid basement, a specialized LMC-reactor reloading CMB was established. The equipment to unload SRU of LMC-reactors of NSs designs #705 and #705K was developed by “Hydropress” Experimental-Design Office [1]. SRU unloading is preceded by a variety of preparatory activities including but not limiting to: -installation of base devices to control reactor subcriticality and -dismantlement of CPS actuation mechanisms with subsequent effective fixing of every rod at the lowermost position. Principal preparatory operations ensuring SRU taking off comprise: the alloy heating, dismantlement of standard equipment of SRU seal, sequential installation of the adapter box and the unloading cask on the reactor vessel flange, and next step-by-step lifting of the SRU into the cask with permanent physical measurements and radiation dose control. Then the unloading cask with SRU is moved to the preliminary cooling storage facility and is installed on its flange; after that the SRU is immersed stepwise into a storage container filed in advance with the necessary amount of alloy heated up to 150-170°C by the stationary heating system of the storage facility. Stepwise immersion of the SRU into the container is accompanied by physical measurements. Finally the SRU is sealed in the storing container, and compulsory air-cooling system is switched on. After decay heat level decrease in the core ≤ 3 kW the SRU-storing container is removed from the storage facility and placed into the long-term cooldown box equipped with natural-circulation air system.

9. Problems of storage of unloaded SNF

LMC reactors of Russian nuclear submarines have the following peculiarity: using a special transport-process equipment their cores are loaded into/unloaded from reactors in the form of a single SRU comprising the core itself with submerged CPS rods, side reflector (designs #645 and #705K) and the upper plug of biological shielding structure [2, 4]. In NS, design #705, the side reflector is not extracted when nuclear fuel unloading.

At present nuclear safety of spent nuclear fuel during storage is ensured owing to “deep” subcritical condition ($K_{eff} < 0.95$) of reactor cores stored inside NSs as well as at land-based storage facilities due to full submerging therein of CPS absorber rods. CPS drives of the cores stored at land-based facilities are dismantled, and sealed steel caps are installed and welded to shrouds of absorber rods cut at a level slightly above the reactor upper head. Both unloaded and non-unloaded SNF is stored within “frozen” lead-bismuth alloy. Unauthorized input of positive reactivity during SNF storage is impossible because lead-bismuth alloy within CPS shrouds is also “frozen”, and absorber rods are immobilized. CPS drives at non-defueled NSs are dead, the alloy in CPS shrouds being also “frozen”. A low-probable unauthorized hand extraction of four scram rods stored within “dry” channels would not take the reactor out of its subcritical state, the integral efficiency of scram rods being < 1.5 %. There is no way either for open pores to be generated within solidified eutectic of NS reactor thanks to the use of the “coolant-freezing” technology in the primary circuit. Thus ingress of condensation-origin (or of other-genesis) moisture into the NS reactor core in a hypothetical case of the primary circuit depressurization is impossible, the latter being pressurized using standard devices; in that way input of additional positive reactivity into such reactor is impossible. In land-based conditions lids of steel containers housing SNF are sealed; in addition, in the area of SRU-container contact a bitumen layer is laid to prevent ingress of condensation moisture, atmospheric precipitations and ground waters therein. Radioecological safety of SNF storage is based on the defense-in-depth principles against potential radionuclide release to the environment comprising a sequence of physical safety barriers. For fission products such barriers include: fuel element matrix, fuel element claddings, “frozen” coolant, pressurized primary circuit/steel container in the land-based facility and strong hull of NS. According to the radiation control data, all the cores of LMC reactors stored at land-based storage facilities and within NSs have virtually no damages of fuel element claddings. In case of radioactive products contained in coolant and in-vessel structures such barriers comprise: coolant matrix, reactor vessel (container) walls and NS strong hull. If SRU “freezing” is performed in the proper way excluding the generation of pores and cavities in eutectic (as is done, e.g., in NS reactors), the concept of long-term storage of SNF of LMC reactors within “frozen” lead-bismuth alloy is the most radical instrument of ensuring nuclear and radiation safety during storage. “Frozen” SRU represents a strongly dense assembly giving no way of penetrating appreciable water amounts into the core under any potential scenario. The matrix of “frozen” coolant has appropriate immobilization properties and extremely low radionuclide-leaching rate in case of contact with marine water or distilled water. Moreover, in such a way the problem of nonproliferation of highly enriched uranium is resolved quite successfully [5]. In case of strongly dense package of fuel assemblies their unauthorized extraction out of SRU is virtually impossible.

Thus, when storing SNF in land-based facilities, the following factors may be considered as favorable ones:

– Absorber control rods are at the lowermost position and immobilized;
– Coolant in tank-container of the land-based SRU facility is “frozen”;
– Subcriticality margin makes up ~10÷20 β_{eff};
– Tanks-containers housing SRUs are installed within a concrete cavity;
– Upper lids are mounted and sealed; and
– Radiation monitoring is performed.

However, when developing PRI with LMC, long-term SNF storage inside NSs and at land-based facilities was not kept in mind. According to initial plans, SRUs were to be transferred to the SNF-processing plant after some years of hold-up at the storage facilities. Under the used technology of SRU loading and storage in tanks-containers the risk of generating pores and cavities within “frozen” eutectics must not be ruled out. Consequently, the probability of penetration of condensed-origin or other-origin moisture into the core under actual SRU storage conditions still exists. According to preliminary calculations, ingress of 1 kg of cold water distributed uniformly over pore volume in the core would cause a positive effect to reactivity (about 0.5 β_{eff}). Thus in case of subcriticality of 10 β_{eff}, ingress of 20 kg of water (~10 % of coolant volume in the core) could result in attaining by SRU of critical state (K_{eff}=1). The main unfavorable factors during storage of unloaded SNF are summarized in Table 17.

TABLE 17. Unfavorable factors during SNF storage

RW type	RW of high specific activity
Sensitivity to moisture	~1 kg of water could increase the system reactivity up to ~0.5 β_{eff}. ~20 kg could lead to accidents with initiation of spontaneous chain reaction ($K_{eff} \geq 1$)
SRU subcriticality margin	~10 β_{eff}
Other unfavorable factors:	~80 g of Pu^{238} determines the presence of a permanent neutron source of ~$4 \cdot 10^9$ n/s
	galvanic and chemical corrosion; phase transformations at SRU storage facility
	sensitivity to external unfavorable factors

10. Problems of subsequent SNF management

- Long-term storage of SNF has not been provided for;
- The risk of water ingress into eventual pores and cavities in “frozen” coolant inside SRUs and thus an increase in reactivity up to initiation of spontaneous chain reaction must not be ruled out;
- One faces the problems of low-temperature embrittlement of steel claddings of SRU fuel elements, CPS absorbers as well as SRU steel structures potentially complicating the process of SNF unloading from land-based containers and subsequent shipment to the reprocessing plant;
- The control over reactivity during SRU storage is indispensable [6];
- Refined calculated-experimental investigations of the effects of potential water ingress inside the SRU-storing volume are necessary;
- Analysis of radiological consequences of hypothetical accidents due to natural and mane-induced impacts on SNF storage facilities is also needed.

Conclusions

From the above analysis the following conclusions may be yielded:

- Stored SNF of naval LMC reactors has accumulated high radioactivity;
- Stored SRUs have relatively low subcriticality and are sensitive to the impacts of unfavorable external conditions;
- Under the used technology of SRU temporary storage the probability of water ingress into the container storing SRU must not be ruled out: the risk of initiating spontaneous chain reaction will exist until every SRU is dismantled;
- Physical and chemical processes inside SRU (galvanic and chemical corrosion etc.) could lead to depressurization of SRU-storing container and the resulting contamination of the nearby territories;
- There is need to examine hypothetical accidents involving SRU-storing containers due to different-type external impacts (fire, aircraft fall, flood, and so forth) and analyze their potential implications;
- Development of international cooperation to deal with the problems of SNF

unloading and long-term storage, including assessments of the risks related to potential release of accumulated activity to the environment, represents a topical world-wide international challenge since stored SRUs are potential sources of radiation hazard for population of the surrounding territories.

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TRANSPORTATION OF SPENT NUCLEAR FUEL (SNF) AND RADIOACTIVE WASTE (RW)

MANAGEMENT OF NON-STANDARD SNF

RW PROCESSING

ANALYSIS OF POTENTIAL EMERGENCIES DURING SNF AND RW MANAGEMENT

ENHANCING THE EFFICIENCY OF RADIOLOGICAL MONITORING AND ON-LINE EMERGENCY RESPONSE IN CASE OF SINKING OF TAKEN-OUT-OF-SERVICE RADIATION-HAZARDOUS OBJECTS

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Retrospective analysis of occurred radiation accidents shows that the most of their-related damages was due to impertinence, incompleteness and biased character of issued information. If the concerned specialists had responded more efficiently on the emergency events, their implications could have been minimized (or prevented).

Adequate decision-making and efficient response actions are impossible without prompt and reliable information and efficient information-analytical expert-support systems aimed at elaborating optimal measures for protection of personnel, population and environment.

To date development of radiological monitoring and on-line-response systems capable of supporting safe decommissioning of nuclear- and radiation-hazardous facilities of the Russian Navy is in the early stage; moreover, they still do not cover one of the most complicated aspects of the problem related to potential accidents at sea.

When preparing taken-out-of-service Nuclear Submarines (NSs) to complex decommissioning, their keeping afloat and towing to shipyards represent the most important phases. Analysis of technical condition shows that to date 45-70% of such NSs have unsealed Driving Ballast Tanks (DBTs). Some DBTs are filled with polystyrene that slightly increases NS buoyancy; the remaining NSs are kept afloat thanks to still intact DBTs. Extension of waterborne storage time for such NSs by another 5-10 years would inevitably result in worsening of technical condition of their DBTs (Table 1).

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 147–163.

TABLE 1. Generalized data on technical condition of nuclear submarines

NS characteristic	NS number (%)	
	Northwest Russia	Far East Russia
NSs after 25-45 years of service life	24 (60%)	23 (72%)
NSs stored afloat over 10 years after withdrawal from service	16 (40%)	21 (65%)
NSs stored afloat over 10 years without in-dock repairs	12 (30%)	20 (63%)
NSs with unsealed DBTs	4 (10%)	12 (38%)
NSs with unsealed DBTs filled with foamed polystyrene	14 (35%)	11 (34%)

In case of partial DBT depressurization, NS's crew is forced to ground such submarine by the nose/stern parts close to coastal line (case of shallow waters) within waterborne storage center – this is one of possible methods of preventing complete NS sinking. The probability of such an event is estimated today at about 10^{-2} per year. However, if a taken-out-of-service NS is stored on a deep-water berth, the above procedure is inapplicable, and special buoyancy-reinforcing measures should be taken allowing decreasing the sinking risk down to ~ 10^{-3} - 10^{-4} events per year.

Let us recall that 20 years ago the risk of initiating Self-sustained Chain Reaction (SChR) during NS defueling operations was estimated at a level of 10^{-5} - 10^{-6} (the present-day estimate is 10^{-8}). Nevertheless, 2 SChR emergencies really occurred in the past (in 1965 at "Zvezdochka" shipyard, Severodvinsk, and in 1985 at Navy Shipyard #30, Primorskiy kray). Other-type emergency occurred in 2003 when a NS (*K-159)* sank in the Barents Sea during haulage to shipyard for complex decommissioning.

The risk of accidents during Spent Nuclear Fuel (SNF) shipment using Floating Service Vessels (FSVs) is yet less, but – as shown above – the very probability of such an event must not be ruled out.

Under accident-free haulage of NSs, Nuclear-Powered Surface Ships (NPSSs), Reactor Compartment (RC) units, nuclear Maintenance Vessels (MVs) and Radioactive Waste (RW) as well as during defueling operations the radiation impact on population is virtually lacking, and the radiation risk does not exceed an unconditionally acceptable level of $1 \cdot 10^{-6}$ [1]. However in a case of emergency the radiation risk would increase potentially reaching $60 \cdot 10^{-6}$ - $7000 \cdot 10^{-6}$ that would be unacceptable for population (acceptable risk: < $50 \cdot 10^{-6}$ [1]) (Table. 2).

TABLE 2. Individual dose commitment, radiation risks and environmental implications under hypothetical emergencies during NS complex decommissioning

Accident scenario	Event probability*, year^{-1}	Dose, mSv/year		Risk, 10^{-6} ***** in case of accident/potential risk		Environmental implications
		Personnel	Population	Personnel	Population	Contamination scale
Sinking of non-defueled NS on berth	~ 10^{-3} - 10^{-4}	0.1–1**	0.01-0.001	56 / < 1	< 1 / < 1	Local contamination
Sinking of non-defueled NS when towing to defueling center	~·10^{-3}	0,1–1**	0.01-0.001	56 / < 1	< 1 / < 1	Local contamination
Fire in NS reactor compartment	~ 10^{-2}	0.1-1	≤ 0.001	56 / < 1	< 1 / < 1	Local contamination
FSV collision with another vessel, fire, sinking	~ 10^{-8}	1-100	1-10***	5600 / < 1	730 / < 1	Large-scale contaminaiton
Aircraft fall onto FSV, destruction of SNF storage facility at FSV, fire, sinking	~ 10^{-9}	10-100	1-100****	10^{5} / < 1	7300 / < 1	Large-scale contamination, transboundary transfer is possible

Comments:

* - estimated on basis of complex decommissioning of 10 NSs per year;

** - case of unauthorized fishing in the contaminated bay and consumption of sea products;

*** - case of accident nearby a settlement 2-3 km from the coast;

**** - case of radioactive cloud passage via settlement.

***** - according to [1], the "risk" is determined as a product of the probability of events contributing to additional dose commitment by a risk coefficient and a value of individual (collective) dose. The life-long risk coefficient characterizes reduction of the duration of full-value life by 15 years (on average) per one stochastic effect (due to fatal cancers, serious hereditary effects and non-fatal cancers with similar-to-fatal-cancer consequences).

Though in a case of emergency environmental implications would be mainly of local scale, the probability of large-scale contamination - up to transboundary transfer of radioactive substances – must not be ruled out. Seawater and air would become the mostly affected environmental components.

Radiation monitoring of marine environment at the waterborne-storage phase of taken-out-of-service naval nuclear vessels is placed on the Russian Naval Service for Radiation, Chemical and Biological Protection (RChBPS). During dismantling operations the responsibility for radiation monitoring rests with Radiation Safety Services (RSS) of the relevant shipyards, whereas during RC unit storage with RSSs of Federal State Unitary Enterprises (FSUEs) “SevRAO” and “DalRAO”.

However in a case of emergency similar to either *K-159* NS sinking or SChR initiation while defueling (similar to the Chazhma Bay accident, 1985), the resources of RChBPS and RSSs of shipyards would be insufficient to organize and conduct full-scale and long-duration radiation monitoring. So far positive results have been yielded only through joining of efforts of different entities: establishing and conducting of radiation monitoring of sunken *K-159* NS and the surrounding area in September-November 2003 demonstrates a spectacular example of such activities.

A large NS *K-159*, design 627, withdrawn from military service in 1989, was stored afloat for a protracted time in Iokan’ga Basing Center (Gremikha, Murmansk region) together with other similar-type NSs. In 2003, in compliance with the approved schedules, transfer of several NSs to “Nerpa” shipyard (Kut Bay, Snezhnogorsk-town) for subsequent defueling and dismantlement began. By August 2003 several NSs were successfully conveyed to the shipyard by rescue ships and maintenance vessels.

Unfortunately, on August 30, 2003 *K-159* NS being hauled to shipyard rubbed into heavy weather nearby Kildin Island. Before putting to sea, necessary technical measures were taken to improve the submarine floodability including DBT filling with foamed polystyrene and NS supporting by two pairs of ship-raising pontoons, which, however, broke during storm. Despite undertaken measures, the NS sank 3.7 miles from Kildin Island (the Barents Sea) at 248 m depth. Nine seamen of the convoy team perished [2] (Fig. 1).

Figure 1. Haulage of *K-159* NS, design 627, to shipyard for complex decommissioning

The Northern Fleet's RChBPS was entrusted with the task of organizing and conducting radiation monitoring in *K-159* sinking area. Taking into account the specificity of occurred event, the monitoring procedure was discussed and agreed with leading experts of Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAS) and Russian Research Center "Kurchatov Institute" (RRC KI). It was on August 31, 2003 at 07:00 a.m. (i.e. only one day after the accident) that a team of specialists began radiation survey of the NS-sinking area.

Radiation monitoring and hydro-meteorological surveys were carried out from the board of the Northern Fleet's "*Horizont*" hydrographic vessel equipped (like other vessels of design 862/II) with special instrumentation and tools allowing taking samples of sea water and bottom sediments and installing various submersible, towing, sea-bottom, meteo- and hydrological gauges for radiation monitoring purposes. In addition to standard crew, the hydrographic vessel was capable of taking up to 20 members of the expedition on board and was fit out with additional radiation monitoring sets (Fig.2).

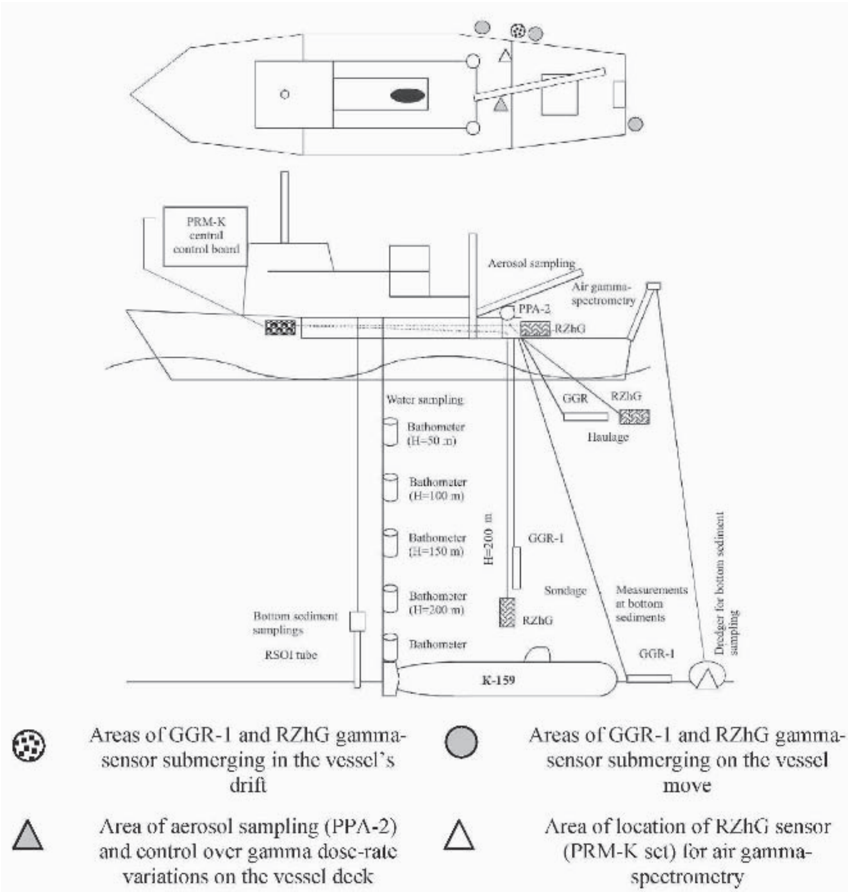

Figure 2. Layout of main radiation monitoring systems on board of "*Horizont*" hydrographical vessel while performing monitoring of *K-159* sinking area (Barents Sea, 31.08.2003 – 09.09.2003)

Environmental survey - being of interest for forecast of radioactive admixture spreading in seawater and interpretation of the results of radiometric and gamma-spectrometric measurements - included identification of the following hydrophysical parameters: temperature, salinity, direction and velocity of sea currents at 50, 100, 150 and 200-m depths and also at 5, 10 and 15 m from sea bottom.

A satellite navigation system was used to identify the vessel location, the root-mean-square error being ± 10 m. To take seawater samples, the bathymetry method was mostly applied. Sea-bottom samples were taken using a dredger and a special sea-ground tube; fish was caught into special fishing tackles. Temperatures, salinity, sea current direction and velocity were determined in a standard way. Specialists of the Northern Fleet's HydroMeteorological Center (HMC) processed the results of hydrometeorological measurements on board using the methods of Reference [3]. Experts of IBRAE RAS and the Northern Fleet's RChBPS estimated the radiation situation according to the procedures of References [4 and 5].

Thus the tasks of real-time radiation monitoring consisted in:

- estimating the radiation situation within NS-sinking area using the results of measuring concentrations of radioactive substances in atmosphere, seawater, bottom sediments and biota and elaborating adequate forecasts;
- estimating the condition of shielding barriers of the sunken submarine's nuclear power installation on basis of the radiation monitoring results;
- -estimating the effects of man-caused radionuclides on population and environment in case of intense release of radioactive substances in air and water;
- regular informing the Northern Fleet's commanders and executive authorities on changes in the radiation situation, and
- elaborating appropriate recommendations in case of unfavorable scenario development.

Due to specific objectives placed on specialists of the radioecological team, the initial phase of monitoring consisted in rapid (*tens of minutes*) acquisition of information necessary to estimate the radiation situation at levels corresponding to/exceeding admissible limits. In case of no intense radioactive contamination high-sensitivity methods of analysis were provided with data acquisition at a level of expected/predicted radioactive contamination (*hours*) – that was the second monitoring phase. If no data were obtained at the second phase, the third monitoring phase came into play aimed at revealing both initial indices of issue of man-caused radionuclide to seawater and identifying radionuclide genesis, i.e. distinguishing "man-caused" radionuclides from "natural" ones (*tens of hours to one day*).

Simultaneously, on the hydrographic vessel board specialists generated forecasts of unfavorable event progression and elaborated necessary recommendations using "*Nostradamu*s" and "*Kassandra*" simulation routines developed at IBRAE RAS for decision-making in case of emergency release at radiation-hazardous facilities [6, 7]. Similar-type work - but with emphasis on broader effects admitting transboundary transfer of radioactive substances - was conducted at crisis centers of IBRAE RAS, RRC KI and Ministry for Atomic Energy of the Russian Federation (Minatom, presently Rosatom). Regular information contact was established between those crisis centers and the on-board team of specialists.

In compliance with the scheduled monitoring program, in the concerned area volumetric air-aerosol activity, gamma Exposure Dose Rate (EDR) and density of surface alpha-beta contamination on the vessel deck were measured, samples taken and next transferred to a coastal center for the situation control and estimate.

One hour after the measurements had confirmed the lack of intense contamination of the near-surface air, sea surface and the vessel itself, specialists began sounding of sea water column (from surface to bottom) using deep-water gamma-roentgenometer GGR-1 and PRM-K complex (RZhG sensor on base of NaJ (Tl) low-background monocrystal, 63·250 mm in size).

Several more hours were necessary to establish the lack of significant man-caused radioactive contamination of seawater and bottom sediments; accordingly, the monitoring program was re-oriented toward discovering initial indices of marine environment contamination. Underwater radiation monitoring complex PRM-K with 1-10-hour exposure time allowed identifying ^{137}Cs in seawater within the range of 30-100 Bq/m^3 that was by far below its Maximum Admissible Concentration (MAC) in seawater (according to the Russian Navy's standards [5], MAC for ^{137}Cs equals 2220 Bq/m^3, for ^{90}Sr - 740 Bq/m^3).

Simultaneously, the use of low-background laboratorial gamma-ray scintillation spectrometer in vessel conditions allowed determining ^{137}Cs in bottom sediment samples at a level of 3 Bq/kg (according to the Russian Navy's standards [5], MAC for ^{137}Cs in bottom sediments of the tidal zone makes up 2590 Bq/kg) and in fish - 2 Bq/kg raw weight (in keeping with the Russian Radiation Safety Standards (NRB-99) [1], MAC for ^{137}Cs equals 11 Bq/kg, for ^{90}Sr – 5 Bq/kg).

In addition, the investigations were organized in such a way as to be capable of comparing in real conditions the results of radiation monitoring close to the sunken NS with the radiation background parameters typical for the studied water area. Systematic observations covered an area of $2.5 \cdot 2.5$ km around the NS sinking point; radiation background was measured and samples were taken at 6-8 km distance from the studied zone.

According to the results of investigations, volumetric activity of radioactive aerosols did not vary throughout the monitoring period and did not exceed MAC and the background levels. Gamma EDR remained within the radiation background variations; surface alpha-beta contamination on the vessel deck was not discovered at all; no man-induced radionuclides (firstly ^{137}Cs) in concentrations exceeding MAC and radiation background were revealed in seawater, bottom sediments and fish species (Figs. 3-5, Table. 3).

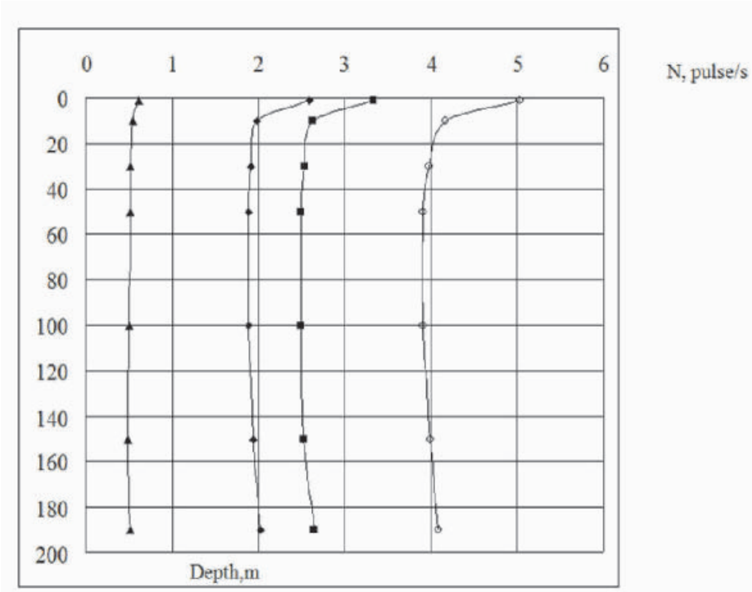

Figure 3. Depth-depending variations in the counting rate of PRM-K scintillation sensor within different energy ranges above the sunken *K-159*

(04.09.2003 ◆ - 0.2–0.55 MeV, ■ - 0.2–0.8 MeV,

▲ - 0.55–0.75 MeV, ○ - 0.2–2.5 MeV)

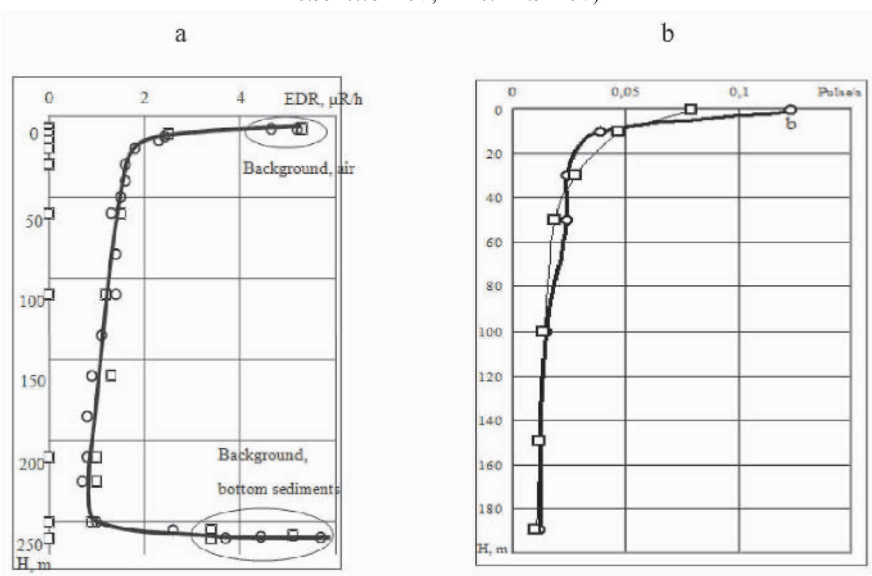

Figure 4. Variations in gamma EDR in air, seawater and bottom sediments obtained using GGR-1 (a) and in the counting rate of PRM-K scintillation sensor within 1.8–2.5 MeV energy range (b) depending on depth in the NS *K-159* sinking area ((○) – August-September 2003; (□) – October-November 2003)

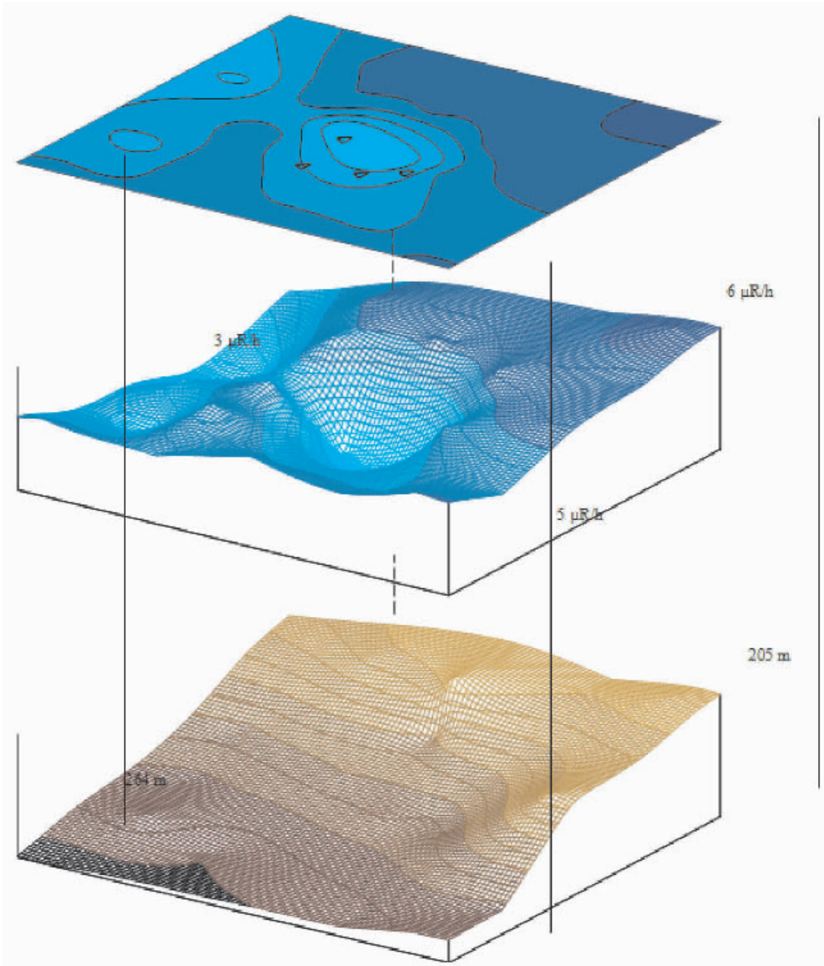

Figure 5. 3D model of gamma-fields of bottom sediments and bottom relief in NS "*K-159*" sinking area (area size: 2.5·2.5 km, the submarine is located between the points indicated at the upper projection (square center))

TABLE 3. Concentrations of man-caused and natural radionuclides in bottom sediments, seawater and fish

Sample type Station # Sedim., Bq/kg	Date	Bottom sediment layer, radionulclide			
		0 – 3 cm		3 – 10 cm	
		^{137}Cs	^{40}K	^{137}Cs	^{40}K
1*	3-8.09.03	< 3	362±70	<3	396±75
2		< 3	387±70	< 3	418±80
3		< 3	423±80	< 3	456±80
4		< 3	323±65	3±1	436±80
5		< 3	375±70	< 3	489±85
1**	8-9.09.03	< 3	346±70	< 3	417±75
2		5±1	384±70	< 3	483±85
3		< 3	469±85	< 3	535±90
4		< 3	469±85	< 3	618±110
5		< 3	592±95	3±1	603±110
6		< 3	373±70	3±1	462±90
7		3±1	400±75	3±1	420±80
8		< 3	368±70	< 3	424±80
9		< 3	359±70	< 3	403±75
10		< 3	403±75	< 3	495±90
11		3±1	372±70	< 3	452±85
Backgr.-1	3.09.03	3±1	498±75	< 3	531±100
Backgr.-2	9.09.03	5±1	359±70	3±1	465±95
Water, Bq/m^3					
- sample***	3.09.03	< 200	-	-	-
-gamma-spectrometry	3-7.09.03	< 30	-	-	-
Fish: Bq/kg *	07.09.03				
- internals;		< 2	144±30	-	-
- head;		< 2	149±30	-	-
- tissues.		< 2	337±60	-	-

Comments: * the hydrographic vessel was placed onto a special floating “tank” 100-300 m from the sunken NS. Were taken samples of: bottom sediments (## 1-5) and fish (cod and Peter's fish) at 150-m depth.

** samples of bottom sediments (## 1-11) were taken within 2.5·2.5 km area around the sunken NS over a grid with pitch 500±100 m. Background samples were taken at opposite points 6-8 km from the emergency area.

*** two water samples were taken 1 m from sea bottom using a special tube of Russian State Oceanographic Institute (RSOI) 100-300 m from the sunken NS. Gamma-spectrometry at 50, 100, 50 and 190 m was performed using PRM-K complex with 10-h exposure time.

From Table 3 data it follows that in the course of the first 10 days after the submarine sinking radionuclide composition of bottom sediments was determined (99%) by ^{40}K – natural-origin radionuclide; the remaining ~1% was due to man-caused ^{137}Cs which concentrations (3-5 Bq/kg) were attributed to global fallouts. That conclusion was confirmed by the results of radiation background measurements and independents surveys (see Reference [8]).

Gamma EDR in air and water was entirely determined by cosmic rays, whereas in bottom sediments – by natural-origin radionuclides. Depth-depending increase in the counting rate of PRM-K scintillation sensor (from 50-m depth and below) within all energy ranges (Fig.3), save for 1.8–2.5 MeV (Fig. 4b), was due to natural factors - namely to increase in salinity, thus of ^{40}K concentration, with depth.

Hydrological observations revealed water temperature variations within 8-10^0 C in 0-50 m seawater layer. The sudden temperature-change layer was located between 50-150 m (temperature drop from 8-9 ^{0}C to 3–4 ^{0} C); from 150-m depth down to sea bottom water temperature remained virtually invariable (3.0-3.5^{0}C). Water salinity throughout water column varied within 34.1-34.5 $^0/_{00}$ increasing by 0.1-0.3 $^0/_{00}$ only after 50-m depth. The resultant currents were mostly of north and south directions; their velocities varied from 0.1 to 0.8 knots (Figs.6 and 7).

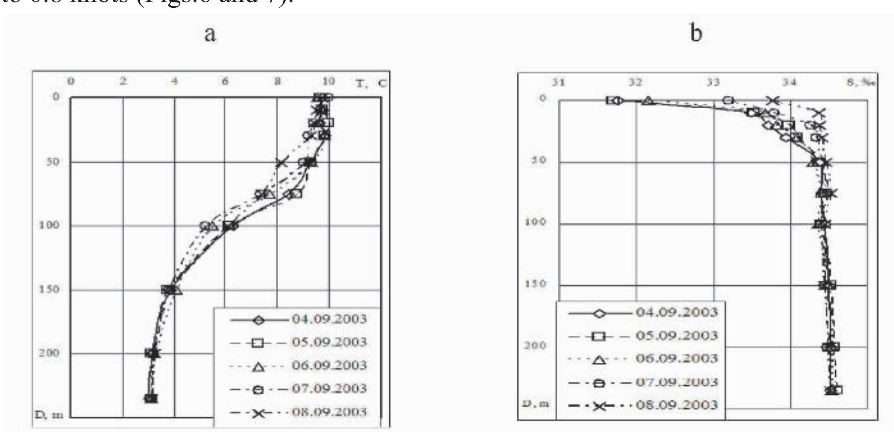

Figure 6. Depth-depending variations of (a) seawater temperature and (b) salinity in NS *K-159* sinking area

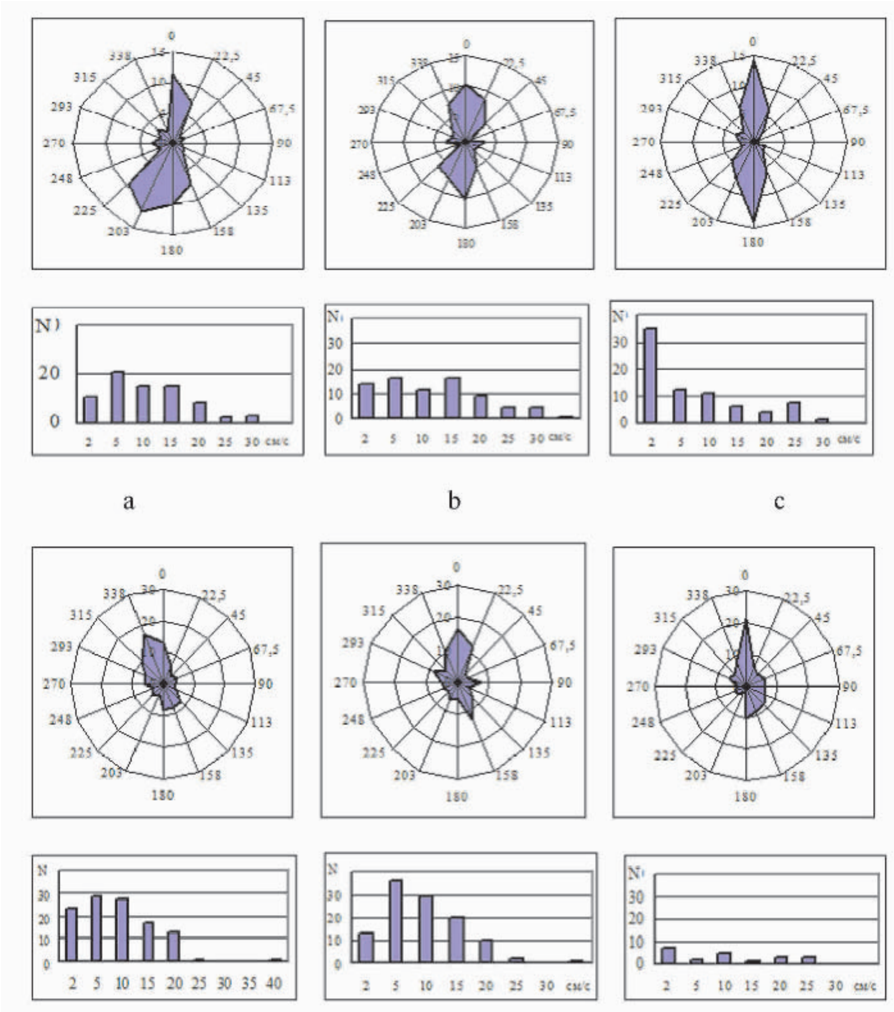

Figure 7. Resultant one-and-a-half-day sea currents in the near-bottom layers: a – 235 m; b – 230 m; в – 225 m (05-06.09.2003 and 07–08.09.2003, D = 242 m)

Analysis of the results of integrated investigations allowed the following conclusion: after the first 10 post-sinking days the radioecological situation in the NS sinking area did not change and remained indistinguishable from the natural radiation background. However that conclusion was considered as a preliminary one for no estimate of the radiation situation was performed on the submarine's surface.

Such investigations became possible only three months later thanks to joint efforts of specialists of RRC KI, IBRAE RAS, Research and Development Institute of Power Engineering (NIKIET) and Russian Navy [9]. A Navy's deep-diving manned vehicle managed to pass along *K-159* hull 1-2-m from the sunken submarine. Thanks to the use of a self-contained high-sensitive low-background gamma-spectrometer (RRC KI's development, monocrystal NaJ (Tl), size 200·200 mm) it was established that several months after *K-159* sinking ^{137}Cs concentrations in seawater just above the submarine did not exceed 50 Bq/m^3 (i.e. the method sensitivity threshold by the measurement time), whereas gamma-spectra of water and bottom sediments were identical to those of background (Figs. 8 and 9).

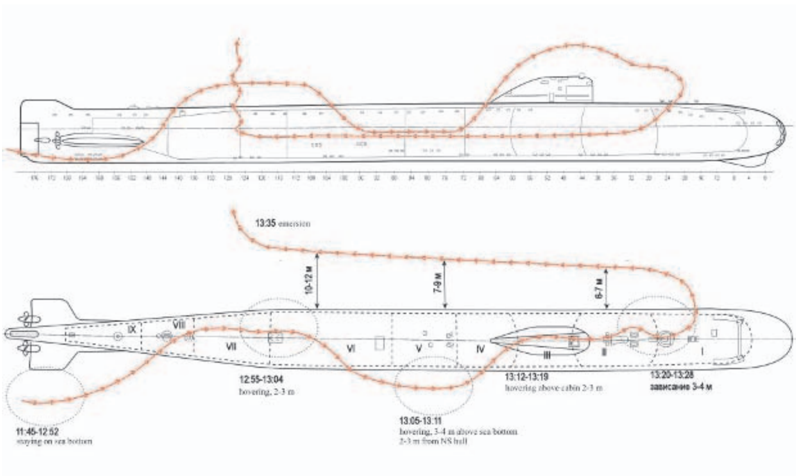

Figure 8. Route of the guided deep-diving vehicle above NS *K-159* during radiation monitoring

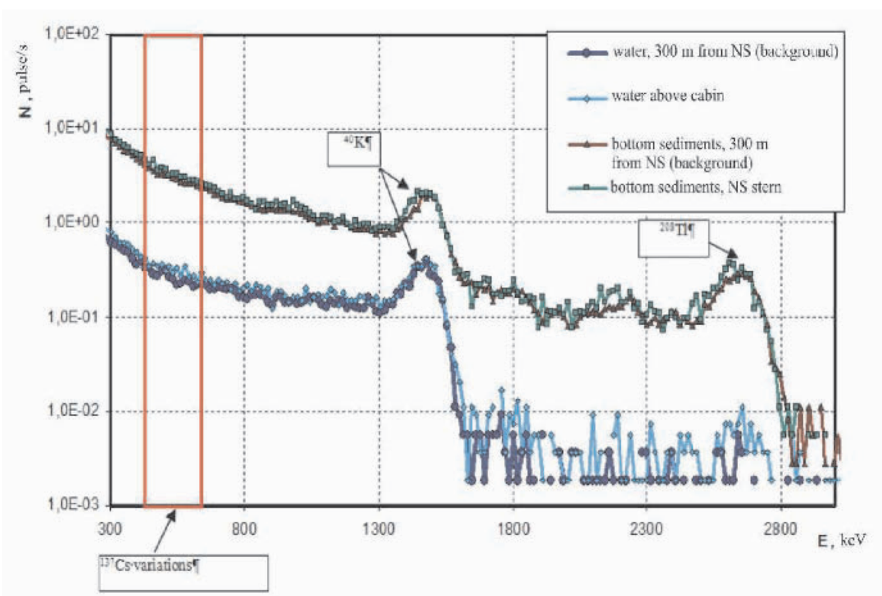

Figure 9. Gamma spectra recorded by deep-diving vehicle in November 2003 when surveying marine environment around the sunken submarine (*K-159*)

Radionuclide analysis of bottom sediment samples taken 10 m from the submarine along the prevailing bottom current confirmed the “background” concentration of ^{137}Cs (< 5 Bq/kg) therein.

Simultaneously performed repeated EDR measurements (using GGR-1) in bottom sediments 50-200 m around the submarine also obtained the results within 3-5 μR/h. At a later time (January-December 2004) the RChBPS specialists put to sea more than once. According to the results of their measurements on sea bottom in the submarine sinking area (using GGR-1), the radiation parameters still remained within the background levels.

Unfortunately, despite a large scope of performed investigations, no continuous radiation monitoring at the spots of most probable man-caused radionuclide release out of NS strong hull (on the submarine surface) has been organized so far due to some technical reasons (no gamma sensor installed). Thus there is presently no way of revealing eventual initial instant of radionuclide issue outboard and estimating the release rate. Information on hydrophysical condition of marine environment is occasional either; it does not cover the whole range of potential variations limiting the possibilities of adequate forecasting of the radiation situation.

The issue of radionuclide migrations from nuclear power installation to reactor compartment and adjacent compartments is still unclear that could result in seawater contamination when salvaging and towing the submarine to shipyard.

Modeling of NS shielding barrier destruction shows that the risk of man-caused radionuclide release to seawater must not be ruled out. Though such a release would be delayed and low active, it could be detectable over many decades. As the result, 20-30 years after the accident the following seawater area could be permanently contaminated above MAC: horizontal plane - 100-600 m, vertical plane - 10-30 m [10]).

Thus, despite many advantages, the monitoring conducted for purposes of identifying and estimating the radiation situation and ensuring safety of the nearby population had a number of important shortcomings, such as:

– - unreasonable delay (by 3 months) in obtaining some specific data necessary to take urgent decisions;
– - no way of acquiring necessary data without involvement of several additional entities –subordinates of different agencies – that required major coordinating efforts;
– - the monitoring program remained unfinished because of insufficient engineering preparedness;
– - both emergency forecasts and estimates of emergency situation in sea conditions were incomplete and rather limited for lack of mobile information-analytical expert-support system aimed at elaborating optimal solutions;
– - during the early post-sinking period no opportune and comprehensive information was issued for attention of general public in Russia and abroad.

When eliminating the consequences of nuclear and radiation accidents, successful implementation by radiation safety specialists of their functions requires prompt solution on-site of the following tasks: measurements of the radiation situation parameters in the emergency area; prompt generation of a forecast of the radiation situation changes; elaboration of appropriate recommendations on eliminating emergency implications and protecting personnel, population and environment. Adequate solution of these tasks is based on use of a variety of information–reference and simulation systems and appropriate databases.

As demonstrated the experience of radiation monitoring in *K-159* sinking area, when eliminating emergency consequences at sea - especially at the early and intermediate post-accident phases - the role of mobile groups and laboratories rapidly deployable at survey vessels is especially important. It should be also taken into account that in a case of potential transport accident or hypothetical terrorist attack no monitoring and data transfer tools could be available at the early instants. However both forecast of radiological consequences of an emergency and elaboration of adequate recommendation on protection of population and environment require non-stop refining of already available forecasts on basis of experimentally obtained data and corrections of previously made radiation situation estimates.

Development of a mobile hardware and software complex for monitoring and expert support during elimination of radiation emergencies aimed at resolving the above tasks at sea is the deed of the near future. As a matter of fact, such complex should represent a diminished modification of stationary information-analytical crisis center and fulfill its main functions being in the immediate vicinity to the accident area.

In addition to standard information-analytical support of stationary information-analytical crisis centers, such mobile complex should be fitted out with tools of on-line measurement of hydro-meteorological and radiation parameters of marine environment and radiation parameters of the acting radioactive contamination source. That would provide for not only acquiring current data on radiation and radioecological situation but also for verifying and correcting the radiation situation forecasts. To establish on-line data exchange with crisis centers involved into work on elimination of emergency consequences, the mobile complex should be equipped with satellite navigation systems and various communication facilities.

The described approach has been already applied when developing a coastal mobile hardware complex [11]. Software and hardware support of this complex comprises: a database on standard documents in the radiation protection area; reference databases on radiation-hazardous facilities, personnel and equipment of emergency-rescue teams of Rosatom, digital map bank, computer systems for on-line forecast and measurement of the radiation situation parameters, different data-exchange communication channels, etc.

Enlarging of the used approach toward establishment of marine mobile complexes will be able to increase safety at the very important phases of nuclear vessel complex decommissioning – long-term waterborne storage and haulage.

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SAFE SHIPMENT OF NAVAL SPENT NUCLEAR FUEL

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Removal of Spent Nuclear Fuel (SNF) of retired Nuclear Submarines (NSs) from the Russian Northwest and Pacific Regions to "Mayak" Production Association (PA "Mayak") began in the 1970s after commissioning of a buffer storage facility at PA "Mayak". Over more than 30-year period about 200 runs of special SNF-shipping train were performed, the length of route from the Northwest region's naval bases equaling 3000 km, from the Pacific's naval bases 7500 km.

At present, the fleet of transport facilities involved into shipment of naval SNF comprises:

1. 52 transportation casks 'TUK-18' certified for all types of SNF reprocessed at PA "Mayak";
2. Two special trains of 4 railcars of 'TK-VG-18' and 'TK-VG-18A' types. One more special train of 6 railcars 'TK-VG-18-2' is to be commissioned in the near future. However for lack of an extra escort-railcar, the new train can be either put into operation instead of one of the two actually-operating special trains or broken-up, its railcars being hooked to each of the two trains. It should be emphasized that 'TK-VG-18A' and 'TK-VG-18-2' railcars were made thanks to the assistance of Norway and the USA;
3. Metal-concrete casks 'TUK-108/1' certified for storage of SNF of NSs of the first and the second generations have been already tested and are to be put into operation in the immediate future. Because this-type containers have been mainly designed for long-term storage of naval SNF, their use under the "run-around-track" conditions necessitates some design adaptations.

Technical condition of 'TUK-18' after 10 years of running may be considered as quite satisfactory. By contrast, that of railcars 'TK-VG-18' and 'TK-VG-18A' calls for special attention. After virtually every run various railcar defects are revealed and eliminated (sometimes with the developer's participation). In most cases the defects are due to excess of limit load for such-type railcars. The new railcar 'TK-VG-18-2', designed with consideration for previous special-train-running experience, will be loaded with 2 'TUK-18'. It is deemed that the new-design railcar will not have so many defects as those of previous designs.

As known, SNF is the source of the following major types of hazard:

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 165–168.

- Nuclear hazard due to fissile materials contained therein;
- Radiation hazard resulting from the presence of a wide range of radionuclides; and
- Decay heat.

Safe shipment of SNF is reached thanks to the following technical and organizational measures:

- observance of the requirements of national standard and regulatory base which development complies (in most cases) with the world tendencies;
- licensing of activities of the involved entities;
- supervision of SNF-shipment-related works by the Department of State Supervision over Nuclear and Radiation Safety of Ministry of Defense of the Russian Federation (RF);
- certification of transportation casks and transport facilities for compliance with the requirements of the national and international standards;
- implementation of quality assurance system during transport facility preparative operations;
- functioning of Emergency-Technical Centers (ETCs) of the RF Agency for Atomic Energy (RF Rosatom) and of special Emergency-Rescue Unit (ERU) of PA “Mayak”.

PA “Mayak” has a license for activities related to the use of atomic energy for defense purposes. Such a license is granted by Rosatom only in a case that the relevant license applicant has certified transport facilities, developed infrastructure and trained personnel to perform servicing and repair of the transport facilities in question in compliance with the environment-protection requirements in force.

Obviously, it is the construction of transportation casks that ensures nuclear and radiation safety during SNF shipment. Since 1994 shipment of naval SNF has been performed in transportation casks ‘TUK-18’ complying with national and international safety requirements to B(U)-type packages under normal running conditions and in a case of emergency.

To ensure safe SNF rail shipment, the following technical and organizational measures are taken by PA “Mayak” and the Federal Agency for Rail Transport:

- only certified packages and special railcars are used;
- SNF transportation is performed in compliance with the requirements of the guiding documents in force on shipment of special cargos by special trains (category of shipment: “Special-Importance Train" - SIT);
- the cargo is escorted by a team of trained specialists of PA “Mayak”;
- special trains have adequate physical protection systems and are guarded;
- throughout the route monitoring of SNF shipment is performed;
- prior to SIT departure PA “Mayak” informs the relevant supervisory body on the planned shipment;

- “Quality Assurance System during Nuclear Material Transportation” is in force at PA “Mayak”; and
- a properly-equipped Emergency and Rescue Unit (ERU) has been established at PA “Mayak” prepared for emergency response.

A “Plan for Elimination of Potential Emergency Situations during Shipment of Nuclear Materials and Radioactive Substances” is in force at PA “Mayak”. Both regular and reserve ERU teams have been trained and certified in compliance with the "Program of Training Rescuers and Members of Special Emergency Groups".

A “Notification Procedure for Special Emergency Group in Response to “Lightning” Priority Signal” and “Duty Regulations of ERU Leaders and Functional Tasks of ERU Teams” have been developed and put into action at PA “Mayak”.

PA “Mayak” has concluded a contract with ETC of RF Rosatom in Saint Petersburg for works on preventing transport incidents and accidents during transportation of nuclear materials and radioactive substances. According to the contract, regular Rosatoms’s ERUs are to render assistance in a case of emergency during shipment of special cargos of PA “Mayak”.

In a case of emergency the escort personnel is to follow the “Emergency Card #914-B for shipment of SNF with ^{235}U concentration above 1% by rail or motor transport”.

In addition to ordinary safety measures provided by the railroad services, some special measures are also taken to ensure safe run of SITs. First and foremost, special timetables and itineraries are developed by the railroad services providing for SIT routes to bypass cities, major towns and passenger terminals via avoiding lines. SITs are moved under permanent on-line monitoring, and a variety of special measures are applied to guarantee safe SNF shipment.

The responsibility for nuclear and radiation safety, physical protection of nuclear materials and payment of damages caused by potential radiation impacts in a case of radiation accident during shipment of naval SNF rests on the following three parties: “The Shipper”, "The Cargo Carrier" and “The Consignee” in compliance with the relevant provisions of the contracts in force.

Special units of PA «Mayak» and special guards of the RF Ministry of Internal Affairs ensure physical protection of nuclear materials.

The actual physical protection system of PA “Mayak” related to transportation of nuclear materials provides for:

- nuclear material escorting by an armed guard throughout shipment;
- performing preliminary trustworthiness tests of personnel having access to nuclear materials and installations;
- minimizing the number of involved persons; and
- prohibition for the persons do not concerned with nuclear material shipment to accompany SITs.

In addition, prior to SIT departure a special committee performs complex check-up of SIT for purposes of physical protection of nuclear materials (control over technical condition, equipment and fire safety; observance of security rules and anti-terrorist preparedness).

Nevertheless, due to increase of threats from different terrorist groups, works must be continued on further upgrading of the actual physical protection level at PA "Mayak"'s transport facilities. First and foremost, this implies prompt equipment of the special train of 'TK-VG-18' railcars and the 'TK-VS' escort-car with up-to-date systems of physical protection and communications.

The 30-year experience of accident-free rail shipment of SNF has brought out clearly that the used so far multi-purpose approach to solution of the challenges of safe shipment of nuclear materials is efficient and is capable of ensuring reliable and safe transportation of SNF under normal and emergency conditions.

MAIN PECULIARITIES OF MANAGING SPENT NUCLEAR FUEL AND RADIOACTIVE WASTE DURING COMPLEX DECOMMISSIONING OF NUCLEAR SUBMARINES IN THE KAMCHATKA REGION

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In the Kamchatka region the retired Nuclear Submarines (NSs) pending/under dismantlement along with made up Reactor Compartment (RC) units are stored afloat at the Federal State Unitary Enterprise (FSUE) North-Eastern Regional Center (NERC) and at the former naval base in Krasheninnikov Bay.

As of September 2004 (Table 1), 23 NSs were taken out of service in the Kamchatka region for subsequent dismantlement. Five of them have been already dismantled down to three-RC units for subsequent waterborne storage; 6 NSs under different dismantling stages are stored at NERC water area.

TABLE 1. Actual status of NS complex decommissioning in the Kamchatka region (as of September 2004)

NSs withdrawn from service for complex decommissioning	23
NS dismantled down to three-RC units	5
NS under dismantlement	6
Afloat-stored NSs pending dismantlement	12
Non-defueled NSs	11
Defueled NSs	7
Total defueled NSs	12

So far 7 NSs have been defueled, 11 NSs pending/under dismantlement are to be defueled in the near future.

At present dismantlement of the following NSs is performed at NERC:

– *Echo-II*-class – 1 NS;

– *Charlie-I*-class – 4 NSs;

– *Delta-III*-class – 1 NS.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 169–178.

The following NSs are pending dismantlement:

- *Echo-II*-class – 2 NSs;
- *Charlie-I*-class – 3 NSs;
- *Victor-III*-class – 6 NSs; and
- *Oscar-II*-class – 1 NS.

The actual condition of buoyancy systems of 5 retired NSs stored afloat in Krasheninnikov Bay is considered as “unsatisfactory”.

The buoyancy and floodability systems of retired NSs pending complex decommissioning require continuous servicing and maintenance of “technical preparedness” to prevent unauthorized sinking.

NERC has no its own defueling facilities. The draft of the Federal Target Program for Complex Decommissioning of Nuclear Submarines and Nuclear-Powered Surface Ships (NPSS) - that has already undergone the Governmental Environmental Impact Assessment - does not provide for establishment of land-based SNF unloading facilities in the Kamchatka region because of high seismic activity. Consequently, NS defueling in Kamchatka is to be performed only using a Floating Servicing Vessel (FSV) of *Malina*-class based permanently in Primorskiy Kray.

Thus to perform operations in the Kamchatka region, the *Malina*-class FSV has to run more than 1300 miles.

By now 11 non-defueled NSs have been stored afloat in the Kamchatka region including:

- *Victor-III*-class – 6 NSs (12 cores);
- *Charlie-I*-class – 1 NS (1 core);
- *Echo-II*-class – 2 NSs (4 cores);
- *Oscar-II*-class – 1 NS (2 cores); and
- *Delta-III*-class – 1 NS (2 cores).

Thus one needs: first - to unload Spent Nuclear Fuel (SNF) from 21 nuclear reactors, next - remove it from Kamchatka to Primorskiy Kray and then - forward SNF to PA “Mayak” for reprocessing.

The Coastal Maintenance Base (CMB) in Gorbushechya Bay, Kamchatka, by now transferred under the jurisdiction of the Russian Federal Agency for Atomic Energy (Rosatom) for environmental remediation, was initially designed to accept and store temporarily Solid Radioactive Waste (SRW); neither acceptance nor storage of SNF/Liquid Radioactive Waste (LRW) was provided at the CMB at all.

At present SRW generated during NS dismantlement are placed into floating RC units according to the NS designer documentation and the related guiding document on SRW management.

Three-RC units are actually stored afloat within Temporary Storage Center (TSC) in Razboinik Bay, Primorskiy Kray.

A Long-term Storage Center (LSC) of the Pacific region - an open-air land-based storage pad on Ustrishny Cap close to Razboinik Bay - is under construction. According to the results of the performed Feasibility Study (FS) and the environmental impact assessment, such layout was recognized as the best option from the technical, economic and environmental safety standpoint.

Construction of special pads for RC storage at the enterprises-executors was considered as an inexpedient and extremely expensive measure. Consequently, such activities were not included into the Program for Complex Decommissioning of NSs and NPSSs.

Three-RC units made up at NERC are to be transported from Kamchatka to Primorskiy Kray for temporary waterborne storage, subsequent making of one-compartment RC and installation at the on-shore pad of the LSC (Figure 1).

Figure 1. Three-RC units of former NSs stored afloat in Razboinik Bay

The actual technical condition of NSs stored afloat at NERC and in Krasheninnikov Bay gives no way of performing their haulage to Primorskiy Kray if no additional buoyancy-supporting measures are taken. With due regard for a long distance to run until the destination station, haulage of retired NSs and/or RC units using standard towboats appears to be a rather sophisticated and high-risk challenge.

Transfer to Primorskiy Kray of NSs for dismantlement and of three-RC units for storage at TSC using a special floating transportation dock seems to be the only possible option.

The actual industrial capacities of NERC allow dismantling 2 to 3 NSs per year followed by making up of three-RC units.

To optimize complex decommissioning of NSs in the Pacific region, the following factors should be taken into account:

- factual location of retired NSs;
- need to complete complex decommissioning of all retired so far NSs by the turn of 2008;
- financial resources of the State Customer for complex decommissioning of NSs; and
- the availability of experienced personnel at the Pacific region's enterprises.

With consideration for the above factors, the following paces of NS dismantlement at the Pacific region's enterprises appear to be the most justified:

- Far East Plant "Zvezda" - 4 NSs per year;
- Shipyard #30 of the Ministry of Defense of the Russian Federation - 1 NS per year; and
- NERC - 2 NSs per year.

However even under adequate financing all retired so far NSs in the Pacific region could be defueled and dismantled only by 2010 at the earliest.

Few years ago by special request of the Russian Ministry for Atomic Energy (Minatom, presently Rosatom) the Research and Development Institute of Power Engineering (NIKIET) performed FS of different options of retired NS defueling and dismantlement in the Kamchatka region.

To select the best option of NS transport and management cycle in Kamchatka, the following factors were taken into account:

- NERC NS-dismantling capacities;
- capacities of "PM-74" FSV on SNF unloading and on-board storage;
- NERC location on the Kamchatka Peninsula;
- SNF management procedures and transportation cycle;
- location of TSC for afloat-stored RC units in the Pacific region;

- location of LSC for RC in the region;
- possibilities of transporting NSs, three-RC units and one-RC units to their storage areas (haulage or transfer within a floating dock);
- necessity of restoring/manufacturing appropriate equipment to support different options of the transport and management cycle;
- cost estimates of different options of the transport and management cycle;
- social factors in the Kamchatka region;
- specific interests of the Russian Navy, and some other factors.

From the environmental and engineering standpoint, storage of three-RC units is more advantageous option as compared to that of non-dismantled NSs: one needs by far less servicing, no crew on board, no nuclear hazard thanks to SNF unloading and considerably shorter quayage.

The options of NS dismantlement at NERC down to one-RC unit and SNF transportation by some other special vessel (e.g., a container ship) were not considered at all because in such a case one would need constructing at NERC rather sophisticated transit infrastructure and RC on-shore interim-storage infrastructure along with building of a special vessel (container ship) to transport SNF.

Development of SNF temporary-storage capacities in the Kamchatka region using the available stock of containers represents an unpromising variant either because after SNF reloading to FSV and removal for reprocessing very expensive empty containers would have no prospects for further use/removal due to high contamination level.

Finally, the following three options of complex decommissioning of NSs in Kamchatka were accepted for examination:

Option 1. All NSs of the Kamchatka region are dismantled at NERC after SNF unloading and removal using the *Malina*-class FSV. All made up three-RC units are delivered using a transport dock to the temporary storage center (or to the one-RC unit making up center) in Primorskiy Kray.

Option 2. NSs in poor technical condition and those under three-RC unit making up are kept at NERC until completion of 3-RC-unit-making operations. The remaining 6 NSs (all NSs of *Victor-III*-class) after restoration of their buoyancy are transported in floating dock to Primorskiy Kray for subsequent defueling and dismantlement. Three-RC units are shipped using floating dock to TSC or to the one-RC unit making up center. The remaining NSs are defueled, their SNF being removed using the *Malina*-class FSV.

Option 3. Defueled NSs in poor technical condition and those under three-RC unit making up are kept at NERC until completion of 3-RC-unit-making operations. The remaining NSs after restoration of their buoyancy are transported in floating dock to Primorskiy Kray for subsequent defueling and dismantlement. Three-RC units are shipped using floating dock to TSC or to the one-RC unit making up center. Because the remaining NSs have been already defueled, no FSV would be necessary.

When developing FSs for the above alternatives, all expense items and the related engineering aspects were taken into account (NS and RC unit keeping afloat, fuel, lease of towboats, *Malina*-class FSV preparing and running, RC making up and defueling using on-shore defueling facilities, gains from sales of products of complex decommissioning in Kamchatka and Primorskiy Kray, FSV operation in Kamchatka region once every two years at best due to its problematic technical condition, need of performing scheduled repairs and restoration of different systems and equipment, and so forth).

The terms of works were also considered as the major-importance factor. If all retired NSs in Kamchatka were dismantled at NERC, the process would last until 2014-2015, whereas in case of transfer of some retired NSs to Primorskiy Kray, one would have good chances to complete the operations by 2009-2011. With consideration for poor condition of NS hulls and buoyancy systems, the risk of initiating different-type incidents increases every year; the expenses for NS servicing and keeping afloat also increase. The cost for maintaining waterborne storage of retired NSs in the "pending-decommissioning" condition in the Kamchatka region is above that in other concerned regions provided that similar-class nuclear submarines are compared.

The FS results are summarized below:

The option of dismantling all retired-in-Kamchatka NSs at NERC represents the most expensive and long-lasting alternative.

The second option providing for transfer of 6 *Victor-III*-class NSs to Primorskiy Kray for dismantlement could be realized in the shortest possible time, the overall cost being estimated as "intermediate".

The third option presuming transfer of all non-defueled NSs to Primorskiy Kray for complex decommissioning is the less expensive and could be realized under the "intermediate"-duration period.

Only the option of transferring 6 NSs to Primorskiy Kray would allow completing complex decommissioning of all retired so far NSs in Kamchatka by 2010. Defueling of the NSs under consideration could be completed in 2008. If accepting this option, transfer of RC units from Kamchatka to TSC in Primorskiy kray could be completed by 2010-2011.

According to the performed FSs, the option of transferring all retired NSs to Primorskiy Kray for defueling would make it possible to gain economic benefits and reduce the terms of works by 3 year. However with due regard for both full load of the actual capacities at the Primorskiy Kray's enterprises-executors and social factors in the Kamchatka region (including expediency of maintaining already available infrastructure), the option of transferring 4 to 6 NSs to Primorskiy Kray appears today the best alternative. Such option would allow gaining economic benefits and reducing by 5 years the terms of both NS dismantlement and RC unit transfer to TSC.

Within the framework of the international cooperation under the G8 Global Partnership Program in the complex decommissioning and environmental remediation area, a project for complex decommissioning of retired NSs in the Kamchatka region has been prepared and submitted for consideration by the countries-participants.

The project comprises the following main phases:

1. Development of design and engineering documentation to perform complex decommissioning of NSs of *Charley-1, Echo-II, Victor-III* and *Oscar-II* classes.
2. Transfer of 4-6 *Victor-III*-class NSs from their actual afloat-storage locations in Kamchatka to Primorskiy Kray.
3. Performing dismantling/mounting operations accompanying NS defueling.
4. Defueling of NSs to be dismantled.
5. SNF shipment in special shrouds on board of *Malina*-class FSV from Kamchatka Peninsula to the SNF transshipment station in Primorskiy Kray.
6. Preparing, removal and reprocessing of SNF unloaded from reactors of NSs to be dismantled.
7. Dismantlement of NSs of *Charley-1, Echo-II, Victor-III* and *Oscar-II* classes down to three-RC units.
8. Shipment of three-RC units of former NSs dismantled at NERC (Kamchatka) to the TSC in Primorskiy Kray for waterborne storage.

The tentative time schedule of the project implementation is given in Table 1.

TABLE 2. Tentative time schedule for implementation of the project for complex decommissioning of NSs taken out of military service in the Kamchatka region

N	Work type	Due date					
		2005	2006	2007	2008	2009	2010
1.	Waterborne storage of retired NSs and made up RC units						
2.	Development of design and engineering documentation for complex decommissioning of NSs withdrawn from military service						
3.	Development of a project for *Victor-III*-class NS transfer from Kamchatka to Primorskiy Kray						
4.	Development of a project to transfer RC units from Kamchatka to Primorskiy Kray						
5.	Transfer of *Victor-III*-class NS from Kamchatka to Primorskiy Kray						
6.	Performing repair and maintaining appropriate technical condition of *Malina*-class FSV to support SNF management						
7.	Performing NS defueling and the accompanying dismantling/mounting operations						
8.	SNF removal and reprocessing						
9.	NS dismantlement down to RC unit making up						
10.	Transfer of RC units to special temporary waterborne storage center in Primorskiy Kray						

The integral cost of the project is estimated at about US$ 230 million.

The project implementation will make it possible to perform and complete by 2010 environmentally safe dismantlement of NSs withdrawn from military service and based in Kamchatka (*Charley-1, Echo-II, Victor-III* and *Oscar-II* classes).

The measures to be taken under the Project will ensure environmental safety in the region thanks to transfer of the above-class NSs from the "nuclear-hazardous" category to the "radiation-hazardous" one followed by their dismantlement and thus elimination of the risk of unauthorized sinking.

Appropriate implementation of the project provisions will allow reducing the number of radiation-hazardous facilities requiring adequate servicing and safety-ensuring measures.

Complex decommissioning of *Oscar-II*-class NS merits a special consideration. Wirth due regards for individual peculiarities of the enterprises-executors and the NS itself, one may expect a considerable distinctions between the cost of work in Kamchatka region and Primorskiy Kray. Tentative work cost is estimated at about US$ 2 million, provided that the NS will be towed and no FSV will be used. Though no such-class NS has been dismantled in the Pacific Russia yet, positive experience collected in the Northwest region (dismantlement of two *Oscar-I*-class NSs under the UK's financial support within the frames of the G8 Global Partnership Program) could be used. It would be wise to consider complex decommissioning of *Oscar-II*-class NS in the Pacific Russia as an individual project to be implemented under the international cooperation programs at one of the Pacific Russia's enterprises selected on a competitive basis.

So far the concept for environmental remediation of Coastal Maintenance Bases (CMBs) in the Pacific region (including the CMB in Gorbushechya Bay, Kamchatka) has been developed and approved.

Principal approaches to remediation of the CMB in Gorbushechya Bay have been developed with consideration for the CMB use for purposes of NS complex decommissioning and running in the Kamchatka region in order to fulfill the following functions:

- Temporary storage of FSVs, tankers and other-type Maintenance Vessels (MVs) on the floating berth;
- Transfer to/from MVs of special equipments to support NS complex decommissioning activities; and
- Storage of SRW generated during NS running and complex decommissioning.

For proper functioning of the CMB during the remediation period, one should perform renovation and support adequate functioning of the following elements of its infrastructure:

- Floating service dock (construction #2);
- Radiation control post (constructions #21 and #22);
- SRW storage facility (construction #16);
- Reloading equipment storage facility (construction #5); and
- Power supply and economic activity structures (constructions ##62, 9, 25 and 29).

Storage facilities #3 and #19 for SRW are to be eliminated.

To ensure safety of works during temporary functioning and remediation of the CMB, the establishment of a radiation monitoring system and of a physical protection system is necessary.

When addressing the issues of radioactive waste and SNF management during complex decommissioning of NSs in Kamchatka, the following problem tasks should be emphasized:

- Need of involving the *Malina*-class FSV – permanently based in Primorskiy Kray - into NS defueling operations in Kamchatka and subsequent SNF removal from the Kamchatka region;
- Insufficient industrial capacities and high relative expenses at NERC hindering the completion of complex decommissioning of all retired so far NSs by 2010;
- Lack of a prepared floating dock to ship both NSs and made-up RC units to Primorskiy Kray for completing their dismantlement and RC making up followed by storage at a special pad of LSC; and
- Environmental remediation of the former naval CMB in Gorbushechya Bay.

It should be stressed that transfer of both retired NSs and made up RC units from Kamchatka to Primorskiy Kray for dismantlement and temporary storage (or RC unit making up for on-shore storage at the LSC) represents the greatest challenge under any selected option.

As mentioned above, haulage of NSs and RC units using towboats is a very sophisticated high-risk operation.

Transfer of NS and RC units using an appropriate floating dock represents a more promising option from the technical and environmental safety considerations.

Potential investors may regulate the issues of interfaces under the considered scope of activities with Rosatom.

There is no doubt that safe complex decommissioning of retired NSs in the Kamchatka region, RC placing for long-term storage at the Primorskiy Kray's LSC and environmental remediation of the CMB in Gorbushechya Bay in the shortest possible time will contribute to a decrease in the risks of environmental incidents in the Pacific region.

Adequate consideration of these challenges, opportune investment of the most topical work areas under the international cooperation and necessary technical support will allow avoiding considerable material and environmental damages and the related expenses for elimination of their consequences.

NUCLEAR AND RADIATION SAFETY DURING LONG-TERM STORAGE OF SPENT NUCLEAR FUEL OF LAND-BASED REACTOR FACILITY STANDS-PROTOTYPES 27/VT AND KM-1

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Abstract

Within the framework of the ISTC Project #2710p between the Russian Research Center "Institute for Physics and Power Engineering" (RRC IPPE, below IPPE) and the Brookhaven National Laboratory of the US Department of Energy an investigation has been performed addressing the issues of nuclear and radiation safety when storing Spent Nuclear Fuel (SNF) of land-based stands-prototypes 27/VT and KM-1 constructed during the period of developing lead-bismuth coolant reactor installations for *Alpha*-class Nuclear Submarines (NS). The detailed results of the investigation can be found in the Final Report under this Project [1].

This paper presents the main results of the study with a focus on: -examination of the effects on the system reactivity of moisture ingress into storage facility of the Spent Removable Unit (SRU) of KM-1 stand and -analysis of potential implications of their resulting emergency situations.

On this basis health radiation risk assessments for population of the nearby territories, including Finland, Estonia and Latvia, have been performed.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 179–194.

1. Land-based Stands-prototypes of Power Reactor Installations (PRI) of Liquid-Metal Coolant (LMC) NSs

From 1959 to 1976 in IPPE (Obninsk) 27/BT land-based LMC stand– a NS PRI prototype - was in operation, eutectic lead-bismuth alloy being used as the reactor primary circuit coolant. Nuclear fuel represented a beryllium-uranium alloy dispersed over a beryllium matrix. In fuel elements highly enriched uranium was used. Rod-type fuel elements comprised fuel briquettes confined into steel claddings; the integral number of fuel elements in the core equaled about 2600. The reactor rated heat power equaled 70 MW [2]. The first reactor core lifetime of about 2000 effective hours was worked out over 1959÷1961. Two months after power resource expiration the core in the form of a single removable unit was unloaded from the reactor, the decay heat equaling 20÷25 kW, and placed into a special long-term cooldown storage facility inside the reactor building. Cooldown was performed via air blowing through the core and release into a special ventilation system via radioactive aerosol catching filters. In 1976 the first-lifetime core of 27/VT stand was dismantled in a special reactor building compartment, and its condition was examined; after that the whole set of fuel elements was transferred to the IPPE's central nuclear fuel storage facility. Presently fuel elements are stored under conditions complying with the requirements of public supervisory authorities on nuclear and radiation safety, their physical protection being adequate. Work is conducted to prepare transportation of the stored set of fuel elements in special containers to PA "Mayak" for reprocessing. The second core lifetime of that reactor (1966÷1976) made up about 3000 effective hours. However due to some technical reasons core running was interrupted after expiration of 50%-lifetime. In 1976 the core was unloaded from the reactor and soon dismantled by assembly at IPPE. Similar to the set of fuel elements of the first core lifetime, the second-lifetime fuel elements were transferred for storage to the IPPE's central nuclear fuel storage facility and are to be forwarded to PA "Mayak" for reprocessing. The building housing SNF of 27/VT stand reactors has a license of running stationary facilities designed to store nuclear materials. As set out in the concluding nuclear safety statement, any potential emergency situation accompanied by fuel element package flooding would not result in increase of K_{eff} over 0.95.

Land-based stand KM-1 – prototype of PRI of *Alpha*-class NS (design 705) with Steam-Producing Installation (SPI) OK-550 - was in operation from 1978 till March 1986 at the NITI site (Sosnovy Bor, Leningrad region) [3]. The stand reactor core worked out 105 % of the lifetime with the integral heat energy generation of about 780 000 MW·h. In autumn 1987 the integral removable unit including the core was unloaded from the reactor and placed into the long-term cooldown storage facility, the removable unit with the core being placed into special steel "cup" filled with pure lead-bismuth eutectic. There is a water-cooling system outside the "cup" to remove decay heat. A control is performed over coolant temperatures in the core along with dose control in the storage room. Information is displayed on PRI operator's control panel. At present coolant temperature does not exceed 60÷80°C under disconnected water-cooling system, and thus the core is located within a "frozen" lead-bismuth alloy ($t_{melt.}$ = 125°C).

Rods of the Control and Protection System (CPS) inserted in the core are immobilized, K_{eff}< 0.90.

The levels of accumulated radioactivity in individual fuel elements and the whole reactor cores of the stands-prototypes, as of December 2003 and December 2010, are demonstrated in Table 1.

TABLE 1. Integral accumulated activity of actinoids and fission products in reactor cores, and the average activity in individual fuel elements of the stands-prototypes of NS SPI

Stand-prototype	Components	A, Bq		A, Bq/fuel element	
		31.12.2003	31.12.2010	31.12.2003	31.12.2010
27/VT First lifetime	Fission products	7.75E+14	6.48E+14	2.83E+11	2.37E+11
	Actinoids	1.52E+13	1.32E+13	5.55E+09	4.84E+09
	Sum	7.90E+14	6.61E+14	2.88E+11	2.42E+11
27/VT Second lifetime	Fission products	1.02E+15	8.35E+14	3.74E+11	3.05E+11
	Actinoids	1.69E+13	1.41E+13	6.18E+09	5.16E+09
	Sum	1.04E+15	8.49E+14	3.80E+11	3.10E+11
KM-1	Fission products	9.75E+15	7.75E+15	2.32E+12	1.84E+12
	Actinoids	1.59E+14	1.29E+14	3.79E+10	3.08E+10
	Sum	9.91E+15	7.88E+15	2.36E+12	1.87E+12

2. Unloading and Storage of SNF of KM-1 Stand Reactor

All preparative operations, unloading and Spent Removable Unit (SRU) installation into the storage facility were performed in compliance with the procedure developed at the Research Institute for Nuclear Technologies (NITI) under IPPE's scientific supervision over the unloading operations. SRU was placed into the storage facility especially designed for its cooldown and long-term storage (Fig.1).

The storage facility consists of the following main units:

– an inner steel "cup" (1) designed to house SRU. The "cup" was filled with a specified amount of lead-bismuth eutectic alloy providing for complete submerging of the SRU core therein. Prior to place SRU into the storage facility the alloy was heated up to 150°C using a heating system and melted. Since SRU installation, the core cooldown has been performed using the storage facility cooling system;
– a storage facility case (2) in the form of a special nest equipped with heating-cooldown coils (3). To perform heating of the storage facility, steam is delivered to the coils, its temperature equaling190°C; for cooldown purposes water of up to +20°C temperature is used. The storage facility's case has an additional side boron carbide biological shielding;
– a steel-coated vault within bulk concrete (4) to install the storage facility case;
– a special box to perform servicing of the storage facility systems (5) installed within the bulk concrete above the storage facility case. From above the box is covered with a concrete plug (6) sealed with a rubber edge (7) to eliminate water penetration inside the box;
– the storage facility is equipped with a drainage system (8) to drain water out of the vault in case of depressurization of tubes of the heating (cooldown) system;

- to perform the control over eventual water appearance in the vault, a special channel is established (9) wherein a tube-jacket is inserted to arrange a Water-Indication Sensor (WIS). There is a hole (10) in the upper part of the storage facility connected to a duct of special ventilation system for air removal.

The storage facility arranged within a “process area” of the building is protected against atmospheric precipitations; its steel surfaces can be easily decontaminated. Positive temperatures are kept in the room. The storage facility is equipped with an intruder alarm, routine lighting and emergency lighting being controlled by duty shifts in the round-the-clock mode. Decay heat in SRU core is due to: beta- and gamma-radiation of fission products; radionuclides generated in CPS rods; ^{60}Co generated in steel constructions of the core and steel SRU shroud. By the SRU unloading instant the decay heat was estimated at 4.2 kW; according to the present-day (2004) experimental and calculated estimates, it makes up ~ 1.1 kW.

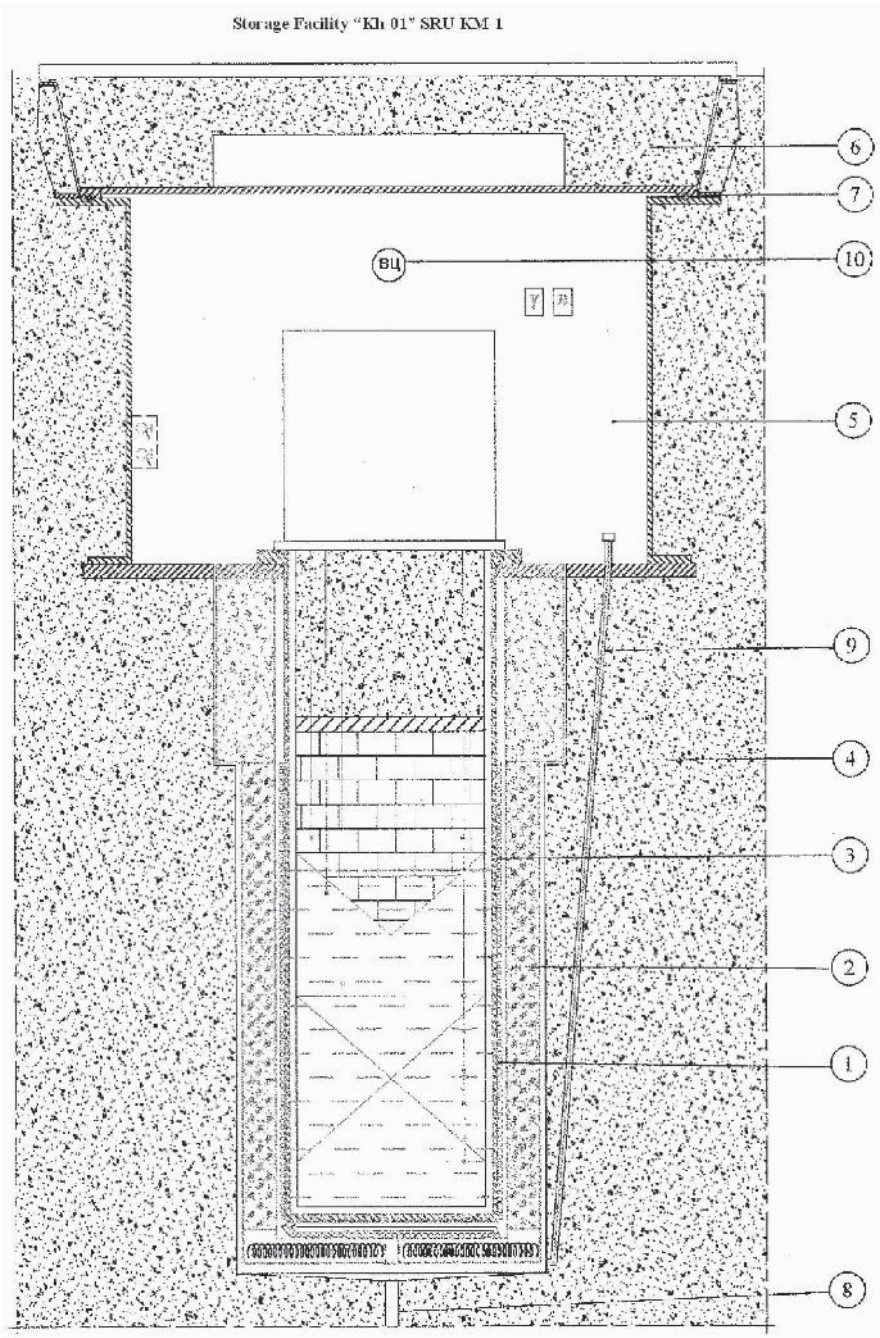

Figure 1. SRU storage conditions

3. Nuclear and Radiation Safety during SRU Storage

Subcriticality of spent core of KM-1 stand stored at special storage facility calculated at NITI in 1987 and coordinated with Nuclear Safety Department of IPPE was estimated as “complying with the nuclear safety requirements, $K_{eff} \leq 0.92$”. Nuclear and radiation safety of SRU storage is ensured via:

- reliable fixation of CPS rods within “frozen” alloy;
- excluding of water ingress into the core via its full submerging into lead-bismuth alloy kept in “frozen” condition due to cooling-water delivery to the storage facility tube shielding;
- proper operation of steady-state equipment controlling the radiation situation within storage facility;
- SRU regular (yearly) physical control;
- nuclear and radiation safety work running in compliance with the NITI’s standard and technical documentation in force;
- on-line control over nuclear safety by specialists of NITI’s Nuclear and Technical Safety Department;
- control over radiation safety by specialists of NITI’s Experimental Radiation Safety Department; and
- high professional skill of personnel running the storage facility.

The controls over neutron flux density in the WIS tube is performed via two channels using scalers with SNM-12 counters allowing operation within the temperature range from -50°C to +100°C under gamma-radiation impact ≤10 Sv/h (1000 R/h).

To ensure continuous radiation control over SRUs, the room housing the storage facility is equipped with DG-3 gamma-sensors (the range of gamma exposure dose rate measurements within 1.0 μR/s÷$1.0 \cdot 10^3$μR/s) and DNP-2 intermediate neutron sensors (the range of intermediate neutron flux density measurement within 10^5 neutr/($m^2 \cdot s$) ÷ 10^8 neutr/($m^2 \cdot s$)) with information output to UIM-2-2 gage installed in duty operator’s room. UIM-2-2 gage has the following signaling thresholds: channel with DG-3 sensor – 5 μR/s and channel with DNP-2 sensor – $3 \cdot 10^5$ neutr/($m^2 \cdot s$).

Recurrence control over air contamination in the storage facility by alpha- and beta-active aerosols is performed using portable blowers.

No case of signaling threshold excess in the two above channels has been recorded over the SRU storage period at the storage facility. No case of the storage facility air contamination by alpha- and beta- active aerosols has been detected either.

SRU temperature is controlled at regular intervals via three independent channels, two of them being standard ones with industrial sensors (TKhT-type chromel-cupel thermocouples) and the third channel being the experimental one (TkhA-type chromel-alumel thermocouple made at IPPE).

The control over water appearance in the storage facility vault is performed using two standard signaling devices which sensors (WIS) are arranged in tubes of special channels in the storage facility bulk concrete (Fig.1), whereas warning lights and chime are brought out to duty operator's control panel. The channel operability is checked once every three months. So far no water has been recorded in the vault at all.

A variety of technical-organizational measures undertaken to store SRU of KM-1 stand allows for the control over nuclear and radiation safety during storage to be performed in compliance with the standards and regulations in force.

4. Nuclear and Radiation Safety in Case of Water Ingress into Reactor Core of KM-1 Stand

The results of studying the issues of safe storage of SRU with OK-550 reactor installation of KM-1 stand are considered below. Under the study both possible generation of contraction cavities in the core during "freezing" of lead-bismuth alloy housing SRU at the storage facility and its subsequent emergency flooding were examined. To calculate the dynamics of physical and heat engineering characteristics, the following parameters were estimated in advance: initial core subcriticality, the contraction cavity size within the core, CPS rod number and location over the core height and reactivity due to hypothetical ingress of different water quantities into the contraction cavity. The dynamics of power and temperature variations under emergency water ingress into the contraction cavity was calculated, and the implications of such-type emergency situation were analyzed.

According to the performed calculations, K_{eff} of SRU in reactor vessel with completely submerged rods makes up ≤ 0.90. For SRU stored in "dry" conditions, K_{eff} equals 0.869±0.004; in case of the "flooded-with-water" storage facility K_{eff} < 0.882.

K_{eff} of reactor in critical condition calculated using the same procedure made up 0.98. With due regard for calculation errors, when examining emergency situations in the storage facility, K_{eff}~0.89 was taken as the initial value, the subcriticality making up 0.11 (11 %) or ~16,7 β_{eff} (according to calculations, β_{eff} = 0.66 %).

After SRU placing at the storage facility, core was cooled down using standard cooling systems until coolant solidification within the steel "cup" housing SRU. Under the accepted cooling procedure, coolant solidification proceeded from the core periphery to its center. Consequently, the probability of a contraction cavity generation was rather high in the central core area solidifying at the latest moment due to decay heat. Moreover, solidified alloy could have contained pores filled with gas dissolved in liquid alloy. In case of potential water ingress into voids free of lead-bismuth coolant (outlet mixing chamber, scram rod shrouds) and a protracted SRU storage under such conditions, the provability exists for water penetration into such cavity (or pores) via gaps generated due to possible detachment of solidified coolant from CPS shrouds and thermocouples. In a hypothetical case of water ingress into the core contraction cavity (pores) K_{eff} of reactor would increase.

To estimate the cavity size, it was accepted that the volume of lead-bismuth alloy in the steel “cup” housinge SRU of KM-1 SRI made up 0.70 m^3 at the alloy-melting temperature, whereas the volume of pores generated during alloy solidification equaled 1.8 % of the integral alloy volume (i.e., 0.0126 m^3 =12.6 l), according to IPPE’s measurements. Thus the contraction cavity volume was determined at 12.6 l. Considering that the alloy filled the inter-fuel-element space of the core, the contraction cavity size (under the volumetric proportion of coolant of 0.304) would extend over a part of the core volume of 41.45 l. During coolant solidification its detachment from fuel element claddings, CPS shrouds and thermocouples, along with micropore generation in the alloy itself due to admixtures and slag would be possible. Thus there is a risk of water penetration into the contraction cavity located in the very core center in a hypothetical case of the storage facility emergency flooding.

The results of performed reactivity calculations due to water ingress into the core were used to estimate the dynamics of power and temperatures. The calculations were performed under 2D (R,Z)-geometry [7] for different water amounts (from 1 kg up to 12.6 kg), i.e., until the contraction cavity filling with water in full. The volume of the central cavity in the core was transformed into an "equilateral" cylinder, its diameter equaling the height. The calculations were performed using the 21-group system of constants [5] taking into account the neutron thermalization phenomenon. The 21st thermal group was subdivided into 17 subgroups with E_{gr} = 1.01 eV. In the calculated model CPS absorber rods were modeled by homogeneily-mixed boron with blocked sections simulating energy dependence of the absorption cross-section by heterogeneous rods of the CPS. Nuclear concentration of boron was selected in such a way as both the calculated K_{eff} and experimental K_{eff} in standard reactor with side beryllium reflector would be equal.

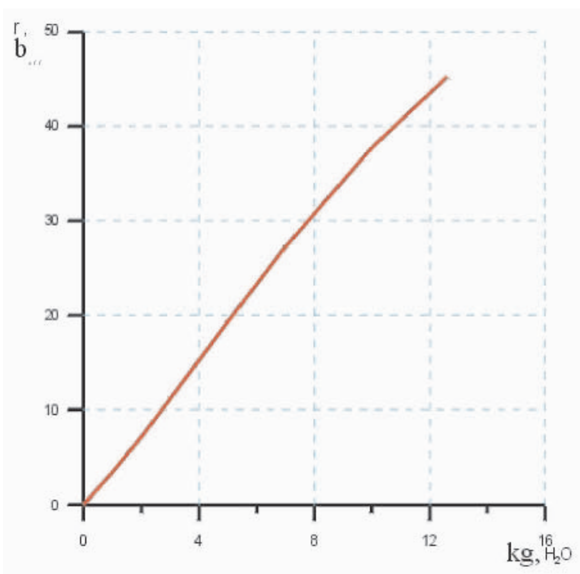

Figure 2. Dependence of reactivity on water penetration into contraction cavity generated in the core center

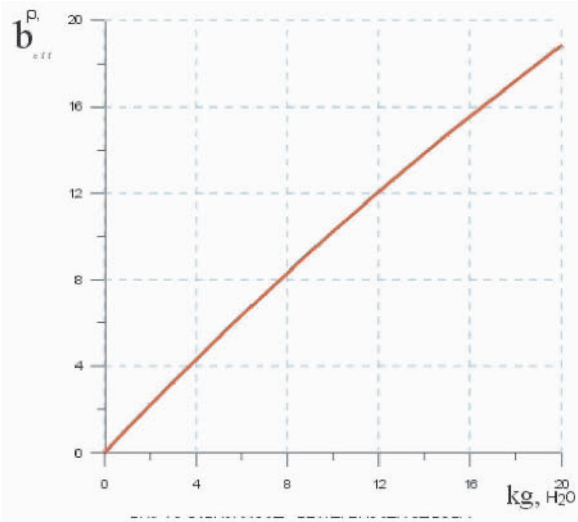

Figure 3. Dependence of reactivity on uniform water distribution over the core

Using the above-described calculated model, a dependence of reactivity on the amount of water penetrated into the central contraction cavity in SRU core center within the storage facility was obtained, the results of calculations being depicted in Fig.2. By analogy with the previous calculations, those calculations were performed in 2D (R,Z) geometry [7] using 21-group system of constants [5] taking into account thermalizaiton of neutrons and similar-way consideration of CPS rods. As distinct from the previous calculations, it was accepted in the given case that different amounts of water penetrating inside the core were distributed uniformly over its volume. The results of calculations are illustrated in Fig.3.

During SRU storage the risks of its flooding with water and water penetration inside the core is eliminated thanks to the application of appropriate engineering devices and organizational measures. Hypothetical water ingress into the core would result in reactivity increase as the result of positive effect of reactivity due to water penetration into the core, SRU subcriticality value making up ~ 16.7 β_{eff}. To attain criticality, the amount of water entering the core should equal ~ 4.4 l.

To determine physical and heat engineering SRU parameters, a calculated study of potential emergency situations related to water penetration into the core was performed. Mathematical description of the processes was based on dot description of both the neutron kinetics and the equations for heat transfer in the storage facility container (under such "container" SRU arranged within steel "cup" with "frozen" alloy was understood).

The neutron kinetics was described with due regard for six groups of delayed neutrons. The core reactivity changes were determined by temperature effects of reactivity of fuel elements and the alloy as well as by water penetration into the core.

Heat pick-up from "the container" outer surface was performed due to natural air convection. In the model under the "cavity" the inter-fuel-element space of a free-of-alloy core part was understood. The flow path between "the cavity" and the surrounding space was realized with assigned hydraulic resistance and cross-section.

When performing calculations, the following parameters were taken as the initial condition of the "container" with SRU: core subcriticality 16.7 β_{eff}; intensity of the own neutron source $4{\cdot}10^9$ N/s; decay heat power 1.1 kW; surrounding-air temperature 20°C; calculated mean temperature in the core 65.3°C.

When determining relative hear power values, the rated heat power of the core N_{rat}=149 MW was used, the calculated power value in the initial condition equaling $0.31{\cdot}10^{-8}$ N_{rat}. To calculate water penetration rate into the core "cavity", water head was taken equal to one meter of liquid column, whereas the flow path cross-section between the core "cavity" and the surrounding area to 1 cm^2. Changes in water flow rate during calculations were attained through varying the hydraulic resistance between the core "cavity" and the environment. The dynamics of the "container" physical and heat parameters was considered for two values of water ingress rate into the core: 0.0038 kg/s and 0.0085 kg/s.

Calculated changes in SRU parameters during water penetration into the core with 0.0038 kg/s flow rate is demonstrated in Fig.4, and with 0.0085 kg/s flow rate in Fig.5.

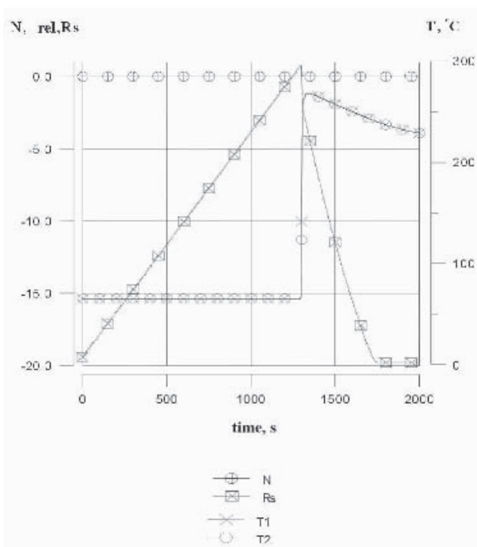

Figure 4. SRU parameter variations in case of water ingress

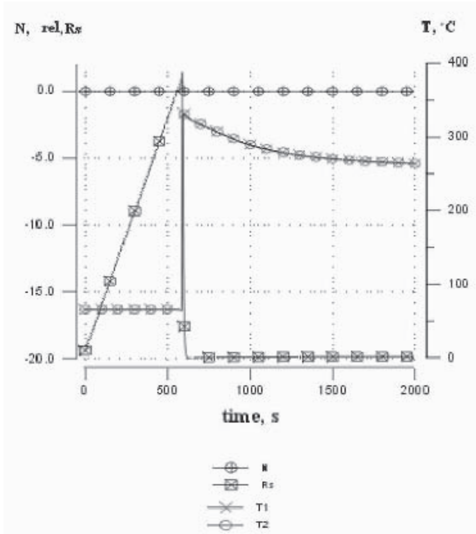

Figure 5. SRU parameter variations in case of water ingress

N – relative heat power released in the core;

T1 – mean temperature of fuel elements contacting the alloy, °C;

T2 – mean temperature of alloy in the core, °C.

R_s – integral value of reactivity.

In case of water ingress into the core with a rate of 0.0038 kg/s, the core criticality is attained ~ 1245 s after water ingress onset, the core power being $0.225 \cdot 10^{-6}$ N_{rat} (Fig.4). Maximal reactivity of 0.787 β_{eff} is realized at the 1297th s of the calculated process, the core power equaling 0.0855 N_{rat}. It is at this instant that rapid temperature increase in the core begins resulting in a decrease in reactivity due to negative temperature effects of the reactivity of both fuel elements and the alloy. Water boiling in the core begins at about the 1333rd s of the transient. Drastic decrease in water mass in the core transfers the latter to subcritical condition. Full water evaporation occurs at about the 1750th s of the calculated time.

The core heat power reaches its maximum of 0.806 N_{rat} at the 1303rd s of the calculated time. Mean temperatures of fuel elements contacting the alloy reach a maximum of 268°C at the 1338th s of the process. Alloy melting in the core proceeds during ~ 4 s (from the 1300th s to the 1304th s of the calculated time). Maximum value of the mean temperature of fuel elements contacting steam reaches 362°C. The integral number of fissions of ^{235}U nuclei over the calculated time 2000 s makes up $1.82 \cdot 10^{19}$ fissions.

In case of water ingress into the core with a rate of 0.0085 kg/s (Fig.5) the core attains criticality at the 558th s of the transient after water ingress onset. Maximum reactivity value of 0.898 β_{eff} is attained in the calculation at the 583rd s, the core power equaling 0.0442 N_{rat}. During this period temperatures in the core rapidly increase that results in reactivity decrease due to negative temperature effects of reactivity.

Water boiling in the core begins at about the 587th s of the transient, and the core passes to subcritical condition. Full water evaporation in the core occurs at about the 605th s of the transient.

Heat power in the core reaches its maximum (~ 1,46 N_{rat}) at the 586th s of the calculated time. Mean temperatures of alloy-contacting fuel elements attain their maximum of 333°C at the 593rd s of the transient.

Melting of lead-bismuth alloy in the core proceeds during ~ 2 s of the transient (from the 585th s to the 586th s). Maximal value of mean temperature of steam-contacting fuel elements reaches ~ 470°C.

The integral number of fissions of ^{235}U nuclei over the calculated time 2000 s makes up $2.24 \cdot 10^{19}$.

5. Consequences of Pulse Power Changes in SRU of KM-1 Stand

Spent Removable Unit (SRU) of KM-1 stand with 105%-lifetime power output has been stored at the storage facility since 1987. Over the storage period double melting-freezing of Pb-Bi alloy took place that could have resulted in fuel element deformations with partial damage of claddings.

The energy of pulse power change corresponding to $2.2 \cdot 10^{19}$ fissions is sufficient to heat fuel up to 470°C, produce melting of Pb-Bi alloy and boiling of water penetrated therein. Steam bubble passing via molten Pb-Bi alloy would contribute to radionuclide release from depressurized fuel elements to the “container cavity”. Because sealing of SRU-housing “container” is not ideal, release of radioactive fission products to the central hall and next to the atmosphere would be possible.

It was assumed that, thanks to deposition of aerosols containing volatile radionuclides, their relative leakage was about 100 times less, as compared to gaseous radionuclides. Diffusion release of long-lived gaseous and volatile radionuclides from fuel element core into cladding during reactor operation was taken equal to 3 % over the lifetime in compliance with post-reactor studying of such-type fuel elements. The proportion of depressurized fuel element claddings was taken equal to 1%. The calculation results of fission product radioactivity “before” and “after” the pulse are demonstrated in Fig.6.

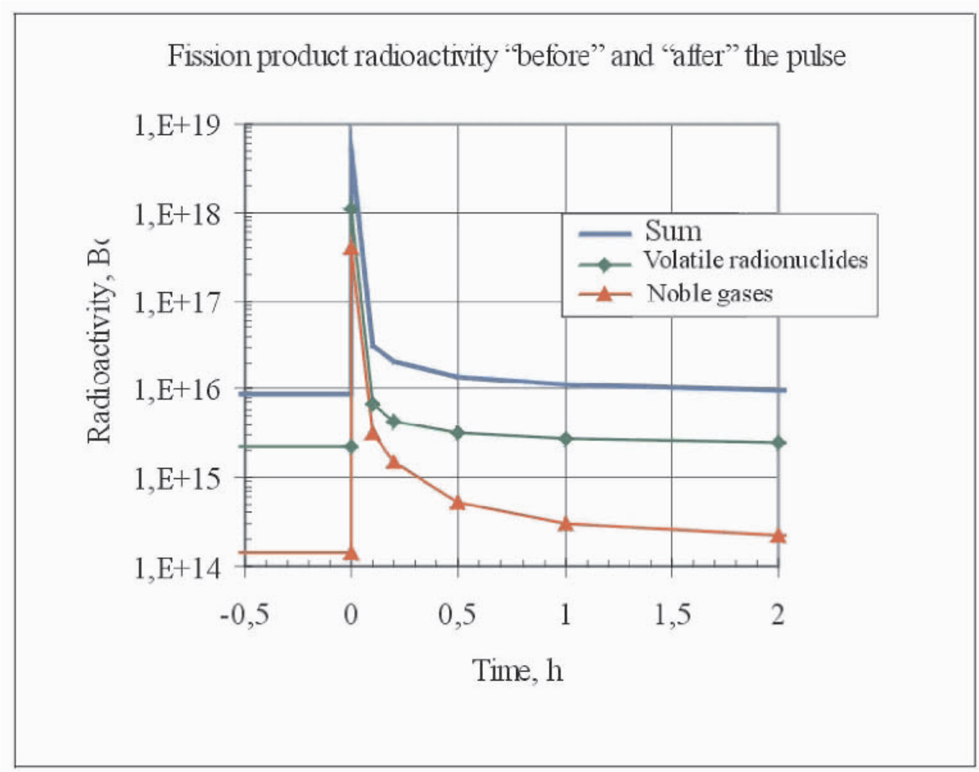

Figure 6. Fission product radioactivity

Radionuclide release to the Central Hall (CH) could lead to contamination of air and surfaces. If ^{137}Cs deposition were not taken into account, its volumetric activity in CH air would reach the maximum permissible value of $2{\cdot}10^3$ Bq/m^3 [6].

The behavior of release intensity and the activity accumulation in the atmosphere are demonstrated in Fig.7. The release intensity reaches its maximum of $8{\cdot}10^4$ Bq/s during the first day after the pulse. The major contribution is due to long-lived ^{85}Kr ($T_{1/2}$ = 10.7 year) generated over previous operation of the reactor. The release of volatile radionuclide (mainly long-lived ^{137}Cs) is considerably less.

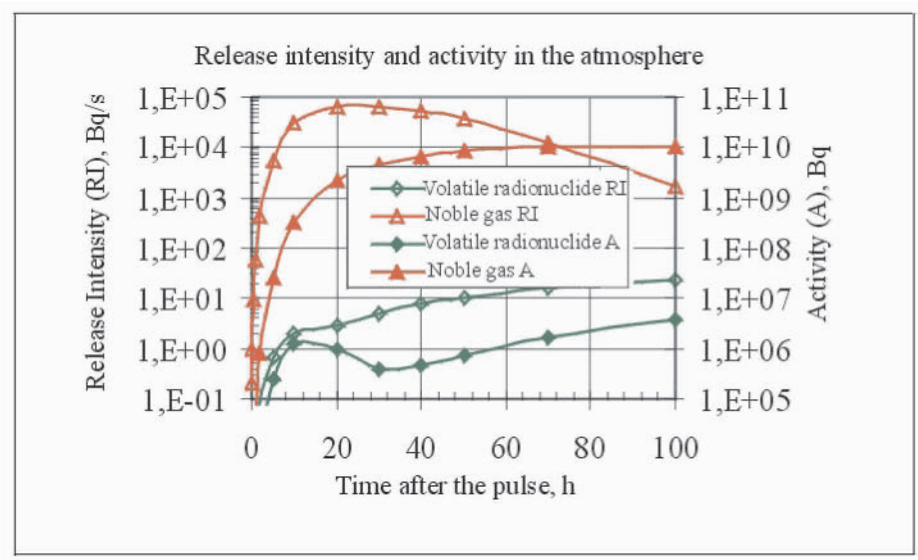

Figure 7. Release intensity and activity in the atmosphere

When calculating radionuclide spreading in the atmosphere, the worst dispersion conditions were taken [8]. The near-surface concentration of long-lived beta-emitter ^{85}Kr reached 13 Bq/m^3 at a maximum, the integral human exposure dose (skin) due to ^{85}Kr during radioactive cloud passage time being equal to 0.016 µSv (Fig.8), i.e. considerably below the natural radiation background. As the distance from the source term increased, both the near-surface concentration and the exposure dose rapidly decreased.

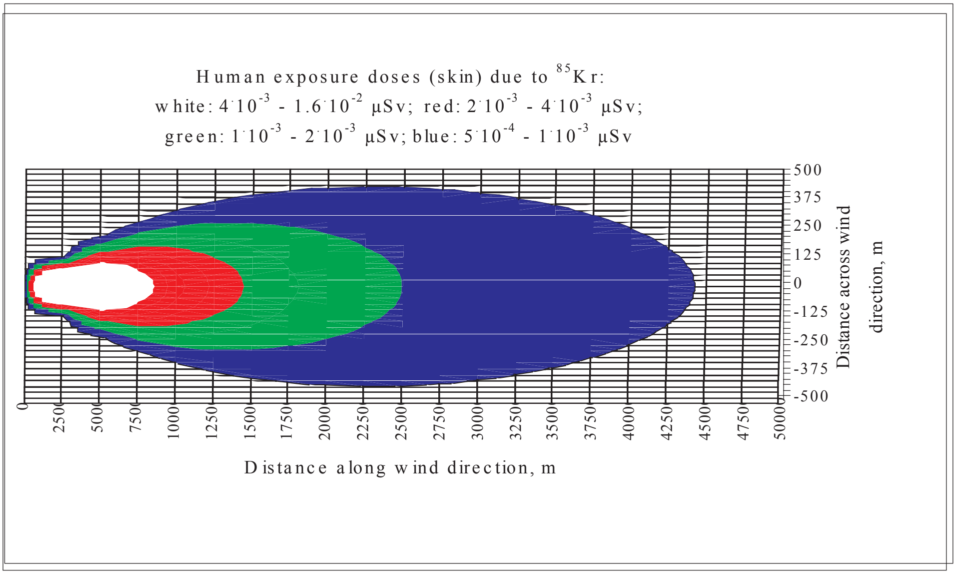

Figure 8. Exposure doses

The expected population exposure doses due to other raduonuclides were considerably less, as compared to ^{85}Kr. There are grounds to believe that beyond the enterprise site no air contamination would be recorded at all.

Thus in case of $2 \cdot 10^{19}$-fission pulse during storage of KM-1 stand's SRU, worsening of the radiation situation could be only possible in the central hall leading, however, to no occupational overexposure. This means that no health radiation hazard for population of either Sosnovy Bor-town, or more remote regions (e.g., Finland, Estonia and Latvia) would emerge in such a case.

6. Conclusions

1. Justification of nuclear and radiation safety of spent nuclear fuel for stands-prototypes 27/VT and KM-1 of steam producing installation of LMC NS has been performed under long-term "normal" and "emergency" conditions.
2. Accumulated activities of actinoids and fissions products in the cores as well as the mean activity in individual reactor fuel elements of stands 27/VT (the first and the second lifetimes) and KM-1 have been estimated as of the turn of 2003 and 2010.
3. It has been confirmed that the complex of technical and organizational measures applied under long-term SNF storage of 27/VT and KM-1 stands ensures their nuclear and radiation safety according to the regulations in force which comply with the requirements of public supervisory bodies on nuclear and radiation safety.

4. The performed calculations of water ingress into SRU core of KM-1 stand have demonstrated that in case of water ingress into SRU the core could attain supercritical state leading to temperature increase in the core, subsequent water evaporation and the core transition to subcritical condition. At the rates of water ingress into the core considered in the calculations the core would not attain criticality due to prompt neutrons. An increase in core temperature would not lead to fuel element depressurization either. Heat energy released in the core would not be of explosive nature being used for heating of SRU elements and the "container".

5. An estimate of the radioactive consequences of a pulse power increase in SRU of KM-1 stand has shown that worsening of the radiation situation would be only possible in the storage facility central hall resulting, however, in no occupational overexposure. There would be no health radiation hazard for population of either Sosnovy Bor-town, or more remote regions (e.g., Finland, Estonia and Latvia) in such a hypothetical situation.

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DEFUELING OF RETIRED NUCLEAR SUBMARINES

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The Russian Federation (RF) Decree #518 of May 28, 1998 "On Measures on Supporting Complex Decommissioning of Nuclear Submarines, Withdrawn from Service in the Russian Navy, and Ships of RF Ministry of Transport" entrusted RF Ministry for Atomic Energy (Minatom, presently Rosatom) with the task of drastic increase in the paces of Nuclear Submarine (NS) complex decommissioning, the heaviest burden falling to the main work executors – major shipyards supporting the whole cycle of works.

NS defueling is undoubtedly the most important phase of nuclear vessel complex decommissioning: the operations are performed at opened reactor and strong hull and thus are a source of nuclear and radiation hazard [1].

In most cases the shipyards concerned with complex decommissioning are situated within boundaries of settlements. This fact imposes special safety requirements to SNF unloading operations.

At present defueling of taken-out-of-service NSs is performed using:

– Floating Service Vessels (FSVs) - Fig. 1;
– on-shore defueling facilities - Fig. 2; and
– slip docks - Fig.3.

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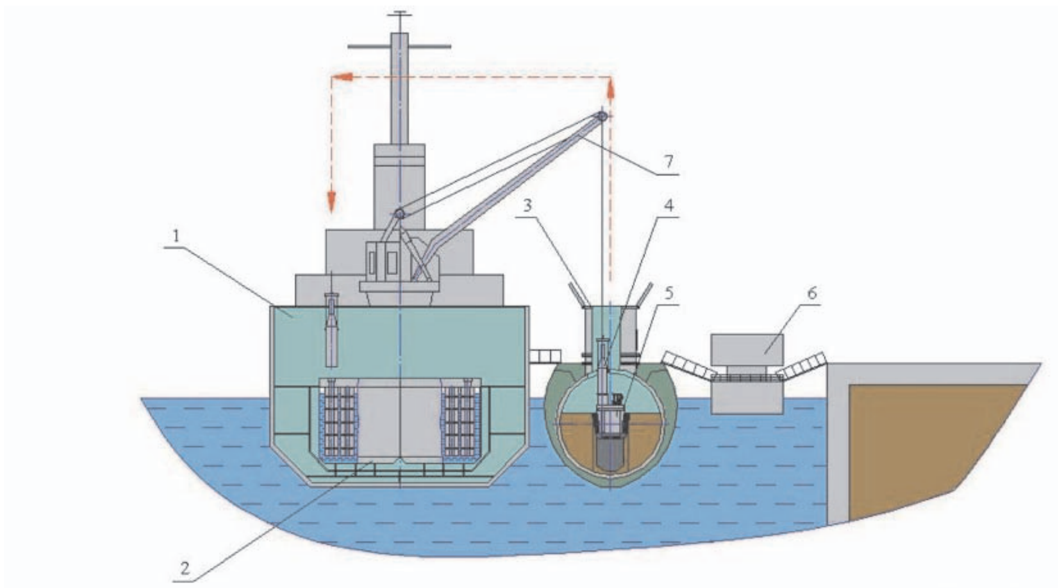
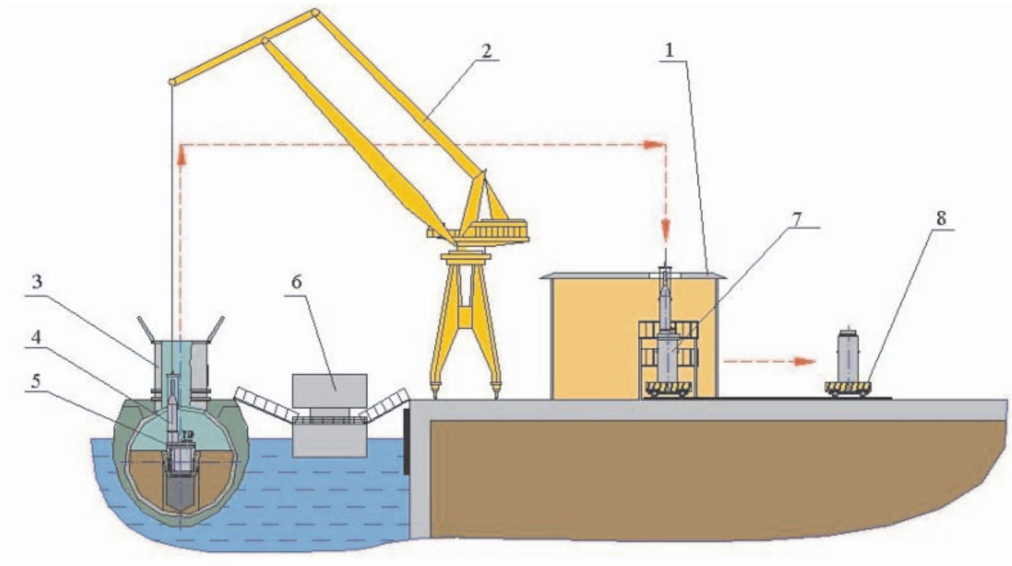


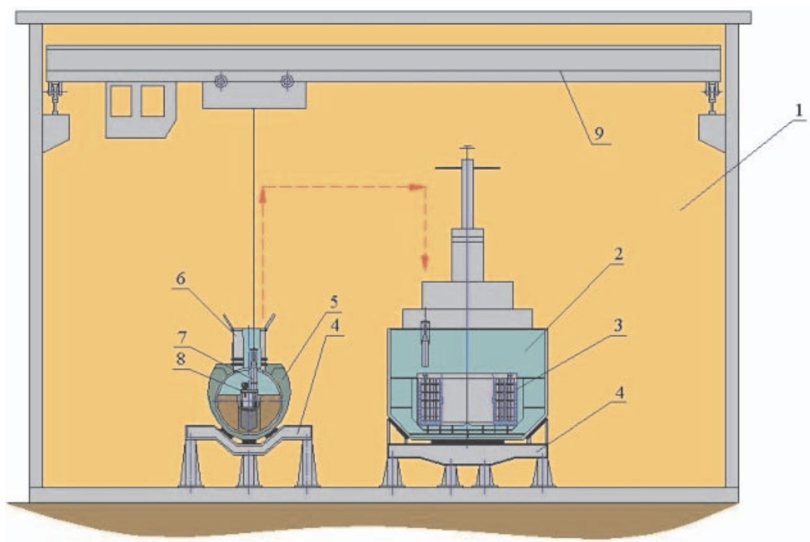
1. FSV
2. Storage facility
3. Closing hatch
4. Unloading container
5. Guiding mechanism
6. Dose-control post
7. FSV crane (16 t-f capacity)

Figure 1. Spent nuclear fuel unloading using FSV



1. On-shore defueling facilities
2. Climbing crane
3. Closing hatch
4. Unloading container
5. Guiding mechanism
6. Dose-control post
7. Transport container TK-18 (TUK 108/1)
8. Traverser

Figure 2. SNF unloading using on-shore defueling facilities



1. Spip dock
2. FSV
3. Storage facility
4. Slip
5. NS
6. Closing hatch
7. Unloading container
8. Guiding mechanism
9. Climbing crane

Figure 3. Defueling in slip dock using "*Imandra*" FSV

Every defueling operation is performed using special equipment, which ensures nuclear and radiation safety of personnel [2].

Safe defueling of NSs is ensured thanks to:

– worked through defueling procedures and their strict observance;
– use of verified, reliable and fully completed sets of reloading equipment; and
– conducting of operations by well-qualified trained personnel [4].

1. Process peculiarities of defueling

Implementation of the method of Spent Nuclear Fuel (SNF) unloading from unwatered NS reactors represents a crucial solution of the nuclear safety challenge because removal of moderator eliminates the risk for reactor core to attain critical condition under any design-basis/beyond-the design-basis operations with reactor control systems.

However the "reactor unwatering" method can be only recommended when the following main problem issues are resolved:

1) Ensuring radiation safety

When water is discharged from NS reactor, biological shielding against ionizing radiation diminishes. According to calculations, when SNF is unloaded from NS reactor unwatered after 3 hold up (waterborne storage) years following reactor shutdown, the available reloading equipment ensures adequate shielding against radiation complying with the Russian Safety Standards (NRB-99) in force. However, though the performed calculations were confirmed by subsequent practical activities, several operations were identified requiring additional shielding. One of them was reactor upper head mounting necessitating presence of personnel close to unwatered reactor. To diminish the risk of radiation exposure, special shielding plugs were inserted into the upper reactor plate (without subsequent dismantlement) and next were included into the defueling equipment sets.

2) Determining an optimal period of time preceding water-discharge before NS defueling and developing recommendations on SNF further management

Because reactor unwatering considerably worsens decay heat pick-up from Spent Fuel Assemblies (SFAs), it should be performed after some period of fuel hold up in non-unwatered reactor. Many calculations were performed taking into account both factual and possible-limiting state of cores depending on fuel burnup. Two options were examined:

– - reactor unwatering immediately before defueling onset; and
– - reactor unwatering followed by protracted storage of SFAs in unwatered reactor.

The option of SFA reloading from reactors directly into transportation casks (TK-18, TUK-108/1) was considered using a similar approach but taking into account the specificity of heat peak up conditions.

As the result of calculations, specific recommendations on optimal duration of fuel hold up before NS reactor unwatering were elaborated and then summarized in working instructions.

3) Estimating conditions of SFA-unloading from unwatered reactors

Unfortunately, reactor construction gives no way of performing its absolute unwatering resulting in development of corrosion. It is obvious that, when unwatering immediately precedes SFA unloading, corrosion processes do not have effect on unloading operations.

To estimate the effects of long-term (up to 10-years) SNF hold up in unwatered reactor on the conditions of subsequent defueling, special experiments using pressure chambers (Fig.4), prototypes and full-scale test benches (Figs. 5, 6 and 7) were performed at Machine Building Design Bureau (OKBM) allowing approaching the test conditions to real situations. It was established that the SFA shearing force increased during tests by 16–20% as compared to that measured before tests. However such increase was recorded at the blasting instant only. Next, while unloading, that force became again virtually equal to the before-test one.

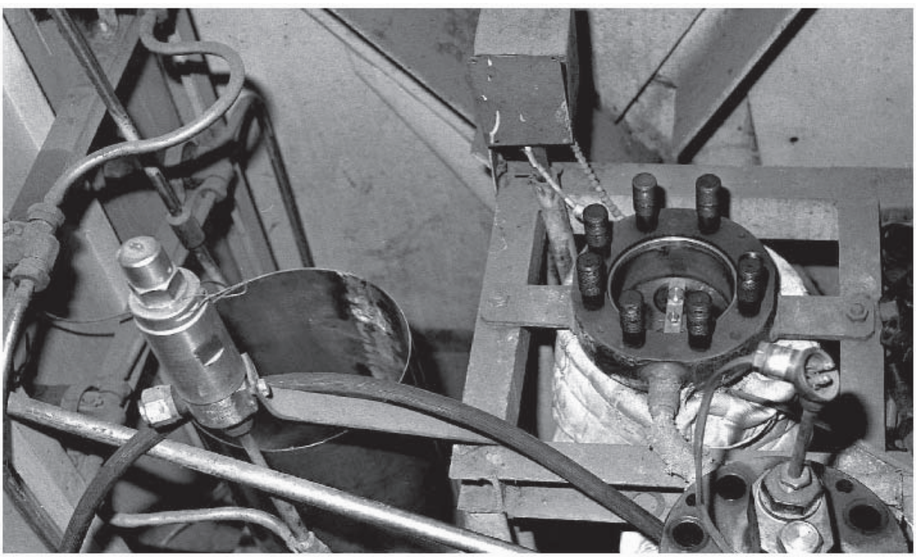


Figure 4. Tank with opened pressure chamber

Figure 5. Full-scale test bench to check guiding mechanisms and containers

Figure 6. Test bench for hydraulic tests

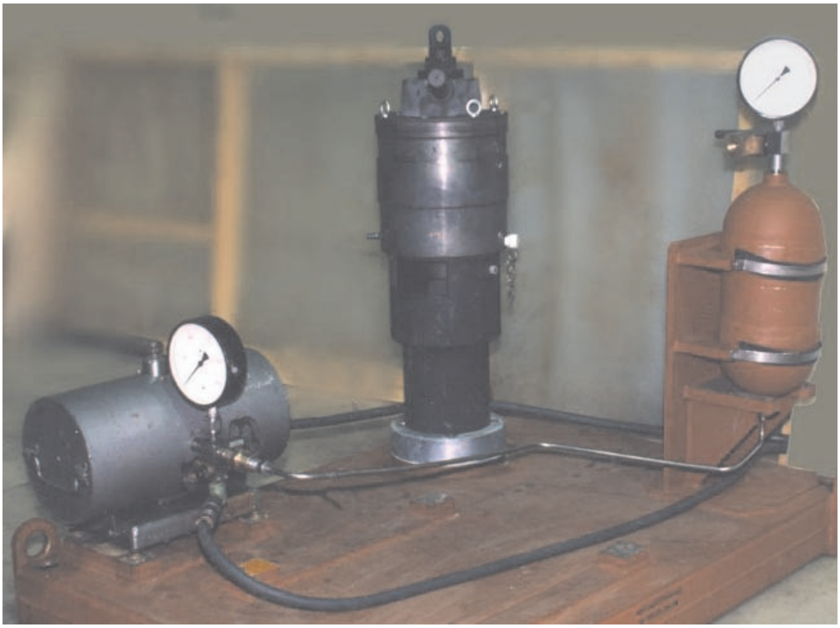


Figure 7. Test bench for hydraulic unlocks

Shearing force increase at the initial instant was mainly due to in-gap iron oxide (Fe_3O_4) depositions on surfaces of the tested SFAs after preliminary hold up in water. However no further densification of iron oxide depositions was revealed. Thus the performed tests confirmed the possibility of SFA unloading after long-term hold up inside unwatered reactors.

Based on the performed investigations, a specific NS defueling process technology was developed [3].

2. Reloading equipment

There were only few operable sets of reloading equipment (type: 300PB/300PBM) at the Russian Navy by the instant of large-scale deployment of NS complex-decommissioning activities. Lifetime of some equipment sets was over; almost all the remaining sets expired their up-to-mid-life-repair service life. OKBM jointly with the contracting parties (Nizhny Novgorod Machine-shop, different shipyards) repaired all sets of reloading equipment.

The reloading equipment under consideration consists of:

– - simple devices having no moving elements to take up dynamic loads (shielding plates, guide structures, etc.);

– - standardized tools; and

– - high-end bulky (hydraulic, mechanical) devices providing for guiding on an assigned point of the core, SFA extracting and reactor upper head blasting (Figs. 8, 9, 10, 11, 12 and 13).

Figure 8. Hydraulic unit

Figure 9. Device to turn/unscrew the screws on main connector

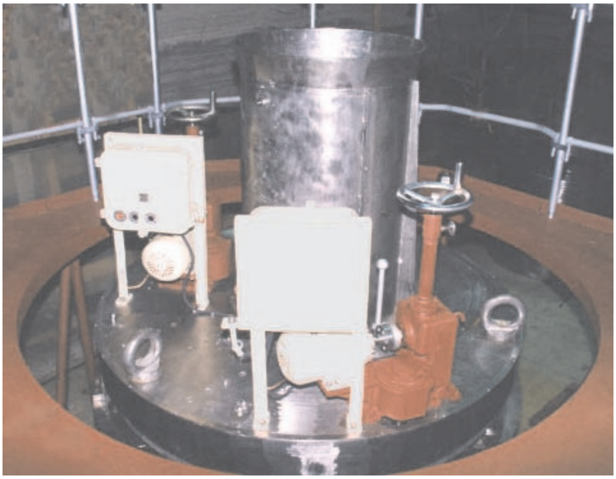


Figure 10. Guiding mechanism

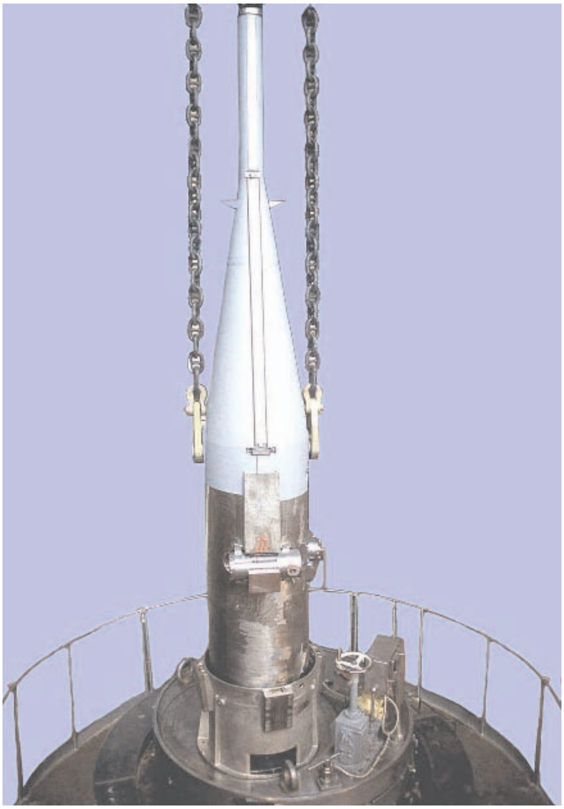


Figure 11. Guiding mechanism with container

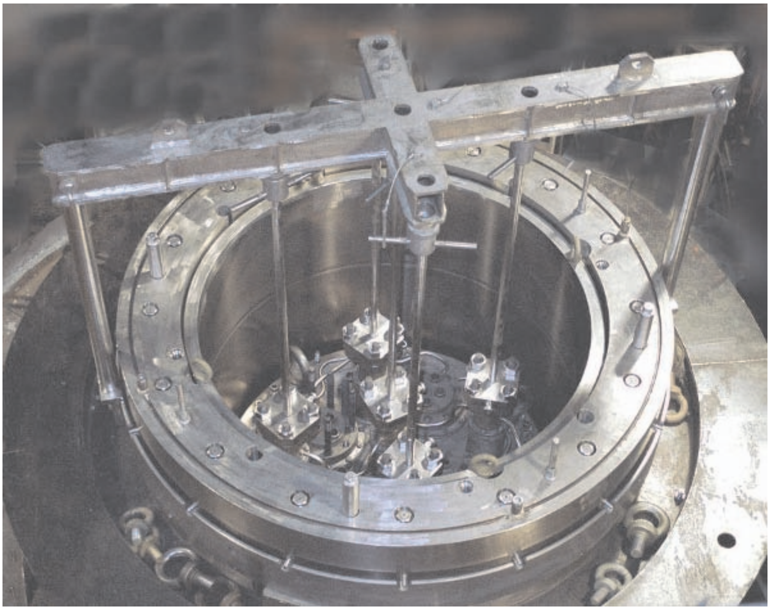


Figure 12. Device for lid shearing and blasting using hydro-jacks

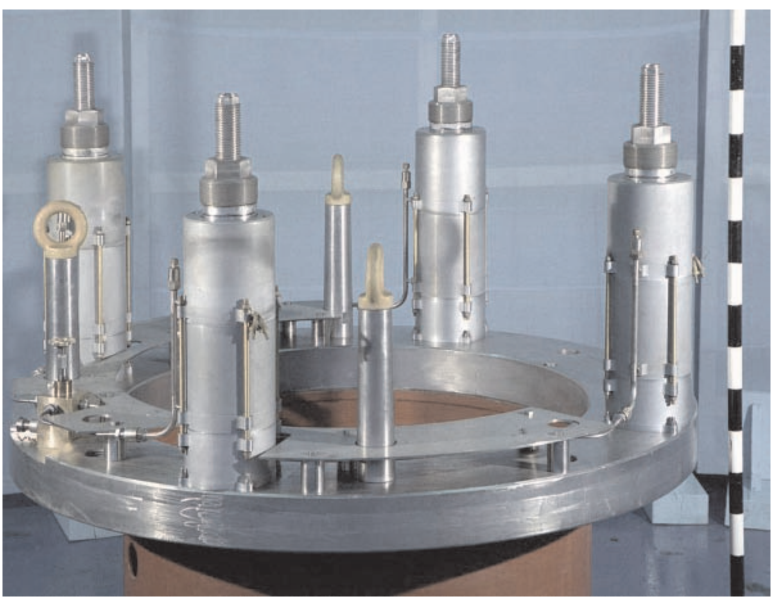


Figure 13. Device to blast pressure flange

All components of the equipment should be operable at any time of the year including severe winter season with considerable temperature variations. The necessity of decontamination represents an additional work-complicating factor. Thus reliability and stable operation of reloading equipment elements is an important component of work safety throughout the nuclear vessel dismantlement process.

Simultaneously with repair of reloading equipment elements, test benches (Figs. 6, 7 and 8) were developed and manufactured to conduct performance verification of equipment after repair and before every unloading. Such an approach to work management allowed attaining stable positive results of defueling operations [2].

However the available then at the Russian Navy reloading equipment sets were unable to cope with the required increase in NS defueling paces. To fit out the newly built on-shore defueling facilities, multi-purpose NS defueling equipment (type 300PBU) was developed. It included the best elements of 300PB/300PBM sets along with some new elements allowing mechanizing individual laborious operations, making easier the work of maintenance personnel and reducing the time of personnel staying under radiation-hazardous conditions. Those new elements comprised the devices for: cutting off welds, blasting pressure flanges and performing simultaneous installation of hydraulic unlocks and pumping units.

In particular, 300PBU-type defueling equipment was used at “Nerpa” shipyard when defueling “*Kursk*” NS reactors. The operations supported by FSV “*Imandra*” were performed at the shipyard’s slip dock (Fig.3).

To maintain all sets of defueling equipment in operable condition and ensure safety of operations, special "Program of Repair-Renewal Works to Maintain 300PB and 300PBM Reloading Equipment Sets and 300PBU Unloading Equipment in Operable Condition Until 2010" was developed providing for gradual replacement of individual components in the operating sets by elements of improved specifications (300PBU components) [4].

3. Training of maintenance personnel

Execution of works by highly qualified personnel is one of most important components of safe NS defueling.

There is presently a two-level training system for specialists concerned with NS defueling. A licensed "Center for Training, Retraining and Further Professional Development of Specialists Involved into Defueling of Naval Nuclear Reactors" (below the Center) has been established at OKBM authorized in training specialists on the subjects in question. Special training programs have been developed for the managing staff providing for profound studies of reloading equipment, process technology of works, nuclear and radiation safety issues and reactor designs. Retraining of specialists is performed at the Center once every three years. Competent specialists of Nizhny Novgorod State Engineering University and leading specialists of OKBM give lectures and run trainings. Specialists directly involved into SNF unloading are trained at shipyards including exercises at test benches.

The Center is fit out with up-to-date equipment, personal computers, training manuals, posters, etc. (Fig.14). It operates under approved programs, which include a variety of special video films. After training courses at the Center specialists are certified by representatives of OKBM, Department of State Supervision over Nuclear and Radiation Safety of RF Ministry of Defense and Rosatom (Fig.15).

Figure 14. Training of specialists

Figure 15. Certification of specialists

Conclusion

At present Federal State Unitary Enterprise Machine Building Design Bureau (OKBM) performs engineering management of the entire defueling process, controls technical condition of reloading equipment, organizes its renewal and repair as well as training and certification of personnel concerned with SNF handling. Special technologies developed at OKBM cover all SNF-unloading-related works.

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METHODS FOR ANALYTICAL AND EXPERIMENTAL JUSTIFICATION OF NUCLEAR AND RADIATION SAFETY DURING UNLOADING AND STORAGE OF SPENT REMOVABLE PARTS FROM NUCLEAR SUBMARINE LIQUID METAL REACTORS

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The analytical and experimental methods applicable for justification of neutronic characteristics of spent removable parts (SRP) of nuclear submarine liquid metal reactors (NS LMR) are briefly reviewed.

The following software code systems can be used to justify nuclear and radiation safety, to determine the SNF nuclide composition and energy release:

– MMKFK-2, based on Monte-Carlo method;
– «SCALE»;
– WIMS|ABBN;
– Updated NucMa version.

The methods of determining SRP subcriticality have been developed to experimentally justify nuclear and radiation safety. This paper considers the pulse, stationary and correlation methods of determining NS LMR SRP subcriticality.

Introduction

Spent removable parts (SRP) unloading from reactor vessels with liquid metal coolant (LMC) was carried out in the period from 1966 to 1990. The unloaded SRP's were assumed to be held in the storage facility for a short period of time to be followed by their shipment for reprocessing. Currently each unloaded SRP is being stored in the "clean" (non-radioactive), "frozen" lead-bismuth alloy in special steel containers located inside concrete pits. The storage time has not been specified yet.

Radiological and environmental safety of SPR's being stored is based on the defense-in-depth principles on the way of possible radionuclides penetration into the

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 209–217.

environment. The fuel element matrix, fuel element clads, "frozen" coolant, the tight steel storage tank serve as barriers for fission products.

Nuclear safety of the SRP's being stored is ensured with profound initial subcriticality and control and protection system rods located either in the "frozen" alloy (shim rods, automatic control rods), or in the fixed position (scram rods) by the efforts of their actuating springs. The maximum K_{eff} value of NS reactor was equal to ~0.86 before SRP's unloading. The indicated breeding ratio (K_{eff}) meets the nuclear safety requirements with no possibility to change it during SRP's storage and management. After the SRP's location in the storage facility the K_{eff} has never been measured.

The analysis has shown that the SRP's with beryllium reflector flooded with lead-bismuth alloy are the most dangerous from the point of view of nuclear safety. The main reasons that can cause K_{eff} variation due to a positive reactivity are air voids (porosity) being present in the SRP's cores and the possibility for these voids to be filled with water, as water reactivity worth in the SRP's core is ~0.7 β_{eff} / L. Thus it means that about 30 Liters of water have to be accumulated for K_{eff} to be ~ 1 in the core.

In this context it follows that NS LMR SRP's nuclear and radiation safety has to be monitored and controlled during their unloading and storage. By now in the leading nuclear centers of the Russian Federation the methods of analytical and experimental justification of nuclear and radiation safety have been developed for SNF management. In the paper consideration is given to the methods of analytical and experimental justification of nuclear and radiation safety that can be used in the course of NS LMR SRP's unloading and storage.

1. Methods of analytical justification of nuclear and radiation safety

The main characteristics of spent nuclear fuel (SNF), that determine its nuclear and radiation safety, are:

– Radionuclide composition;
– Residual energy release;
– Burnup;
– Subcriticality (criticality).

For the indicated SRP's characteristics to be calculated a number of software code systems can be used. They are briefly considered below, whereas their detailed description is given in [1-4].

The NucMA code developed in the RSC Kurchatov Institute, is meant to calculate SNF residual energy release both for separate SFA's and for the whole inventory of accumulated SFA's or its any sampling. The code can be used to calculate SRP's burnup, radionuclide composition and residual energy release. Radionuclide composition is determined as a function of burnup (or power generation) at the averaged reactor parameters, i.e. power, coolant density and temperature. Besides, the code uses such parameters as input fuel enrichment, core irradiation time, post-irradiation hold-up time.

The WIMS/ABBN code system developed in the SSC RF IPPE makes it possible to calculate actinoids content (activity) as a function of burnup. The code algorithm includes computation of neutron fields in the cells and clusters of several different cells of heterogeneous thermal neutron reactor grid and preparation of homogenized micro- and macro- small-group constants (nuclear data) in view of burnup and accumulation of actinoids and fission products. Depending on the user's task the following data can be calculated: small-group homogenized macro- and blocked micro- cross- sections of isotopes, neutron absorption-generation balance with its break-down in terms of nuclides; non-blocked cross-section of secondary actinides and fission products, delayed neutron lifetime, fission neutron worth, isotope concentration in terms of burnup steps, etc.

The code has five external data libraries. The operation of all the code system units is well adjusted, the code system has been verified.

The SCALE code system developed in the SSC RF IPPE is the analogue of the US SCALE system (Standardized Computer Analysis for Licensing Evaluations) [4]. The code system allows the computation of fission product content (activity) and activation in SNF as well as energy release. The system integrates the certified software modules and data banks.

The specific features of the SKALE code system are as follows:

a) the use of Monte-Carlo method that allows the geometry of multiplication system under calculation to be described in details and procedure errors of calculated results to be practically ruled out;

b) the use of maximum precise, complete and comprehensively verified system of neutron data;

c) validation of the code system being used (nuclear data system and computation code) by comparing the calculated criticality results with the data represented by a set of critical experiments. In particular, the validation result consists in evaluation of calculated results uncertainty that is taken into account when determining the optimal subcriticality margin.

The MMKFK-2 code system can be used for nuclear safety analysis, namely, for calculation of NS LMR SRP's K_{eff}, L, α and β_{eff} parameters (where K_{eff} is an effective breeding factor; L is a prompt neutron lifetime in the multiplication system under calculation; β_{eff} is an effective fraction of delayed neutrons; α is a damping coefficient).

For the self-consistent calculation of K_{eff}, β_{eff}, L and α by the Monte-Carlo method in the MMKFK-2 code package there is a code called MCDENSP [5]. By means of the perturbation theory this code allows the calculation of prompt neutron lifetime L as a functional of quasi-steady-state neutron flux and conventionally critical neutron worth. The flux is determined from the quasi-steady-state transport equation that describes the asymptotic fall-off of prompt neutrons after the pulse impact on the given non-critical reactor. The computation technique and software are justified theoretically [6] and experimentally. Besides, the MCDENSP code was validated for the given problem by direct modeling of the transient process after a pulse in one of the cells, with the QRT code [9] of the MMKFK-2 code package. The α asymptotic value in the transient process agreed with the α value obtained in the result of modeling quasi-steady-state flow with the MCDENSP code.

The L calculation accuracy with the MCDENSP code slightly depends on the K_{eff} calculation accuracy due to the use of perturbation theory formulae and is primarily determined by the accuracy of task and medium diffusion (scattering and absorbing) property modeling.

Neutron transfer in the epithermal energy region is modeled in the 26-group approximation on the basis of BNAB-78,85 constants. The BNAB-90 data are used for delayed neutrons [10]. Neutron thermalization with the energy below 1 eV is modeled in the 40-group approximation on the basis of the MOFITTG physical module and TEPKON-90 nuclear data library from the MCU-2.0 code package.

In order to ensure nuclear and radiation safety control during NS LMR SRP's unloading and storage it is reasonable to develop an integrated code package that is characterized by the following:

- it integrates and combines possibilities to make calculation with each of above-mentioned codes, i.e. to calculate burnup, isotopic composition and energy release;
- it takes into account the experimental data on NS LMR SRP's available;
- it has the interactive mode of work with an operator.

2. Experimental methods proposed for determining SN LMR SRP characteristics

The experimental measurements of SNF burnup fraction is carried out by means of recording fission product gamma-radiation [11, 12] and neutron radiation of actinides accumulated in the fuel [11, 13]. If SRP's are stored in the "frozen" heavy metal, these measurements are impossible. For this reason the experimental methods are mainly used for measuring subcriticality of SRP's being stored.

The well-known methods of measuring subcriticality of any multiplication systems can be grouped into dynamic (active) and statistical (passive) ones. Dynamic methods incorporate different devices that vary reactivity of the system under study or its neutron flux density according to a certain law. With the application of statistical methods these devices are usually not used. And if they are used, they do not result in determined variation of reactivity and neutron flux density.

The most promising among dynamic methods of reactivity measurements is the pulse method. The method is based on measuring the system response after neutrons from the pulse source have been introduced into it. Among all the modifications of this method related to the processing technique of the system response to a narrow pulse of neutron generator the preference should be given to the modified Simmons-King method, because it contains the least number of parameters whose errors can cause the uncertainties of the results. According to the available data [14] the Simmons-King method applicability limits vary for different system. This fact has to be taken into account for each specific case. The advantage of this method consists in the fact that the use of a neutron pulse with the energy of 14 MeV facilitates the set-up of fundamental spatial harmonic within a big volume of the system under study and thus results in a less spatial dependence of prompt neutron damping decrement.

If SRP $K_{eff} \geq 0.9$, the Sjostrand approximation can be used, with which reactivity is determined in terms of prompt-to-delay neutron area ratio.

In order to determine profound subcriticality ($K_{eff} \leq 0.9$) on the basis of the measured value of asymptotic prompt α-flux decay constant, the following formula is used [15]:

$$-\frac{R_0}{\beta_{eff}} = \frac{\alpha \wedge}{\beta_{eff}} - 1 \rightarrow K_{eff} = \frac{R_0 - \beta_{eff}}{\wedge} \qquad (2.1)$$

where $R_o = (K_{eff} - 1)/K_{eff}$ is the system radioactivity;

β_{eff} is effective fraction of delayed neutrons;

$\wedge$ is prompt neutron generation time.

The $\wedge$ value is inversely proportional to the fission neutron generation rate, so it significantly depends on the sought value of K_{eff}, which we would like to determine on the basis of experimental value α and calculated values β_{eff} and $\wedge$. Thus, another formula is used:

$$\alpha = \frac{1 - K_{eff}(1 - \beta_{eff})}{L} \rightarrow K_{eff} = \frac{1 - L\alpha}{1 - \beta_{eff}} \qquad (2.2)$$

where L is prompt neutron lifetime in the system. It is determined by calculation and is related to generation time $\wedge$ through the function $L = \wedge K_{eff}$. L is significantly less dependent of fuel assembly multiplication uncertainty than $\wedge$. However, similar to $\wedge$, the L value significantly depends on accuracy of the system modeling. So for its calculation it is reasonable to use the Monte-Carlo method.

In contrast to the pulse method of determining subcriticality, which because of its labor-consumption and limited neutron tube lifetime is only used at considerable medium variations, the stationary method is used for constant and non-stop control of subcriticality.

The stationary method of subcriticality control is based on a well-known property of multiplication system to increase neutron flux caused by primary neutron sources

[16]. In SRP's the primary neutron sources are induced by spontaneous fission of the built-up transuranium isotopes (^{242}Cm, ^{244}Cm, ^{240}Pu), as well as beryllium (α,n) reaction.

In order to control subcriticality of SRP's storage facilities by the stationary method, it is necessary to know the induced fission neutron generation rate per gram of fuel (Q_{ind}) and the rate of neutron generation by above-mentioned sources per gram of fuel (Q_{sp}) in the maximum neutron flux points [17]. If Q_{sp} and Q_{ind} are known, the system multiplication can be written in the following way:

$$M = \frac{Q_{ind} + Q_{sp}}{Q_{sp}} \qquad (2.3)$$

where Q_{ind} is the number of induced fissions (fissions induced by source neutrons) in the system per mass unit, and Q_{sp} is the primary source neutron power.

As $M = \frac{1}{\Delta K}$, and $\Delta K = 1 - K_{eff}$, then $\frac{1}{1-K_{eff}} = \frac{Q_{ind} + Q_{sp}}{Q_{sp}}$,

$1 - K_{eff} = \frac{Q_{sp}}{Q_{ind} + Q_{sp}}$, thus

$$K_{eff} = 1 - \frac{Q_{sp}}{Q_{ind} + Q_{sp}} = \frac{Q_{ind}}{Q_{ind} + Q_{sp}} \qquad (2.4)$$

The maximum K_{eff} value is estimated from the equation:

$$K_{eff} \le \frac{\frac{Q_{ind}}{Q_{sp}}}{1 + \frac{Q_{ind}}{Q_{sp}}} \qquad (2.5)$$

The primary fission source neutron generation rate per gram of fuel (Q_{sp}) was calculated with the MMKFK-2 or STEPAN codes with the use of data on FA power output.

The main equation for calculation of K_{eff} with the fission chamber count rate is determined by the formula:

$$K_{eff} \le C \frac{I_{max}}{1 + I_{max}} \qquad (2.6)$$

The relative error of subcriticality determination by the stationary method, $\Delta K_{eff}/K_{eff}$ is decreasing with the K_{eff} growth. So that the value of $\Delta K_{eff}/K_{eff}$ could be equal to ~1%, the required accuracy in determination of spontaneous and induced fission neutron generation rate has to be no worse than 7 % at K_{eff} ~ 0.9 and 1.4 % at K_{eff} ~0.5.

From the brief description of proposed pulse and stationary methods for SRP subcriticality calculation it is quite obvious that they require the introduction of calculation constants, i.e. the specified values of delayed neutron lifetime and effective fraction for the pulse method and the neutron source for the stationary one. Thus, it follows that for these experimental methods to be realized the SRP's parameters that characterize multiplication properties have to be calculated with a good accuracy.

In order to avoid the necessity to obtain SRP parameters calculated with a high accuracy, a correlation method was proposed for determining the SRP's conditions after their long-term storage. The method is based on correlations between calculated and experimental values of relative neutron flux density that will make it possible to verify the assumptions about the voids being filled with water. Actually these assumptions determine the level of risk related to nuclear, radiation and ecological safety during the further SRP storage.

In the correlation method the neutron flux density of the specified region is determined from the ratio of pulse measurement channel count rate to detector sensitivity $\varepsilon(E)$. If we assume that the neutron spectrum in the area of neutron detector location does not depend on the type of SRP being stored in the pit, then for all the measurements $\varepsilon(E)$ is a constant value. In the next calculations it can be assumed as 1 and thus, the count rate being measured can be assumed equal to the value of proportional neutron flux density.

The operation principle of pulse channel for measuring the count rate is based on the neutron registration by the detector of ionization fission chamber or neutron counter, pulse transmission from the detector via the communication lines to the preamplifier inlet and their following amplification, formation and processing by means of the auxiliary electronic equipment.

The pulse channel allows the count rate to be measured in the specified area of SRP storage with the statistical accuracy up to ~1%. In order to take into account the electronic equipment drift and variations in the detector operation conditions it is necessary to have the external device for calibration and control of pulse channel stability.

3. Analytical and experimental methods being used for determination of NS LMR SRP characteristics in the settlement called Gremikha

The SSC RF IPPE in cooperation with "Sev RAO" is conducting the work on implementation of analytical and experimental methods for determination of NS LMR SRP characteristics in Gremikha. The correlation method is assumed to be used. Within the framework of this cooperation the experimental equipment was manufactured and supplied to Gremikha. This equipment is planned to be used for control of SRP conditions during their unloading from the NS reactor vessel and storage. The vertical section of SRP storage facility is shown in Fig. 1.

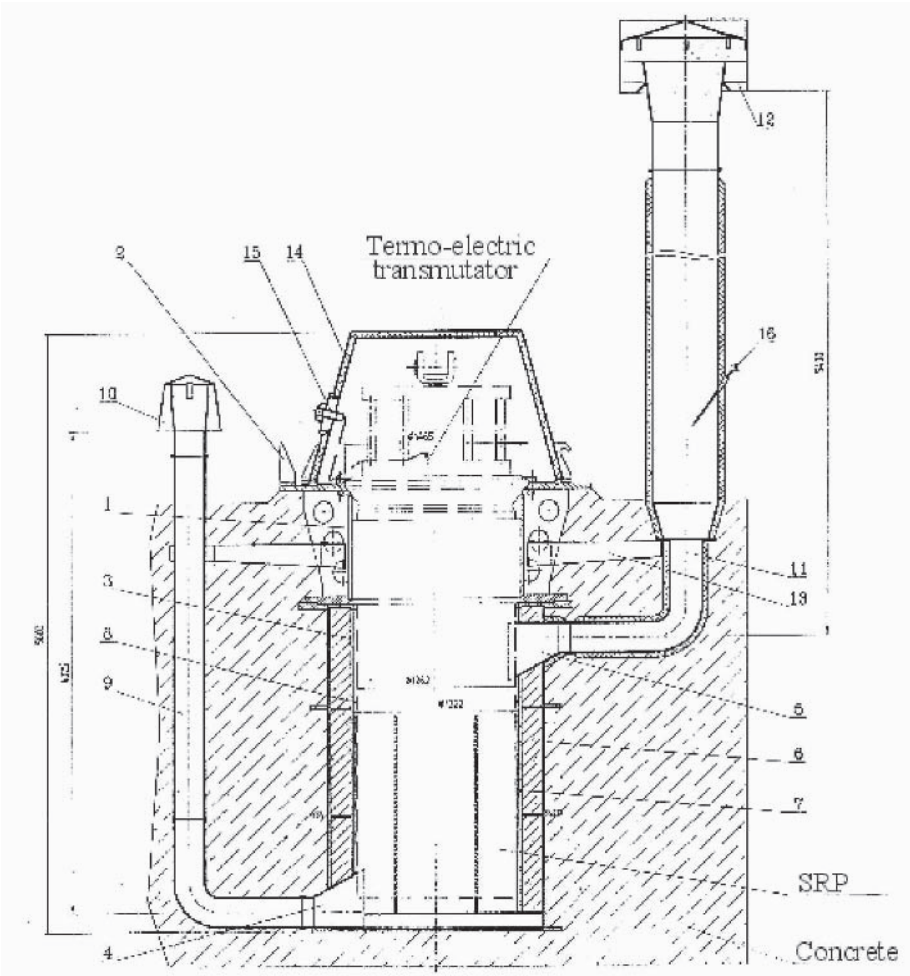

Figure 1. Vertical section of SRP storage facility

1. Top vessel	2. Shoe
3. Bottom vessel	4. Nozzle
5. Nozzle	6. Physical protection
7. Rib	8. Ring
9. Air flue	10. Deflector
11. Air flue	12. Deflector
13. Console	14. Dome
15. Opening box	16. Resistance thermal element

The experimental equipment serves to measure a relative neutron flux density and can be used for recording and time analysis of neutron distribution when determining SRP subcriticality by pulse and stationary methods.

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JUSTIFICATION OF A POSSIBILITY OF COMPACTED STORAGE OF NAVAL SPENT NUCLEAR FUEL

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A permanently increasing gap between the growth in Spent Nuclear Fuel (SNF) amount, on the one hand, and the deficiency in storage facilities taken together with insufficient capacities of SNF-processing enterprises, on the other hand, are characteristic for the actual phase of nuclear power industry development throughout the World including Russia.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 219–222.

In case of Nuclear Power Plants (NPPs), compacted storage of Spent Fuel Assemblies (SFAs) in cooling ponds appears to be actually an optimal solution, provided that nuclear safety is ensured. However such a solution has, undoubtedly, some safety and technological limitations.

1. Prospective technologies

To date, among a variety of SNF storage options, the following ones are used most often:

– SFA storage at dry storage facilities;
– SNF storage "on-site" within dual-purpose containers followed by shipment to SNF-processing plant after several years; and
– SNF storage in special containers (temporary storage – tens of years; long-term storage – hundreds of years).

There are considerable qualitative distinctions between naval SNF and that of NPPs, thus it is very important that the specificity of naval SNF is taken into account throughout the management process. The most important distinctions and peculiarities of naval SNF management are summarized below:

– residual enrichment of naval SNF exceeds that of NPPs by about an order of magnitude;
– cores of naval nuclear reactors have considerably smaller dimensions as compared to cores of similar-type reactors of NPPs;
– decay heat of naval cores is by 1-2 orders of magnitude less as compared to NPP cores of the same hold up: due to protracted (up to 10 and more years) forced storage of naval cores in reactors their decay heat does not exceed few kW (or even hundreds of W) being comparable to decay heat of one SFA of NPP after the same hold-up;
– as distinct from SFAs of nuclear power plants, in most cases linear dimensions of individual constructional elements of naval SFAs are comparable allowing their more compact loading into dual-purpose containers;
– the integral quantity of metal-concrete transportation casks (type TUK-108/1) manufactured so far in Russia for naval SNF are capable for housing only 20% of SNF of the taken-out-of-service NSs;
– TUK-108/1 casks with naval SNF are stored at special pads of shipyards concerned with NS dismantling operations; thus the quantity and capacities of such pads are limited.

Multi-factor approach, i.e. integrated consideration of physical, economical and engineering factors, provides for new opportunities when dealing with the problem of naval SNF storage in single-purpose and dual-purpose containers.

So far Russian specialists have developed a number of prospective technologies for compact storage of naval SNF. Some of them have been already patented in the Russian Federation (RF), e.g.:

– RF patents on SFA storage in reactor and/or storage tanks of floating service vessels, cut out power reactor installation placed into underground repository and cut out NS compartments,
– RF patents concerning: -SFA storage in reactor vessels within shielding containers using the “caterpillar” principle and -compact storage of containers with SFAs in special rooms, underground galleries and separated non-reactor NS compartments,
– RF patents on compact storage of SFAs in reactor vessels as core set components.

The above patents integrate technologies allowing not only enhancing nuclear and radiation safety of SFA storage and improving economic indicators, but also ensuring additional physical protection (including hypothetical terrorist attacks).

2. Protection of naval nuclear fuel against terrorist attacks

In the author’s opinion, the challenge of protecting naval SNF, entire cores and individual SFAs against possible terrorist attacks could become especially important in the near future and would concern not only Russia and its the adjacent countries but also the whole “nuclear-industry” world.

The problems of nuclear and radiological terrorism merit a special consideration.

Modeling of scenarios of potential terrorist attacks, forecast of their radiation consequences and protection of nuclear- and radiation-hazardous facilities against such threats is a many-sided challenge varying from the risk of local incidents to that of provoking a new world war with mass application of strategic nuclear armaments.

At present, when determining goals, objectives, structure and procedures of functioning of the entire counterterrorism system, an insight into potential nuclear and radiological threats should be at the heart of the problem.

Among typical potential terrorist attacks menacing not only Russia but also the entire world, the following two types of nuclear threats should be considered apart:

2.1. POTENTIAL TERRORIST ATTACK AGAINST OPERATING NUCLEAR FACILITIES

To date the operating nuclear facilities are adequately protected against different-type external impacts. However it is obvious that, to withstand both external and internal threats, their safety and security levels could be increased infinitely.

2.2. POTENTIAL TERRORIST ATTACK AGAINST NUCLEAR FACILITY STORING SNF

The situation related to SNF management is quite different. Taking into account SNF management technologies (see, e.g., [1]) and after performing correct estimates of potential risk and damage, one would be able to decrease the risk of SNF (especially of damaged one) storage/disposal by several orders of magnitude with no increase in the expenses for prevention of emergency situations including terrorist attacks.

At present qualitative rather then quantitative aspects of this problem are especially acute at the Russian Navy due to large-scale decommissioning of NSs, nuclear-powered surface ships, nuclear icebreakers, floating service vessels and Coastal Maintenance Bases (CMBs). In recent years some problems of managing highly enriched SNF stored at nuclear vessels have been successfully resolved, and today top-priority attention should be focused on CMBs with their storage facilities containing tens of thousands of spent and damaged fuel assemblies.

A model of possible terrorist attack (single shot at metal-concrete container from grenade cup discharge at on-shore site of "Atomflot", Murmansk Shipping Company) was presented in the 2002 NATO-Russia Advanced Research Workshop "Remaining Issues in the Decommissioning of Nuclear Powered Vessels" [2]. According to the model, even such – not very powerful – hypothetical terrorist impact could lead to serious implications for personnel and environment.

At a later time some other models of catastrophic events provoked by terrorist attacks were developed: for example, a similar-to-"September 11" scenario of FSV striking (including striking of "*Lepse*" FSV with SNF storage facility mostly filed with damaged SFAs) leading to rather serious consequences [3, 4].

As shown in References [1-4], some of the above-mentioned alternatives of compacted storage of naval SNF allow increasing safety by 2-4 and more orders of magnitude and protecting adequately defect fuel and SNF against terrorist attacks.

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SPENT NUCLEAR FUEL CARRIAGE BY SEA

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Complex decommissioning of Nuclear Submarines (NSs) and Nuclear Powered Surface Ships (NPSSs) is one of major challenges the Russian Shipbuilding Industry has presently to deal with. Despite important recent achievements thanks, firstly, to the efforts of the teams of "Zvezdochka" Shipyard, Far East Plant "Zvezda" and 'Onega" Research and Design Engineering Bureau, many serious problems still persist. Among them, safe management of Spent Nuclear Fuel (SNF) unloaded from retied nuclear vessels represents one of the most intricate problems.

The problem is further aggravated by the following circumstances.

So far only in the Russian Northern Fleet about 110 NSs with 200 nuclear reactors have been withdrawn from service for subsequent dismantlement. In addition, 160 nuclear submarine reactors have been reloaded during running of the Northern Fleet's NSs. Thus in the Northwest Russia the total number of cores which fuel is to be reprocessed exceeds 360.

In the Pacific region the number of reactor cores to deal with makes up 240.

Taking into account the available capacities for SNF processing and temporary storage at PA "Mayak" and the related shipment resources, today SNF of up to 24 cores can be shipped per year using two special trains (i.e. SNF of 2 ½ cores by every freight once every 2-3 months). Such transportation paces would allow removing SNF from storage facilities in Andreeva Bay and Gremikha (Kola Peninsula) only after 15 years at the earliest.

Acceleration of SNF removal from the Northern Fleet's former naval bases would be only possible through the establishment of a new long-term SNF storage facility in less-hazardous area, as compared to Kola Peninsula, e.g. on the Novaya Zemlia (the New Land) Archipelago.

To resolve the challenge of defueling the retired NSs of the Northern Fleet followed by SNF transportation from storage facilities in Andreeva Bay and Gremikha to special transshipment pads for SNF reloading into special railcars and to temporary and long-term storage centers, one needs (along with commissioning of new capacities at PA "Mayak") speed up works on construction of a specialized vessel to ship SNF using special transportation casks.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 223–227.

The necessity of constructing such a vessel results, first and foremost, from an understanding of major actual importance of the issues related to human and environment safety.

The widely known recent sea tragedies – loss of "*Kursk*" and "*K-159*" nuclear submarines – are evidence of the huge scale of potential threats to human beings and the World's ecosystems. However the expenses for salvaging operations, according to "*Kursk*" NS's salvaging experience, are quite comparable with those needed to construct a new specialized SNF transportation vessel.

The collected so far foreign experience of SNF carriage by sea (such transportations have been performed since the mid-1960s) shows that, if appropriate vessels are used, the safety problem is resolved rather successfully: to date the integral flow of the relevant sea-traffic operations nears 20 million container-miles, and no container damage during SNF carriage by sea has ever occurred.

The available Russian Maintenance Vessels (MVs) could be used for SNF shipment purposes, but only in theory. Analysis of their specifications and actual condition shows that safe shipment of large flows of extremely hazardous cargos (containers with SNF) on board of actual MVs is rather problematic.

In fact, Russian naval MVs - Floating Service Vessels (FSVs) design #326 or #326M (the latest FSV was constructed in 1966), and FSV, design 2020 (the latest vessel was commissioned in 1989) - are entirely obsolete. The idea of reequipping special vessel "Amur" (design #11510, built in 1986) has not been implemented either because of various technical and economic reasons.

Similar situation characterizes MVs of the Murmansk Shipping Company (MSC). To date the MSC has 5 MVs used in SNF-reloading operations and storage of Solid and Liquid Radioactive Waste (SRW and LRW).

In theory, FSV "Imandra" could be used for shipment of SNF, LRW and spent sorbents. However this vessel (built in 1981) does not comply with the present-day safety requirements, and its use for shipping Metal-Concrete Casks (MCCs) to SNF reloading/long-term storage centers casts doubts.

MVs "Lotta" (built in 1961), "Lepse" (1936) and "Volodarskiy" (1929) actually used for storage of SNF and LRW are even less suitable for purposes of SNF and Radioactive Waste (RW) shipment.

Thus to ensure safe management of SNF and RW unloaded from NSs and NPSSs, construction and commissioning of a new specialized vessel is necessary. Such vessel would allow shipment of different-type containers with SNF and RW to: -the transshipment pads for subsequent reloading into a special train and -the areas of MCC temporary and long-term storage at special pads of shipyards or in the Novaya Zemlia, the most rigid safety requirements being observed.

To ensure safety when running, the specialized vessel should have:

– increased strength of hull;

- improved floodability and stiffness;
- additional high-efficient fire extinguishants;
- double-side plating in the special-hold areas;
- high maneuverability;
- on-board dose control systems;
- physical protection equipment; and
- extra navigational equipment, communication facilities and radars.
- One should make efforts for the new vessel to be inexpensive in running: 2.5-3-thousand t displacement at a maximum (i.e. by 2-3 times less as compared to the vessels of design #11510) and a relatively minor crew on board – 25-30 men at the most. The vessel should be capable of shipping all-type transportation casks with SNF and should have sufficient carrying capacity for RW transportation.

Unfortunately, today the construction of such a vessel using only the Russian Budget's funds is hardly possible. But considering major importance of this challenge for environmental safety in the Northwest Russia, we set our hopes on the international assistance.

Specialists of Acad. Krylov Central Research Institute have already performed a rather detailed work on designing the general view and determining principal specifications of such specialized vessel.

In compliance with the main functions to be fulfilled and with due regard for the main area of future running (the Barents Sea, the White Sea and the Kara Sea), the specialized vessel is to comply with LU4 Ice Code of the Russian Maritime Traffic Register at the least and be equipped with appropriate reinforcement of the hull to comply with LU5 Ice Code.

Reasoning from the capacity of special train for SNF shipment (12 transportation casks), the new vessel should be capable of shipping 12 TK-18/TUK-108 casks of 40 t each in two cargo holds (an alternative option is also under consideration providing for the vessel capacity up to 18 transportation casks to be arranged within three cargo holds). In addition, a hold for reloading of packages with spent fuel assemblies into the casks and a hold for special-fittings are to be arranged at the stern part of new vessel. Crew's quarters, control rooms, main power installation room and general-purpose rooms are to be placed within the nose superstructure and the fore body of the vessel (Fig.1).

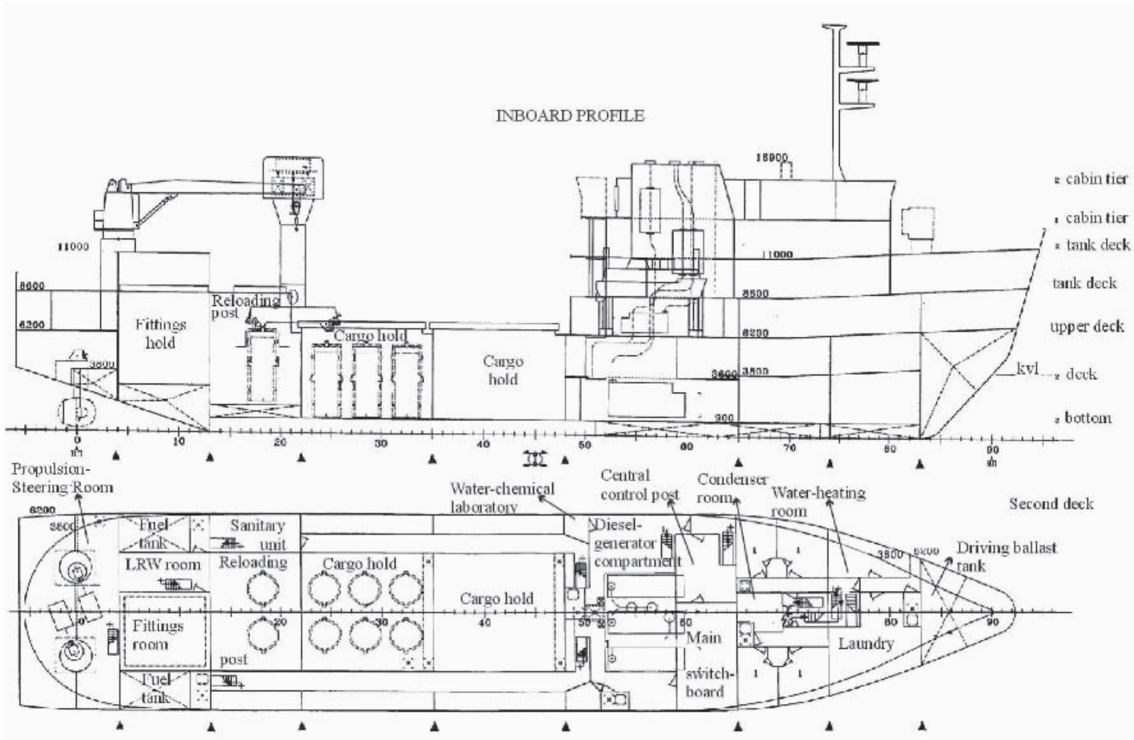

Figure 1. Special container vessel to transport SNF

Principal specifications

Length = 60.6 m	DW = 800 t	Crew = 17 men
Width = 12.0 m	Nps = 2 x 600 kW	S-s = 20 days
Depth = 6.2 m	N_{PI} = 3 · 700 kW DG	V = 12 knots
Draft = 4.0 m	200 kW EmergDG	R = 2000 miles
	Carrying capacity = 12 casks TK-18	

The specialized vessel is to be equipped with: -a 50-t full gantry crane to move TK-18 (TUK-108) casks between the reloading post and cargo holds and -a 18-t-capacity rotary crane to load packages into the vessel from the shore/another vessel. If necessary, reinforced concrete ~ 100-t casks TUK-114 could be reloaded using on-shore facilities.

To ensure safe shipment and observe the requirements of both the Russian Register and the International Supervisory Bodies to MVs, the new vessel should be equipped with: double bottom, double sides, necessary number of cofferdams, biological shielding and physical protection devices, radiation control systems and equipment to ensure the above-water floodability.

Advanced running characteristics, high maneuverability and motion redundancy are to be reached thanks to the use of a new-type propulsion facility with two rotary screw-steering columns. The vessel crew and special personnel (20-25 men) are to stay only in single cabins.

Tentative specifications of the new vessel are listed below:

– length: 60 – 70 m;
– width: 12 m;
– depth: 6.2 m;
– draft: 4 m;
– speed: 12 knots;
– power installation capacity: ~1500 kW; and
– self-sufficiency: 20 days.

If appropriate financing was available, the collected experience and already developed documentation would allow designing such a vessel in the shortest possible time at Acad. Krylov Central Research Institute and “Baltsudproekt” Central Design Office with the participation of some other experienced institutions. The vessel could be most successfully constructed at “Zvezdochka” shipyard, Vyborgskiy shipyard, “Severnaya Verf” shipyard, etc.; the vessel running could be supported by the Rossudostroenie enterprises: “Zvezdochka” and “Nerpa” in the Northwest Russia and FEP “Zvezda” in the Far East Russia.

MODULE COMPLEX TO PROCESS LOW-ACTIVE LIQUID RADIOACTIVE WASTE

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Mobile complex to process low-active Liquid Radioactive Waste (LRW) has been developed at A.P. Alexandrov Research Institute for Nuclear Technologies (NITI) on basis of a LRW concentrating Module Membrane-Sorption Facility (MMSF) and a LRW concentrate Mobile Cementing Facility (MCF-S). The mobile complex provides for decontamination of LRW of complex physical and chemical composition up to sanitary standards allowing discharging liquid fractions to open water bodies and hardening of obtained radioactive concentrates into cement blocks suitable for safe transportation, storage or disposal. The mobile complex facilities are made of domestically produced serial materials and components which cost is several times less as compared to that of similar modules manufactured by foreign companies.

MMSF is designed to process contaminated waters with salt concentration up to 10 g/l and volumetric activity up to $3.7 \cdot 10^4$ Bq/l decontaminating them up to volumetric activity 37 Bq/l at the most and obtaining salt concentrates of up to $3.7 \cdot 10^5$ Bq/l volumetric activity. Filtrate ("final water") decontaminated from radionuclides can be discharged or reused, whereas LRW concentrate can be stored or disposed. LRW decontamination using the membrane-sorption technology has similar efficiency as the distillation-sorption technology presently used at Nuclear Power Plants (NPPs) but is free of shortcomings of the latter allowing reaching high purification coefficients from volatile (oil products, ammonia, etc.) and foaming noxious substances.

MMSF of up to 0.5 m^3/h capacity and up to 8 kW/h power consumption comprises: a MicroFiltering Module (MFM), an UltraFiltering Module (UFM), a Reverse-Osmosis Module (ROM) with kettle-sump, an Ion-Exchange Module (IEM), a Concentrate Reagent Softening Module (kettle) and an Ion-Exchange Resin Regeneration Module. To provide for automated control over heat engineering and chemical parameters of ROM and IEM operation and the value of volumetric activity of final water, the modules of: Gamma-Specrtometric Control (GSM), Beta-Spectrometric Control (BSM) and Chemical Control (salt concentration and pH) of final water (CCM) are used. MMSF also includes: a switching unit and a personal computer for collection and processing of coming automated information. MMSF is equipped with automated systems of Heat Engineering Control (HECS) and Dose Control (DCS) ensuring safety of maintenance personnel.

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The modules are connected into two processing decontamination-purification lines. The first line includes micro- and ultrafiltering modules; the second line comprises reverse-osmosis and ion exchange modules which masses and dimensions are demonstrated in Table 1. The first line provides for decontamination from radionuclides and purification from toxic chemical substances adsorbed on suspended particles and being in the pseudocolloid form; the second line ensures decontamination and purification from radioactive and toxic chemical pollutants being in ionic or molecular form. General view of microfiltering and ultrafiltering modules is given in Fig. 1; that of the reverse-osmosis module and the ion-exchange module in Fig. 2.

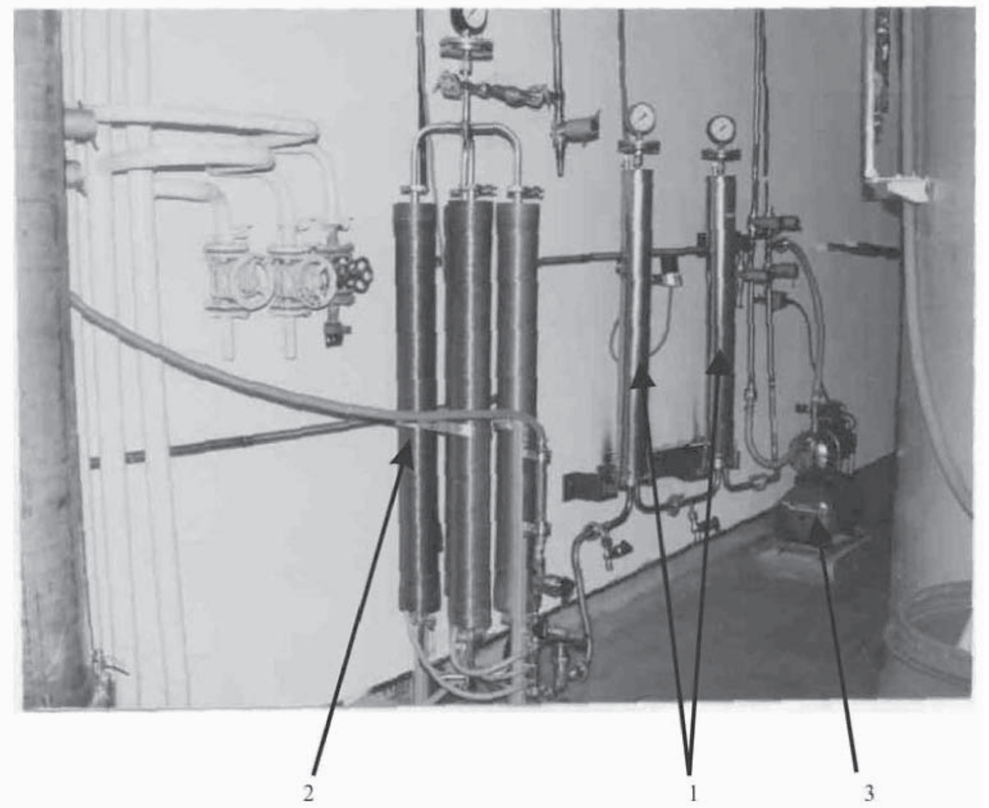

Figure 1. General view of MMSF microfiltering and ultrafiltering modules:

1 – microfiltering modules (MF1 and MF2);

2 – ultrafiltering module (UFM);

3 – pump.

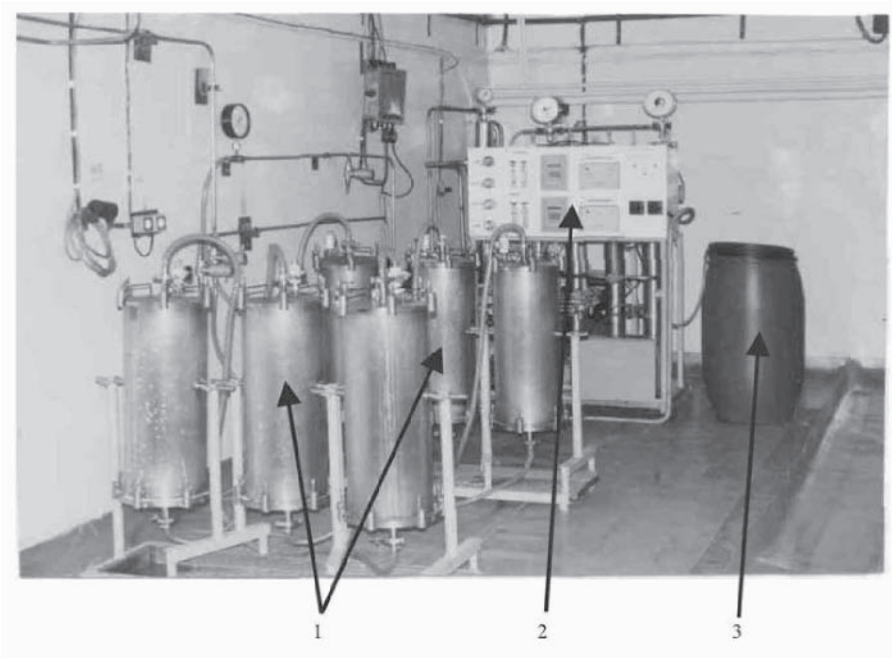

Figure 2. General view of Module Membrane-Sorption Concentrating Facility (MMSF):

1 – reverse-osmosis module;

2 – ion-exchange modules;

3 – tank to wash reverse-osmosis elements.

TABLE 1. Masses and dimensions of MMSF basic modules

Module type	Mass, kg	Overall dimensions, mm
Microfiltering module	20	560·140·1360
Ultrafiltering module	57	860·630·2180
Kettle-sump	120	422·420·6600
Reverse-osmosis module	280	1050·700·1800
Ion-exchange module (loaded)	85 (175)	1170·400·1050
Chemical control module		575·400·700

MMSF can also operate using additional tanks (T-1, T-2 and T-3). Tank T-1 serves to collect input LRW. From Tank T-1 via the first decontamination-purification line (Scheme A, Fig. 3) LRW using pump P-1 via two - primary (20 μm) and fine (5 μm) - microfilters (MF-1 and MF-2) are delivered to ultrafiltering module. After UFM, LRW (concentrate) enriched with suspended particles is returned to Tank T-1, whereas LRW (filtrate) purified from suspended particles enters Tank T-2. Tank T-2 is used as an intermediate capacity wherein concentrate is accumulated. Next, from Tank T-2 via the second decontamination-purification line (Scheme B, Fig. 4), LRW using Low-Pressure Pump (LPP) through microfilter (MF-3) is delivered to the Reverse-Osmosis Module

(ROM). Then salt concentrate enters the kettle and after that it is returned once again to Tank T-2. After passing through ROM desalted filtrate is aftepurified at the Ion-Exchange Module (IEM). Purified water enters the final water tank (T-3) via flow-through chambers of the Gamma-Specrtometric Module (GSM), Beta-Spectrometric Module (BSM) and Chemical Control Module (CCM).

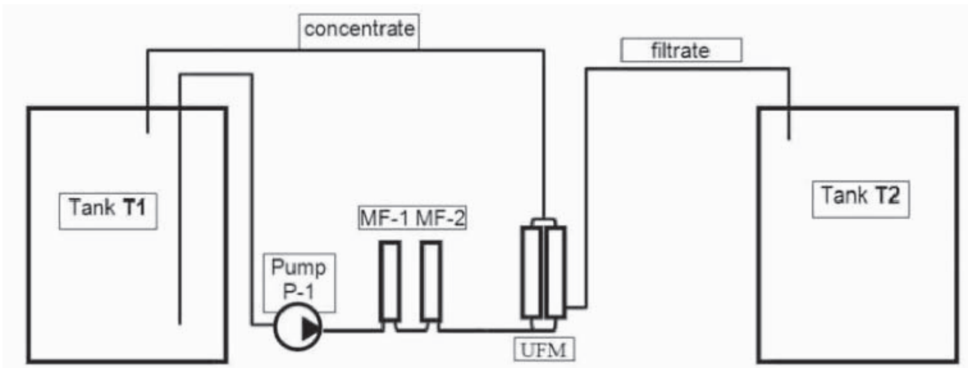

Figure 3. Scheme A. Operation of decontamination-purification line #1

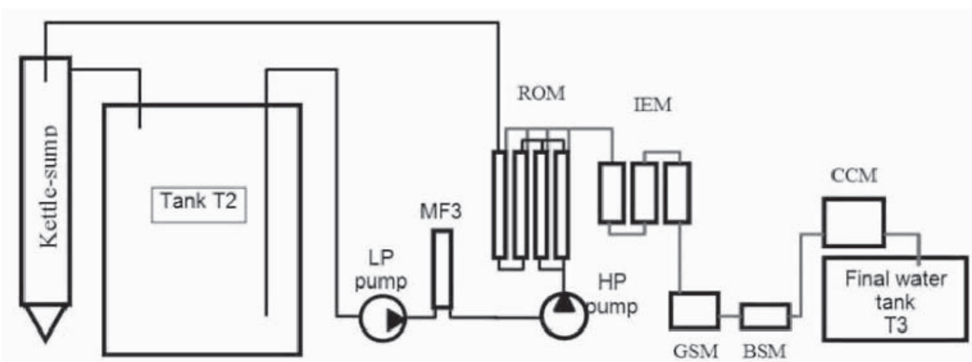

Figure 4. Scheme B. Operation of decontamination-purification line #2

If necessary, the concentrate reagent-softening module provides for correction of chemical composition and radionuclide deposition into tanks of initial LRW T-1 and intermediate tank T-2 for purposes of improving the decontamination-purification coefficient. The Ion-Exchange Resin Regeneration Module is designed to elute sorbed radionuclides and wash both ion-exchange resins and detecting devices of final water control systems.

FPG-type deep cartridge filtering elements of 20 μm and 5 μm filtering capacity are used in microfilters. Element ERU-100-1016 (filtering capacity 10 nm) at a pressure ≤ 0.3 MPa is used in ultrafilters; in reverse-osmosis filters element ERO-KM-100-1016 (selectivity for NaCl 95-99%) at a pressure ≤ 7.0 MPa is used. Ion-exchange filters are charged with cationite KU-2-8 in H^+-form and anionite AV-17-8 in OH^--form or with specific sorbents. Measurements in GSM are performed using BDS-G scintillation NaJ (TI) sensor unit, in BSM using BDS-B scintillation sensor unit, and in CCM using KATS-017TK conductimeter and "KVARTS-pH/1" pH-meter.

The module operation efficiency is determined: on the one hand, by selectivity of membrane elements and sorption characteristics of ion-exchangers and on the other hand, by physico-chemical and radionuclide composition, concentration of suspended particles, salts and radionuclide activity levels. The integral decontamination-purification coefficients $K_{pur_{tot}}$, depending on LRW radionuclide, physical and chemical composition, vary within 10^3 - 10^6. Depending on composition of initial LRW and in compliance with on-line control data, MMSF can operate either under the full-cycle mode involving all basic modules or under reduced-cycle mode using only some of modules.

To obtain economically sound (for cementing purposes) salt concentrates – at least 50 g/l, - the concentrating regime is used. In such a case filtrate is not afterpurified at ion-exchange filters after ROM but enters Tank T-1; consequently, under invariable concentrating level salt concentration in both tanks (T-1 and T-2) increases until reaching at least 50 g/l in Tank T-2. Next from Tank T-2 salt reverse-osmosis concentrate via kettle-sump is delivered for storage/hardening, whereas reverse-osmosis filtrate with increased salt concentration accumulated in Tank T-1 comes for decontamination-purification using standard flowsheet with disconnected UFM.

To harden concentrates of low-active LRW through introducing them into non-organic binding materials, Mobile Cementing Facility (MCF-S) is used (see Fig. 5). In MCF-S (see specifications in Table 2) the process of preparing cement compound is carried out inside the primary package (200-l metal container) with built-in mixer. Next waste is shipped and disposed in the same container.

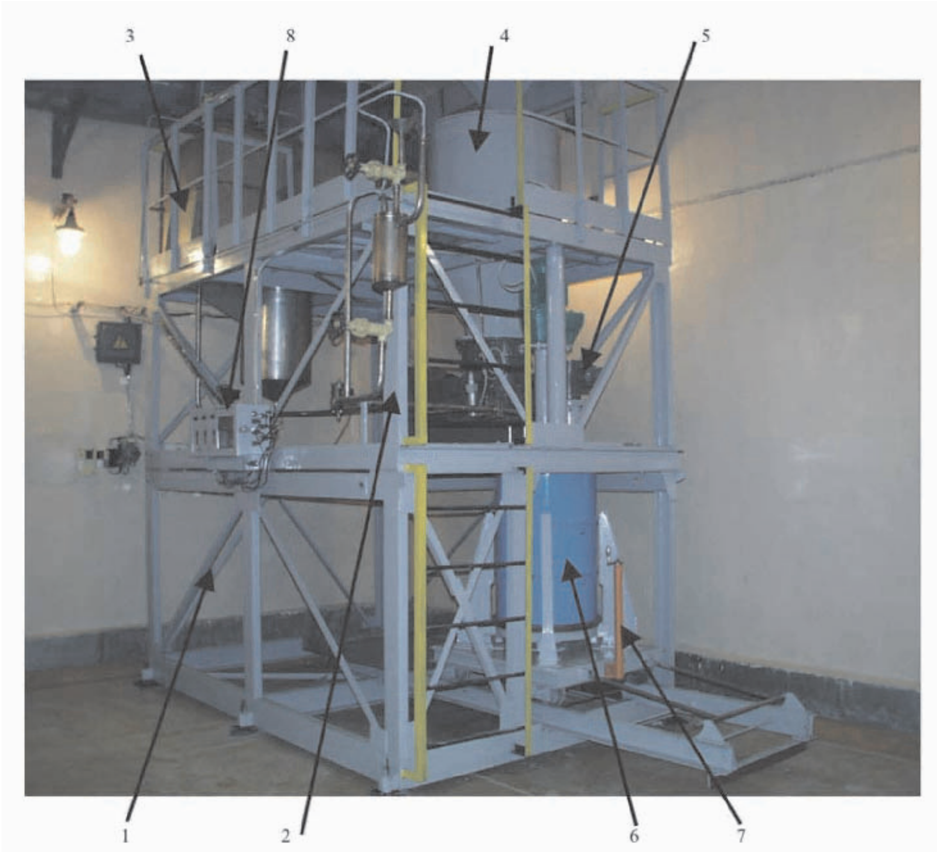

Figure 5. General view of Mobile Cementing Facility (MCF):

1 – transport module; 2 – process module;

3 – intake-dosing device; 4 – cement bunker;

5 – cement batcher; 6 – container;

7 – hand truck; 8 – control panel.

If using such a technology, not only the need of displacing radioactive cement solution is excluded, but also the risk of working site contamination is minimized. Moreover, depending on the type of waste to be hardened, an optimal cement-mixture composition could be assorted for every individual container. Hardened cement compounds have high strength (10 MPa at least) that ensures their safe shipment and low diffusion leaching of radionuclides ($<10^{-4}$ g/cm^2·day) allowing their subsequent burial in ordinary ground repositories.

TABLE 2. Basic specifications of MCF-S

Parameter	Value
Overall dimensions, m	up to 1.2·3,2·4,2
Mass, t	up to 2,5
Capacity, container/hour	2
Container volume, m^3	0.2
Force at container lifting handle, MPa	up to 0.2
Electric power consumption, kW·h	up to 7.5
Supply-line voltage, V	220/380 (+10/-20)
Current frequency, Hz	50±5

MCF operation is organized as follows (Fig. 6). First, cement and clay are loaded manually by operators into the facility bunker (1) up to 0.3 m^3 in volume. Next, batcher (3) 0.12 m^3 in volume is filled with LRW using an independent pump, electromagnetic valve being opened. Operation of both the pump and the valve is controlled from Control Panel (CP), the valve closing and the pump stop being performed in response to a signal of actuation of one (of two) upper-level signaling devices. After that 200-l container (7) with mixer introduced therein (6) is placed onto a hand truck (8) and moved under the loading unit (4) where using a table-lift (9) it is fixed tightly to its lid through which pass: -shaft of the mixing device and –pipes of LRW and compound component delivery. A special sensor gives a signal on leaktight connection of container (7) with loading unit (4), which dog catches on the container mixer. The mixer is set in rotation by loading unit dog via an electric drive guided from CP. Under permanent mixing from batcher (3) some measured off LRW amount is delivered to container (7) via opening of electrical-drive valve guided from CP. From bunker (1) via loading unit (4), which drive is guided from CP, a calculated amount of cement with clay is delivered to container (7). Then the cement mass is mixed by mixer (6) over a necessary period of time. When cement compound is prepared, contained (7) is disconnected from the loading unit (4) and is removed from the facility. The prepared container with cement compound is covered with a special lid and next is placed to “hold up” for compound hardening, after that it is forwarded to a burial area.

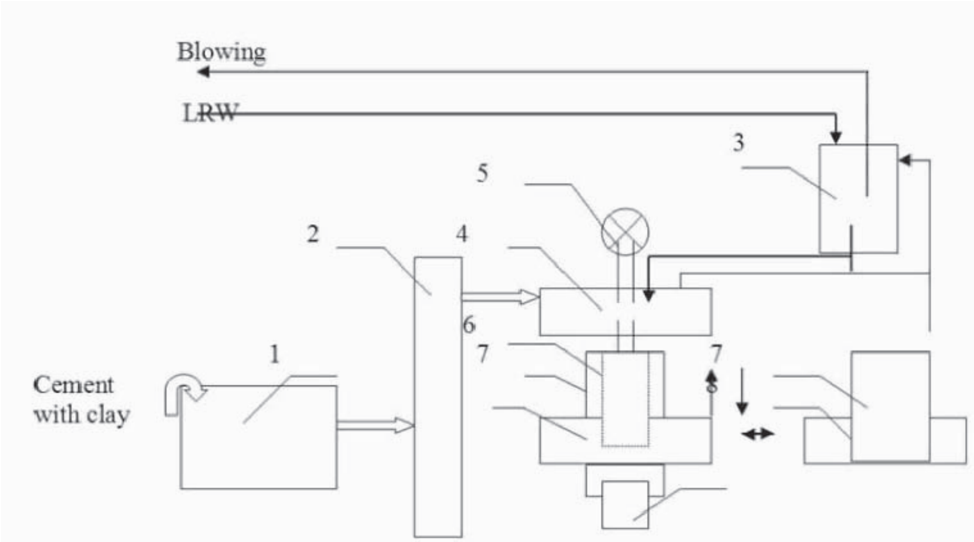

Figure 6. Schematic diagram of Mobile Cementing Facility (MCF-S) operation:

1 – cement bunker; 2 – cement intake-dosing device;
3 – LRW batcher; 4 – loading unit;
5 – motor reducer; 6 – replaceable mixer;
7 – container for cement compound; 8 – hand truck;
9 – table-lift.

Specifications of LRW to be hardened:

volumetric activity, kBq/l	up to 10^3
operating temperature, ^{0} C	10-30
operating pressure, MPa	≥ 0.1
pH	4-12
suspended particle concentration, % mass	up to 20
ballast salt concentration, g/l	up to 200.

The mobile complex facilities have many advantages, such as: possibility of multi-purpose application, low power consumption, compactness and simple engineering design allowing their installation and running in stationary and "field" conditions. A MMSF prototype has been successfully operating at NITI since 1998. In 2003 MMSF #2 was delivered to "Applied Chemistry" Russian Research Center. At present a possibility is considered on MMSF use at "Radon" Special Combine (Saint Petersburg). MCF has been successfully running at NITI since 2001. In 2001 a MCF-S was delivered to "SevRAO" Branch #2 in Ostrovnoy-town (Murmansk region).

To date implementation of the module facility complex is hindered by lack of legally accepted procedure of standard justification of such-type technologies. Only standard documentation concerning stationary radioactive waste processing facilities at NPPs has been developed so far. There is urgent need of developing standard documentation especially for mobile module facilities.

FOREIGN ASSISTANCE FOR HANDLING SPENT NUCLEAR NAVAL FUEL IN RUSSIA: SETTING PRIORITIES[i]

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For Russia's foreign partners, prioritizing tasks related to spent fuel management first involves identifying urgent needs, potential bottlenecks or gaps in assistance programs, safety and security issues, and possible technical solutions. Partner goals, and partner governments' ability to explain how potential assistance projects meet those goals, are additional critical factors in choosing which projects to undertake in Russia. The goals of Russia's partners vary, but they generally share the desire to avoid environmental accidents and the theft of any radioactive materials, particularly spent fuel. The elimination of bottlenecks and gaps is also critical, since they can not only cause inefficiencies but also lead to security and safety problems. This paper identifies some of the factors needed for foreign partners to set priorities, focusing on the area of handling spent submarine fuel, and reviews which tasks address critical, time-sensitive goals, and which might be less urgent. Finally, it makes some observations regarding how certain tasks are interrelated.

While foreign countries have cooperated with Russia in the sphere of nuclear submarine dismantlement for a decade, and have learned a great deal during that time, this cooperation is still not as successful as it could be. In part, this appears to be caused by a continued lack of detailed information, difficulties in project coordination and planning, and political hurdles in partner countries resulting from a popular perception that programs are running into unforeseen hurdles.

In setting foreign assistance priorities, it is important to consider how that assistance can have the biggest impact as soon as possible, along with working toward more distant goals. Some tasks, such as handling spent fuel, storage containers, or other equipment that is deteriorating and becoming increasingly dangerous, are urgent. Successful action in the near future is needed to reduce risks. Such action has the additional multiplier effect of improving the political climate in Russia's partner countries, making further assistance projects more likely to succeed at home – which is critical if they are to prevail in Russia. Thus, Russia's partners want to be able to identify projects that will improve the situation as soon as possible, and do so successfully.

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How can such projects be identified? The Strategic Master Plan for Northwest Russia has been tasked with identifying urgent tasks. To do so, its authors have created a detailed matrix that factors in risks and dangers (in particular, it would appear, focusing on nuclear safety issues). It is hoped that the plan will take security, bottlenecks, and efficiency into account too, along with other relevant factors such as donor and Russian interests, existing technology, etc. It would be most useful if the plan could break down large, long-term needs, since most foreign donors can only take on manageable tasks. Identifying an entire facility and all possible tasks at that site as a priority is not as likely to elicit a quick response as identifying the urgent, concrete tasks at that site. The plan should then explain how these most critical tasks fit together with all of the other elements needed at the site. Only if projects are clearly identified, will Russia's partners be able to set long-term priorities and move forward with confidence.

Information

The first main need, in order to prioritize tasks related to handling spent nuclear fuel (SNF), remains information. If the information needed does not exist, there has to be funding to investigate and obtain that information. Then the information must be shared to be useful – not with the public, if that would create new security risks, but among all relevant parties doing work in this area. This is the only way to identify potential problems, whether they are bottlenecks, security concerns, potential political problems, or dangerous gaps.

Information that remains sorely needed includes reliable data on locations and conditions of storage, amount, type and activity of SNF. Russia's partners need a clearer picture of what is needed to determine the condition of SNF in the various storage containers at all sites. At present, there is information on a large portion of Andreyeva Bay and Gremikha SNF, but not all. There is no information on damage (how much fuel cannot be reprocessed and needs permanent storage/new containers/other special assistance?) If it is impossible to assay this fuel without removing it from damaged containers, then a solution has to be found that involves this removal. For instance, some donors have suggested providing relatively cheap containers for temporary storage, providing the double benefit of hard data on the status of the fuel (needed for further planning) while immediately improving storage conditions.

According to experts, the condition of many of the SNF assemblies at both Andreyeva Bay and Gremikha, in Northwest Russia's Murmansk Oblast, have been reported as "unsatisfactory"—that water has penetrated containers—but what does this mean for their long- and short-term handling? Are they likely to leak liquid radioactive waste (LRW) in the near future? Are increasing problems expected or have they been stabilized? Answers to these questions are needed to determine the level of urgency. Norway, the United Kingdom, and Sweden are undertaking a lot of work in Andreyeva—is it enough to stabilize the situation, such that more urgent needs elsewhere (perhaps Gremikha) should take priority, or does it remain a critical area? Have security measures been upgraded to make certain infrastructure upgrades have not made materials more, not less, vulnerable to theft? Since there is a problem with removing SNF from Gremikha at the current time (particularly since there is no other facility ready to house that material—facilities at Andreyeva Bay should probably be secured and cleaned up before yet more fuel is moved there), what can be done to make SNF in Gremikha safe and secure for the near future? Given Gremikha's remote location, it is most unlikely that the facility faces an outside threat; securing the facility should be easier than securing sites that are easily accessible by road. It may make sense to safely secure SNF at Gremikha on-site, in order to free up resources for more urgent tasks elsewhere in Russia (including Kamchatka and Primorye, in the Russian Far East), and come back to it in a few years when higher priority tasks have been completed.

Those planning assistance projects also need to know how much solid radioactive waste (SRW) and LRW is likely to result from the repackaging of legacy SNF. A rough estimate is critical to determining the LRW and SRW treatment capacity and storage needed at each location. Is assistance needed to undertake such an estimate? Will the upgraded LRW treatment facility at Atomflot, in Murmansk, ever be completed, and, if so, will it be able to handle all of the resulting LRW? If this facility is to be used for LRW treatment, what is needed to transport the LRW to Atomflot? Finally, what can be done to bring the LRW facility at Atomflot up to code, so that it may commence operation?[ii]

Finally, Russia's partners must realize that there is information that Russia is not willing to provide, for military reasons, or cannot provide because obtaining that information would be excessively difficult. Therefore, partner countries must be careful to request only that information that is critical for successful project management. For instance, obtaining detailed information on the characteristics of individual cores, and calculating accident risks, would be very expensive and time-consuming. Therefore, Russia has suggested using the average radionuclide composition to make these risk calculations. While making basic accident predictions based on an average core is quite reasonable where fuel is "normal," and time is of the essence (delay in taking care of this SNF increases the likelihood of accidents), is there fuel that strays far from the average? If so, it is critical that this SNF be identified, and risks calculated for the extreme, not just the average. A solution must be found that makes it possible to identify damaged fuel. For planning purposes, it is also important to have some idea of how prevalent this problem is.

Security

Another critical need is that all projects address security issues, in the broadest sense. In addition to radiation safety and environmental concerns, the increase in terrorist activity of the recent past suggests that even if thefts of SNF or attacks on facilities are not very likely, the high level of potential damage from such an event means that security concerns should remain at the fore when prioritizing actions. This would appear to imply that securing SNF outside of nuclear submarines should take priority over further defueling.

In considering facility security, *access* is a critical consideration. This implies that those sites that are closer to population centers or accessible by road face higher security risks than those in more remote locations that are inaccessible much of the year. Therefore, facilities in the city of Murmansk (in particular, Atomflot) and the Russian Far East face a higher threat level than those in Gremikha or Andreyeva Bay. To date, very few of Russia's partners have focused on the Russian Far East. It is important that a strategic master plan be written for that region as well, and that it include information about nearby roads and populations centers, so that Russia's partners can appreciate just how urgent the needs in that region remain.

The *human element* of security has also been given insufficient attention to date. The reports in May 2004 regarding thefts of nearly 30 metric tons of titanium bulkheads from Sayda Bay indicate that, although engineering measures might be helpful, the biggest security gap in that case was the human factor.[iii] Even journalists from the conservative military paper *Krasnaya zvezda* pointed out that that there were several guard posts to pass, and that guards at Sayda quickly come to ask unfamiliar people at the site what they are doing. However, guards failed to stop the thieves, who used portable welders and made multiple trips into the site. Former military personnel and Gadzhiyevo police are among the suspects. In a separate incident that underlines the importance of the human factor yet further, Russia's General Prosecutor Vladimir Ustinov stated that the Fall 2004 terrorist attacks on Russian airplanes were apparently enabled by a ticket speculator and several airport employees, including the head of counterterrorism at the airport.[iv] In order to ensure physical protection, the human factor must be fully appreciated and addressed—through training, education, increased salary, or perhaps a system of bonuses, as well as technical measures. Foreign donors should be involved in upgrading human resources, not just equipment, or the equipment will prove to be of little use.

Physical protection during *transportation* is also critical, since it is during transport that materials are generally thought to be most vulnerable. The United States has assisted in upgrading the security of some service ships and railcars. However, further upgrades are needed. Here too, it is important to know how the human element is being handled, and what might be done to improve personnel reliability.

Efficiency

In order to prioritize programs, Russia and its partners must also take efficiency considerations into account. Inefficiencies greatly endanger public support for programs. In the past decade of foreign nonproliferation assistance in Russia, there have been delays, mistaken identification of needs, and wasted funds time and again. One of the causes is the lack of clarity of relevant Russian plans and intentions. Of course, plans can and do change, particularly as new information comes to light. Nevertheless, it is important that Russia identify its plans as clearly, completely, and in as much detail as possible. Where spent fuel is concerned, this means how much will be reprocessed (and when and where), how much will have to be stored for a long period of time (when and where), and how much is damaged. What are the costs and risks of transport, vs. the difficulties and costs of construction and continued physical protection in the Arctic? Further, local political issues must be understood: is there permission for the construction of new storage facilities, if they are needed? Have the public been included in the decision-making process so that unexpected protests are less likely to arise? Foreign partners too must pay attention to these issues, in order to provide the most useful and successful assistance possible.

Transportation

Since programs in this sphere imply moving a large quantity of SNF a great distance, and brings SNF to geographic areas that are more accessible than most naval sites, transportation must be given particular attention.

While the needs of the Russian Far East and Northwest Russia have generally been examined separately, where transport and long-term storage is concerned Russia's needs should be considered as a whole. As the same railcars are used for the entire country, calculations of capacity, frequency, and timing must be calculated to maximize the removal of SNF, including the SNF unloaded in previous years, from the most vulnerable sites first.

Upgrading service ships that transport SNF from shipyards to railroad access points is a critical bottleneck in moving SNF to Mayak or elsewhere in both the Northwest and Far East. PM-12, the only such ship in the Northwest, needs an overhaul, or, better yet, a replacement. This should be a top priority projects, as it is a critical transportation link (there is no land access to most of these locations), and the SNF must either be moved (the current plan) or put into better, more secure storage on location (a possibility that should be explored as a temporary measure if transportation issues and more permanent storage locations cannot be readied in the very near future).

Possible dangers along the transportation route need to be identified. Foreign assistance providers need assurance that the entire route, from submarine to storage facility, has been examined for dangers—from weak bridges, tunnels, roads, or rails to locations with heightened security risks. They should be willing to support research in this area, as well as projects to eliminate potential hazards, to be certain they are not contributing to accidents or other dangers.

Transport security extends beyond the time when materials are actually being transported to securing transport vehicles during loading and even when they are not in use, so that they cannot be tampered with, damaged or misused. Thus, Russia's plans should include information about the security situation at temporary storage locations. And the analyses that lead to the identification of priorities should include a security analysis determining whether it is safer to leave newly removed SNF at new, temporary sites and move the old SNF from current storage facilities, or slow submarine defueling in order to remove the old SNF (if this material will indeed be sent to Mayak).

Environmental Safety

Environmental safety is another priority area of concern, for those involved in on-site activities as well as nearby populations. Russia has provided a great deal of information about current radiation levels. In addition to on-site radiation monitoring, however, every site requires offices trained and equipped to handle emergencies. There must also be monitoring and a response capacity during transit. If any rapid upgrades can be made or equipment sent to improve environmental safety in the very short run, before much of the dismantlement work is completed, then this should be a top priority.

Conclusion

If environmental safety and physical security are our top priorities, then what work might be postponed? Not all problems can be tackled at once. While the defueling of liquid metal cooled reactors is an important long-term goal, how urgent are the risks involved in maintaining these reactors, and what is the minimum that can be done to buy time for longer-term solutions? The technical and scientific aspects of designing methods to manage these reactors may certainly attract foreign partner participation for scientific reasons. Assistance for these reasons, though, cannot be viewed as contributing to near-term environmental and security improvements in the same way that other projects do. On the other hand, if there are real dangers associated with postponing LMC reactor dismantlement, what is the level of risk? What must be done to determine if the huge effort and cost involved in finding a way to handle LMC reactors is as urgent as improving site security, removing SNF and other items that are most likely to contaminate the environment in the near future, upgrading service ships, or removing legacy spent fuel? While uncontrolled chain reactions in an LMC reactor sound frightening, if scientific study indicating that a reaction caused by water leaking into the reactor will halt as soon as the water evaporates is reliable, the safety risks posed by these reactors may indeed be far more long-term than the more likely problems posed by SNF in poor storage conditions at on-shore technical bases. The LMC reactors, meanwhile, pose almost no security threat, while the SNF poses the highest such threat.

Prioritization means making choices. Russia’s partners need to understand what these choices involve. Only partner countries can set their own priorities. But they need to have information to choose the most useful projects, and justify their contributions at home. Long-term projects are also necessary, but are more likely to succeed if partners can see that their efforts to deal with urgent tasks are successful.

[i] The author would like to thank the Nuclear Threat Initiative for funding her research on Russian nuclear submarine dismantlement issues. This research included interviews via telephone and e-mail with officials from the Russian Ministry of Atomic Energy, U.S. Departments of State, Energy and Defense, Canada's Department of Foreign Affairs and International Trade, Norwegian Radiation Protection Authority, Japanese Foreign Ministry, France's Technicatome, Swedish Nuclear Power Inspectorate, United Kingdom Ministry of Trade and Industry, and European Bank of Reconstruction and Development, and experts in the field. However, the opinions expressed here are the author's alone.

[ii] The Murmansk Initiative, the U.S-Norwegian-Russian project launched in 1996 to increase the capacity of the experimental liquid radioactive waste treatment facility at Atomflot to industrial scale, has ended but the facility has yet to begin operation. At present, information about the project continues to be lacking. Rosatom has yet to inform its foreign partners, despite numerous requests, of the exact nature of the continuing flaws in the facility, and what must be done (at what cost) to bring all equipment up to code. "The Murmansk Initiative-RF: An observer's point of view," Bellona Website, http://www.bellona.no/en/international/russia/navy/northern_fleet/decommissioning/31925.html, December 5, 2003; Ole Reistad, "Naval Nuclear Clean-Up in Northwest Russia: Lessons Learned and a Roadmap to Completion," Strengthening the Global Partnership Project Issue Brief, http://www.sgpproject.org/publications/publications_index.html#SGPIssueBriefs, November 2004.

[iii] For detailed information on the Sayda Bay thefts, see "5/20/2004: Titanium Stolen from Retired Russian Submarines," Russia: General Naval Developments, *NIS Nuclear and Missile Database*, NTI Website, http://www.nti.org/db/nisprofs/russia/naval/nucflt/gendev.htm.

[iv] See, for instance, Boris Yamshanov, "Vzyatka pakhnet geksogenom," *Rossiyskaya gazeta*, 16 September 2004.

MANAGEMENT OF DAMAGED AND NON-REPROCESSIBLE SPENT NUCLEAR FUEL

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1. Introduction

To date at least 2 fundamental problems related to management of Spent Nuclear Fuel (SNF) of transport reactors are pending decision: -removal and reprocessing of damaged fuel and –management of non-reprocessible fuel. Real situations resulted from protracted storage of SNF at naval Floating Service Vessels (FSVs) and Coastal Maintenance Bases (CMBs) accompanied by violations of storage regimes (non-observance of requirements to water quality in cooling ponds and temperature conditions, leaks out of shrouds housing Spent Fuel Assemblies (SFAs), etc.) have caused considerable uncertainties when estimating factual status of SNF.

The presence of many SFAs depressurized while in operation is the reason of additional uncertainties. The possibility of further destruction of SFAs during storage has been confirmed by many experiments; according to their results, actual condition of some SFAs causes serious anxiety.

Though lack of statistical data still gives no way of performing a credible estimate of the proportion of damaged (non-reprocessible) spent fuel in the non-processed yet amount of SNF of transport reactors, if taking the fraction of such SNF at 10% (a very optimistic estimate in opinion of many specialists), the issues of damaged SNF management appear to be very topical. There is no way of performing rehabilitation of the former naval CMBs without adequate solution of this challenge.

The notion of "damaged fuel" should not be interpreted unambiguously. Not quite correct use of the term “unsealed SFA” (i.e. SFA comprising unsealed fuel elements) as a criterion of damage allows considering over 50% of the remaining non-processed SNF as “damaged” fuel. Thus one needs refining the very definition of “damaged” SNF, the main criterion being the possibility of SNF shipment for reprocessing and SNF reprocessing itself in compliance with standard/specially developed technologies.

One also needs considering potential fuel degradation mechanisms during storage taking into account SNF storage duration and conditions and estimating factual status of non-reprocessed SNF stored, firstly, in Andreeva Bay. Based on the results of a forecasting estimate, some measures could be proposed aimed at facilitating SNF

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shipment for reprocessing. Elaboration of appropriate recommendations on managing non-reprocessible SNF is also necessary.

2. Ranking of SNF condition

To estimate SNF condition, different approaches can be used. The “sealing” criteria characterizing fuel elements and the entire cores, while in operation, based on the results of radiation and chemical control allows identifying SNF condition in the most reliable way. However such an estimate gives practically no way of determining the possibility of SNF shipment for reprocessing because it is deemed that all SFAs unloaded from reactors using standard procedures can be certainly reprocessed.

The duration of SFA staying at storage facilities represents an additional factor contributing to generation of different defects, which could have effect on the very possibility of SNF reprocessing. Depending on the initial condition (“sealing” status) and storage peculiarities, further depressurization of SFA canisters is possible (in the worst cases even fragmentation of fuel elements). As the result, their SNF could become non-reprocessible.

With due regard for the above said, possible states of fuels elements and their typical characteristics are considered below.

2.1. REQUIREMENTS TO SFAs TO BE FORWARDED FOR REPROCESSING

In compliance with the standards in force, SFAs unloaded from reactors with no damage, easily installable into special shrouds under the gravity effect and having no defects of the gripping mechanism (head) are to be delivered for reprocessing.

When transferring SFAs, the supplier executes and submits the relevant certificates for SFAs in shrouds to representatives of the reprocessing plant. Such certificates include: initial characteristics, power generation data, core number, date of the first putting into operation and that of the last reactor shutdown before SNF unloading, the operational documentation of the relevant reactor installation being the source of such data.

Because many documents describing SNF stored at CMBs were lost, PA “Mayak” issued a special decision for acceptance of SNF with unknown history.

2.2. SNF RANKING BY THE RESULTS OF RADIATION AND CHEMICAL CONTROL

When running a reactor installation, the control over activity of coolant is performed according to the established procedure. The methods of control and its periodicity are determined by a special standard document specifying 3 possible conditions of reactor cores while in operation.

Normal condition presumes intact fuel element claddings. In this case fission-fragment activity of coolant is determined by either surface contamination of claddings by uranium occurred during fabrication, or by residual contamination of fuel composition transferred to coolant from unsealed cores of previous lifetimes.

Admissible condition means the presence of microcracks in claddings through which gaseous and volatile fission products (noble gases, iodine isotopes) are released to coolant.

If the damage increases, the core transfers to inadmissible condition, and the reactor installation running must be stopped. In such a case, fuel composition contacts directly with coolant that is evidenced by α-activity therein.

It should be pointed out that the above criteria are the integrated ones describing the state of reactor cores as a whole. Of SFAs having been in operation mainly before 1978 and being actually stored at storage facilities, over 30% are in “inadmissible” condition; the condition of about the same amount of SFAs is estimated as “admissible”.

Protracted storage could considerably contribute to SNF degradation. Fuel defects appeared first during reactor operation could develop further in the course of storage affecting appreciably general condition of SFAs. The consequences of SNF degradation could become rather serious depending on duration and the specificity of storage conditions.

After estimation of potential consequences, appropriate engineering solutions and procedures should be recommended and developed with the ultimate aim of attaining maximum possible fuel amount to be transferred for reprocessing.

3. Real status of damaged fuel

3.1. RESULTS OF CONTROL OVER CLADDING SEALING

As said above, the activity of coolant represents an integrated characteristic of intact condition of fuel, while in operation; however this criterion cannot be used when estimating the sealing status of individual SFAs. However it is the condition of individual SFAs at reactor defueling instant that is of much interest because subsequent (during storage) fuel degradation depends in many respects on its “initial” (at the unloading instant) condition, i.e. on the type of failures.

To obtain such information, special measurements are necessary - the so-called Cladding Leakage Test (CLT). CLT is performed via gas blowing through heated Fuel Assemblies (FAs). Though so far both the equipment and procedure of such tests have been sufficiently worked through, CLTs are still laborious and expensive requiring rather sophisticated equipment. Consequently, CLTs could hardly be used for purposes of large-scale control over SFA condition.

CLTs are routinely conducted at the cores of Russian nuclear icebreakers.

Analysis of the results reveals that a monotone increase in activity of coolant during operation is caused by seal failure of a considerable proportion of FAs, the damage level differing considerably from one FA to another. Considerable damages transferring the core to inadmissible condition are recorded at 6–10% SFAs. The majority of remaining SFAs has only gas leakage.

It is worthy of notice that in rather rare cases of drastic activity increase (by 4 to 5 orders of magnitude over several hours) CLTs are capable of recording seal failures at individual (1–3) SFAs.

3.2. STUDYING OF INDIVIDUAL SFAs

CLT allows selecting a specific object for the following investigation stage – estimate of fuel-damage level using factual condition of fuel elements within unsealed SFAs. Russian Research Institute of Atomic Reactors (RIAR) performed investigations of SFAs selected – on basis of CLT results - from different cores being mainly in “inadmissible” condition.

Those investigations revealed the following 2 types of fuel rod defects at unsealed SFAs:

- short (2–3 mm) through cracks over the outer surface of claddings without edge opening; and
- long (up to 10 – 15 mm) cracks with edge opening (0.05 – 0.2 mm).

There is virtually no contact between the primary circuit medium and coolant in case of the first-type cracks. During operation only gaseous and volatile fission products could be released via such cracks to the primary circuit medium. The second-type cracks could cause fuel corrosion and washing out of soluble fission products (cesium, strontium) and fuel particles to the coolant circuit.

However in standard SFAs of cores, which transfer to “inadmissible” condition has occurred not at once, the number of depressurized fuel elements does not exceed 50%.

The performed analysis allows suggesting rather optimistic general conclusions. Firstly, most of SFAs of the cores transferred to “inadmissible” condition either have preserved their intact condition or have got only minor seal failures. Secondly, even in considerably damaged SFAs most of fuel elements have no through damages of claddings.

The above generalizations allow applying a differentiated approach to the development of degradation processes during protracted storage of SFAs.

4. Fuel degradation mechanisms during storage

4.1. CONDITIONS AND OPTIONS OF SNF STORAGE

Based on analysis of possible and already implemented options of either storing SNF of NS reactors before loading into transportation casks and forwarding for reprocessing or SNF storing in containers, the following classification of spent nuclear fuel can be proposed:

4.1.1. By storage areas

- NS reactors
- Floating Servicing Vessels (FSVs)
- Coastal Maintenance Bases (CMBs) and
- Storage facilities of research laboratories.

4.1.2. By storage conditions

- "Dry" storage in intact/damaged shrouds
- "Wet" storage in standard-quality medium/off normal-quality medium and
- Mixed storage.

4.2. NS REACTORS

At present fuel is stored within NS reactors in a “wet” medium which condition complies with the requirements of relevant standard documentation.

No cases of medium composition deviations from standards in NS reactors have been ever reported. However in NS reactors with leaky primary circuits increased oxygen concentrations can be recorded in the storage medium. The medium of such NS reactors (as well as that of NSs with damaged cores) is inhibited using phosphates.

Trial storage of SNF with one-every-three-month control over water-chemical indices and coolant activity has been performed at 6 reactors of *Yankee-2*-class NSs (4 cores being in “admissible” condition). The control has revealed stable state of the medium and stable activity indices, the latter being of crucial importance evidencing no fuel degradation during storage. It is possible that the presence of minor - virtually unopened - cracks on fuel element claddings, though leading to release of gaseous fission products while in operation, does not result in substantial fuel-coolant contact and, consequently, in active corrosion of fuel composition with release of Cs and Sr ions to the circuit.

Presently a considerable amount of SNF is stored in unwatered reactors. It is expected that “dry” fuel storage will increase nuclear safety level and decrease the rate of leaky fuel degradation.

4.3. FSV STORAGE FACILITIES

Since 1984 SFAs have been stored at storage facilities of FSVs in “dry” shrouds placed into filled-with-water cooling ponds. Thus if such shrouds are intact, SFAs are stored in virtually dry medium. There is an excess of cladding temperatures above that of the environment throughout the storage period due to decay heat in fuel elements, and thus no water condensation on such claddings occurs.

The following quality standards for cooling-pond water are prescribed by a special service instruction:

Chlorine-ion concentration	≤ 0.05 mg/l
Salt concentration	≤ 1,5 mg/l
pH (at 25^0 C)	5 – 7

According to the available information, chlorine-ion concentration at virtually all FSV storage facilities exceeds 5 mg/kg, salt concentration being ≥50 mg/kg. In individual situations such concentrations could cause damage of shrouds resulting in direct fuel-medium contact with a variety of implications.

4.4. COASTAL MAINTENANCE BASES (CMBs)

Having been in operation since the mid-1960s, SNF storage facilities at CMBs were designed as special ponds, which cells were to house dry shrouds with SFAs. CMBs were initially designed for interim storage of SNF before loading into transportation casks and subsequent forwarding for reprocessing.

It was anticipated that after nuclear vessel defueling shrouds with SNF would be stored at FSVs, next fuel in the same shrouds would be transferred to CMBs and then reloaded from CMB storage blocks to transportation casks.

When it was established via calculations that the temperature on cladding surface under "dry" storage did not certainly exceed the admissible value, a decision was made on water removal from cooling ponds.

However the attempts of attaining true "dry storage" have failed. CMB storage facilities being subject to seasonal temperature variations accumulate condensate and atmospheric precipitations. At the initial storage phase appreciable amount of moisture penetrated damaged shrouds; at the subsequent storage phase they were affected by rainwater and snow. Considerable activity of the medium measured in shrouds (and previously in the cooling ponds) is evidence of serious SFA damages.

Repeated freezing of water in shrouds could lead to: considerable deformations of SFAs, difficulties when extracting from shrouds and, possibly, fuel element fragmentation.

4.5. MECHANICAL SCHEME OF DEGRADATION

Previously, the effects of duration of SNF storage on its condition were studied using fuel element samples remained after investigations of damaged rod-type SFAs. The studied samples were stored within unsealed cases at RIAR's storages from 1976 to 2000. As the result of storage crack length increase and fuel composition release were recorded.

However the attempts at revealing the degradation mechanism have failed. The hypothesis on corrosion-nature of the degradation processes has not been evidently proved. As a probable hypothesis, one may suggest intergrowth of cracks at claddings embrittled during operation under the impacts of fuel swollen during burnup.

4.6. CORROSION SCHEME OF DEGRADATION

When assessing general condition of SNF, the mechanism of corrosion behavior of fuel assemblies seems also rather important. Many SFAs have been stored in water medium (SNF in reactors, SFAs at FSVs in damaged packages and SNF that prior to "dry" storage stayed in water over a protracted period).

If fuel is stored in non-unwatered reactors, development of corrosion is possible; however, the corrosion processes can be minimized via introducing special water regimes.

Low corrosion capacity of the medium and introduction of inhibitors contribute to only minor corrosion rate and almost no fuel degradation during storage in reactors. On the contrary, the state of SNF actually/previously stored in "wet" or "pseudo-dry" conditions appears to be considerably worse.

Under the "pseudo-dry" storage one understands the presence of water in unsealed shrouds due to either insufficient unwatering of storage pools or atmospheric precipitations. In case of considerable deviations of the storing-medium quality from the established corrosive-admixture-concentration standards rather active fuel corrosion is possible.

No systematic investigations of SNF corrosion resistance in water solutions with different salt concentrations have been performed yet. So far only individual corrosion tests of non-irradiated fuel compositions and fuel element fragments have been conducted.

When estimating potential implications of NS sinking, corrosion tests of SNF fragments in marine water were also performed. According to their results, intact fuel elements under protracted hold-up times (up to 3200 hours) were not cracked even at 80^0C temperature.

An appreciable increase in medium activity was revealed in tests of fuel elements with simulated defects/cracks generated during operation. The activity release rate increased during the tests evidencing flaw growth.

The corrosion processes are developed as follows:

- via the available flaws the corroding medium contacts matrix composition comprising dispersed fuel particles;
- fuel corrosion leads to transfer of fission products to water and thus to increase in the medium activity;
- corrosion of the matrix composition is accompanied by a considerable increase in volume;
- generated corrosion products contribute to opening and growth of cracks; and
- crack growth intensity depends on salt concentration in the medium, fuel burnup and initial number and development of cracks.

4.7. FUEL CONDITION AFTER PROTRACTED STORAGE

Let us estimate potential consequences of corrosive damage of fuel.

The corrosive processes at fuel elements with minor damage (only gas leaks via individual non-open cracks 1–2 mm in length) develop very slowly; thus in this case no considerable flaw development is expected.

Unsealed fuel elements of SFAs unloaded in “inadmissible” condition have many through damages of claddings: several cracks up to 25–40 mm in length with up to 0.2 mm openness in the maximum burnup area throughout the core height. As corrosion progresses, such-type cracks appreciably grow up. Corrosion of fuel composition results in crack openness increase. In case of important leaks and cracks corrosion processes could lead to release of fuel particles and diametrical cracking of claddings and thus to fuel element fragmentation.

The experimentally established growth rate of cracks (determined through activity increase when corroding) makes up 6–7.5 10^{-4} cm/h at 100^{0}C and diminishes with temperature decrease. The above said allow concluding that protracted many-year storage could cause considerable degradation of unsealed fuel elements.

5. Expected condition of SNF stored in Andreeva Bay

Many SFAs actually stored in Andreeva Bay (≥ 50%) were unloaded from reactors in “inadmissible” and “admissible” conditions. This means that appreciable proportion of SFAs is presently unsealed; according to the CLT results, among the “inadmissible condition” category considerable depressurization characterizes 25–30% SFAs.

During storage SFAa comprising unsealed fuel elements undergo substantial degradation: increase in crack length and openness, spills and, possibly, fuel element fragmentation. Though such processes are more active in corrosive media, in case of considerable burnup they could also develop under “dry” storage. It is deemed that periodic freezing of water in shrouds leading to fuel element deformations and displacements of SFA structural elements makes a major contribution to the degradation processes.

Considering that at first degradation processes affect unsealed SFAs, one may expect considerable damages of ~ 10% of SFAs after many-year storage. This means that after reloading, preparing to and during shipment >2000 SFAs stored in Andreeva Bay could become “non-reprocessible”.

6. Possible options of managing damaged and non-reprosessible fuel

The presence of many damaged SFAs necessitates development of new process technologies which application would diminish the integral amount of “non-reprocessible” fuel. It is also obvious that one would be unable to ship some SFAs to PA “Mayak” due to their poor condition, and thus controlled long-term storage of such SFAs would be necessary.

However so far the problem of storage of “non-reprocessible” fuel (e.g., SNF of liquid-metal-coolant reactors) has been neither raised nor properly formulated yet. The presence of “non-reprocessible” SFAs at CMBs should be also considered under this problem.

To all appearance, a special storage pad should be established at the facility of planned disposal of Solid Radioactive Waste (SRW) where, along with SNF, high-active long-lived SRW would be stored (including rods of the control and protection system). In our opinion, the issue of establishing such a pad (repository or special storage area) should be considered within the rehabilitation activities of the Andreeva Bay CMB.

In addition to the problems of organizing long-term storage of “non-reprocessible” SNF, it would be wise to consider the prospects for decreasing the amount of such fuel via development of special technologies of safe SFA management when reloading, placing into transportation casks and transfer for reprocessing.

As a possible solution, one could propose packaging of some SFAs into special boxes, when reloading from obsolete-design shrouds, to be next reprocessed together with the boxes. For SFAs stored at “*Lepse*” FSV such-type technology has been already agreed between PA “Mayak” and Murmansk Shipping Company. Unfortunately, if employing such a technology, only 21 SFAs could be loaded into one transportation cask (instead of 49 SFAs in standard case) that would considerably increase the cost of transportation.

At present OKBM and Dollezhal Research and Development Institute of Power Engineering (NIKIET) together with Bochvar Research Institute for Non-organic Materials (VNIIIM) examine the possibilities of SFA monolithization using fusible organic compositions. Being easily removable, such compositions would influence only slightly the reprocessing procedure. It is not improbable that the development and implementation of new technologies with application of appropriate materials would allow decreasing considerably the amount of “non-reprocessible” fuel.

Conclusions

1. It is recommended to estimate SNF condition using the “non-reprocessibility” criterion considering the very possibility and safety of reprocessing and SNF shipping costs.
2. Forecasting estimate of the integral amount of “non-reprocessible” fuel depends on duration and conditions of its storage being determined in many respects by SFA status just after operation. The integral amount of “non-reprocessible” fuel in Andreeva Bay could exceed 2000 SFAs.
3. One needs developing general approaches to management of “non-reprocessible” fuel including the establishment of a special long-term controlled storage. Development of new technologies aimed at decreasing the amount of “non-reprocessible” fuel is also appropriate.

A LONG-TERM MOTHBALLING TECHNOLOGY FOR REACTOR COMPARTMENTS WITH DAMAGED CORES

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Among the problems of Nuclear Submarine (NS) complex decommissioning, that of long-term mothballing of Reactor Compartments (RC) with damaged cores in nuclear, radiation and environmentally safe conditions merits a special consideration. Because of damaged condition of fuel assemblies defueling of such cores is impossible, and the currently used in Russia dismantlement technology (making up of three-compartment units for long-term storage) cannot be applied in this particular case.

NS with damaged cores have been forcedly stored afloat for almost 20 years. Such a protracted storage of damaged non-defueled NS kept afloat by pontoons because of driving ballast tank seal failure is a source of radioecological hazard in their basing areas.

Waterborne storage of damaged NS is accompanied by continuous contamination of water area. Only peeling of their rust and paint has resulted in persistent contamination of aquatic systems including bottom sediments. Specific activity of water exceeds the background values by a factor of 2; dose rate values on sea bottom reach 140 mR/h.

An accident at NS K-175 (design 675, serial #175) led to the most severe implications. Due to heat explosion caused by spontaneous chain reaction, the port side reactor core of this submarine was completely destroyed and rejected. Consequently, fuel fragments spread over RC. Control room baffles inside RC were destroyed, and strong hull became damaged. Because of non-welded removable plate and cracks on strong hall RC became unsealed representing a source of radioactive product release into water area. Depressurization of pipelines of the primary circuit led to coolant carrying and spreading over NS compartments that contributed to their further radioactive contamination. Dose rate values measured in different areas of this NS make up: 12-15 R/h in control room baffle, 220 mR/h on RC strong hull and 1.5-3.6 mR/h in inner rooms of nose and stern compartments [1].

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Though the radiation situation at other damaged NS (e.g., NS K-314, design 671, serial # 610) is less hazardous, their damaged cores and high radioactivity levels creating nuclear, radiation and environmental threat still give no way of performing any works due to unacceptably high occupational dose levels.

To bring damaged RC into environmentally safe condition and reduce the risk of hazardous environmental impacts, Central Design Office of Marine Engineering "Rubin" (CDOME "Rubin") together with Nuclear Safety Institute (IBRAE) of the Russian Academy of Sciences (RAS) and Russian Design Institute of Power Engineering (NIKIET) were entrusted with designing and constructing sealed sarcophagi around such RC to prevent any release of radionuclides and induced activity to the environment from the inner compartment space and the outer surface of strong hull. As sarcophagus hulls, missile compartments of strategic missile cruising submarine (RPKSN), design 667B, were proposed [2]. According to the design, such sarcophagi were to be kept afloat. However this option is only a temporary solution of the problem for, according to estimates of Central Research Institute for Constructional Materials "Prometey" (CRICM "Prometey"), their intact condition can be guaranteed for only about 15 years. High occupational exposure doses (exceeding the permissible levels hundreds of times) during making up of such sarcophagi represent one more important shortcoming of this alternative.

Placing of damaged RC into a floating dock followed by disposal of "damaged NS – floating dock" complex within a special area is another RC mothballing option developed by CDOOE "Rubin" jointly with the State Marine Design Institute #23 under Russian Ministry of Defense and NIKIET (R&D "Shelter") [3]. However legal uncertainty (observance of the international agreements in force on safety of aquatic systems) represents a substantial shortcoming of this option.

Thus to ensure reliable mothballing of damaged RC over a protracted (up to 100 years) period of time in nuclear, radiation and environmentally safe conditions, new unconventional design and technological solutions are necessary. When developing an appropriate mode of reliable mothballing of damaged RC, one faces an additional problem – that of dismantlement of other NS compartments. The problem is due to the fact that the present-day cutting and metal-reprocessing technologies cannot be used in this particular case because of high contamination levels.

To ensure reliable and safe long-term mothballing of damaged RC, a new proposal has been recently put forward providing for "monolithization" of both NS inner compartments and the inter-board area. For this purpose a magnesia-mineral-salt composition has been recommended as immobilization material that can be prepared from widely occurring natural minerals (caustic magnesite and bischofite) and metallurgical waste [4].

Immobilization of damaged RC in such a way would be reached thanks to long-term high mechanical strength, chemical, radiation and water (including marine water) resistance of the proposed composition.

In laboratory investigations samples of magnesia-mineral-salt composition had ultimate strength under squeezing of 86 MPa after 28-day hold-up in marine water and that of 117 MPa after the same-duration hold-up in air; moreover, those samples were of extremely low (<1%) surface porosity. To justify the expediency of using magnesia-mineral-salt composition for immobilization of radioactive waste and contaminated constructions, the composition was tested for compliance with the requirements of the Russian Standard "Cemented Radioactive Waste: General Technical Requirements" (GOST R 51883-2002). According to this standard, the initial permissible ultimate strength under squeezing for hardened compounds should be above 4.9 MPa and should retain their properties after different-type operating impacts. Principal characteristics of magnesia-mineral-salt composition are given in Table 1 [5].

TABLE 1. Principal parameters of magnesia-mineral-salt composition

No	Parameter	Value	Comment
1	Mechanical strength under squeezing, MPa	330 - 350	
2	Radiation resistance: decrease in mechanical strength at 10^6 Gy maximum absorbed radiation dose does not exceed 25%; mechanical strength under such impact is above the permissible strength limit under squeezing	5.7 %	Mechanical strength of irradiated samples was determined after 8-day hold-up in storage facility for spent sources. Absorbed dose = $1.2*10^6$ Gy
3	Maximum leaching rate, $g/cm^2 \cdot day$	$< 10^{-4}$	
4	Resistance to long-term staying in water: decrease in mechanical strength over 90 days does not exceed 25%; mechanical strength under such impact is above the permissible strength limit under squeezing	< 5%	Mechanical strength of samples was determined after hold-up over 90 days in marine, sweet and distilled water
5	Frost-resistance: decrease in mechanical strength under repeated freezing- defrosting cycles (from - 40^0C to + 40^0C) does not exceed 25%; mechanical strength under such impact is above the permissible strength limit under squeezing	5.7 %	Mechanical strength of samples was determined after 30 "freezing-defrosting" cycles from –40 to +40°C

Because neither physical nor chemical properties of the immobilizing material in question change over time, "monolithization" of inner volumes of NS compartments would provide for reliable long-term insulation of immobilized structures from the environmental effects. "Monolithized" inter-board area would become a reliable protective barrier against radionuclide release to the environment from the inner compartment space and contaminated outer surface of the strong hull.

Moreover, "monolithization" of NS inner volume would completely "mothball" both damaged cores and primary coolant systems and, simultaneously, would rigidly fix in their actual position nuclear fuel fragments scattered over damaged RC. This would ensure nuclear safety of damaged RC.

Decay heat in damaged RC would not have negative effect on the properties of immobilizing material: already 5 years after the accident (NIKIET's estimates) decay heat did not exceed 300 W and continued further monotone its decrease.

Filling of NS compartments with consistent immobilizing mixture could be performed via standard hatches and openings. The process cycle for "monolithization" of inner volumes of compartments and inter-board areas is illustrated in Figure 1.

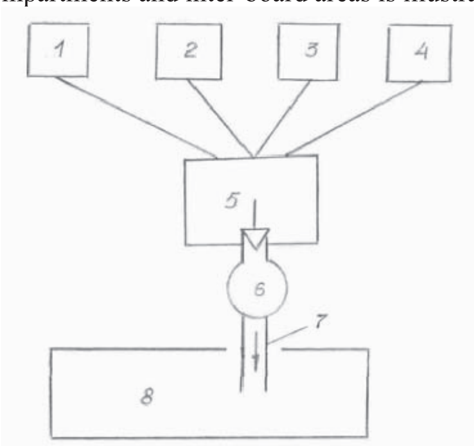

Figure 1. "Monolithization" of inner volumes of NS compartments and inter-board areas

From "magnox feeding unit" (I), "magnesium chloride feeding unit" (2), "mineral filler feeding unit' (3) and "water batcher" (4) the components would enter "mixture" (5). The obtained cream-consistency mixture would be pumped by "pump" (6) via "pipeline" (7) into "unit" (8) for curing. If one has to do with a submerged facility, as immobilizing composition would fill its volume, water would displace. Such a process would go continuously.

Next, submerging of NS immobilized in such a way followed by its deepening into bottom sediments is proposed. Chazhma Bay or Razboinik Bay, where long-term storage facilities for RC of dismantled NS are actually under construction could be used for these purposes. Excavations on sea bottom could be realized through repulping method widely used in fill-ground operations. Bottom sediment repulping could be executed using the present-day equipment of rather high capacity (up to 1000 m^3/day per one unit of equipment with pulp displacement up to 200 m). After NS deepening into bottom sediments its natural silting would occur, and thus no additional sarcophagus would be necessary. Such NS would be completely insulated from the environment and nuclear, radiation and environmental safety of the region would be ensured.

Because magnesia-mineral-salt composition can be made from widely occurring minerals, and the technology of its application is identical to standard cementation, technical and economic indices of the proposed method would be rather high. Because the scope of works using the proposed technology would not be too large, they could be initiated in the near future provided that all operations would be performed using the available equipment.

Another advantage of magnesia-mineral-salt composition to “monolithize” inner volumes and inter-board areas of damaged NS is due to a possibility of using saline Liquid Radioactive Waste (LRW) as an additional mixing liquid when preparing the immobilization mixture. The proposed proportion of LRW (~ 22%) would not influence the compound strength characteristics. Such engineering solution would also contribute to resolving the problem of LRW accumulated at naval bases.

The above technology would also make it possible to defuel the intact starboard reactor core of NS serial #175. After damaged RC “monolithization” the radiation situation within the starboard RC would improve, and thus defuelling under permissible (or close to permissible) occupational dose levels would be possible.

Reliable long-term mothballing of damaged NS in nuclear, radiation and environmentally safe conditions using magnesia-mineral-salt compound followed by NS deepening into sea bottom sediments could be realized in the shortest possible time. The legal aspects of such disposal should be worked over at the international level.

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STANDARD AND REGULATORY SUPPORT OF SAFE MANAGEMENT OF NON-STANDARD SPENT NUCLEAR FUEL OF FLOATING STORAGE VESSEL

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By Decree #401 of July 30, 2004 of the Russian Federation (RF) Government the Federal Nuclear and Radiation Safety Authority of Russia (Gosatomnadzor) was reorganized into the Federal Service for Environmental, Technological and Atomic Supervision (FS ETAS). According to the Decree # 401, the FS ETAS:

– is the regulatory body under "The Convention on Nuclear Safety" and the competent authority of the Russian Federation under "The Basel Convention on the Control of Transboundary Movements of Hazardous Wastes and Their Disposal"; and
– performs standard and legal regulation of the issues related to collecting payments for deleterious environmental impacts.

Because the FS ETAS reorganization has not been finished yet, and it was Russian Gosatomnadzor that developed the below-considered documents, the former name of the regulatory authority – Gosatomnadzor - is used in this paper.

Any activity related to safe use of atomic energy in the Russian Federation, including safe management of Spent Nuclear Fuel (SNF) and Radioactive Waste (RW), is performed on basis of the RF legislation forming a part of the entire system of ensurance and regulation of nuclear and radiation safety in Russia. As a whole, the hierarchy of standard and legal documents in Russia is similar to standard structures functioning in the developed countries and consists of:

– RF Constitution;
– RF Federal Laws;
– standard and legal acts of RF President and Government;
– federal standards and rules concerning the use of atomic energy;
– safety guides;

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 263–269.

– standard documents of managerial bodies controlling the use of atomic energy and of other federal executive authorities (standards, building codes, etc).

System and Types of Gosatomnadzor's Standard Documentation

The Law #170-FL "On the use of atomic energy" of November 21, 1995 declares establishment of federal standards and rules on safe use of atomic energy which observance is mandatory when performing any activity related to the use of atomic energy.

The RF Government approves the list of such federal standards and rules, as well as the related amendments and supplements. The order of development and approval of federal standards and rules is also approved by the RF Government.

The Federal Standards and Rules (FSR) form the upper regulatory level in the Gosatomnadzor's system of standard documentation.

Safety Guides (SGs) forming the following – inferior – level of the Gosatomnadzor's standard documentation are aimed at providing insight into possible solution of technical tasks. SGs do not except alternative, if legally acceptable, solutions. Thus SGs should be considered as "soft" regulatory documents.

Management Directives (MDs) represent the bottom hierarchic level in the Gosatomnadzor's system of standard documentation. MDs determine the procedure of Gosatomnadzor's interfaces with the license applicant and the licensee for one or another type of activity and regulate the activities of Gosatomnadzor's structural subdivisions and its specialists.

Procedure of Developing Gosatomnadzor's Standard Documentation

The procedure of standard documentation development is determined by the RF Government Decree #1511 of December 01, 1997 "On Approval of the Regulations on Development and Approval of Federal Standards and Rules Concerning the Use of Atomic Energy and of the List of Federal Standards and Rules in the Area of Use of Atomic Energy" and detailed in the management directives MD-03-22-98 "Regulations on the Order of Examination, Preparing Conclusions, Agreement and Approval by Gosatomnadzor of Standard Documents Related to the Use of Atomic Energy" and MD-03-23-98 "Regulations on the Order of Developing Federal Standards and Rules Related to the Use of Atomic Energy to Be Approved by Gosatomnadzor". The main phases of standard documentation development are:

– Development of the Terms of Reference (ToR);
– ToR approval;
– Development of the first draft of document;
– The first draft issue for reviewing;
– Drawing up of a summary of reviews;

- Conciliatory meeting to discuss summary of the first draft reviews;
- Development of the second draft of document, etc. – altogether over 20 phases, which terminate after approval of the developed FSR, by drawing up of a draft of RF Government decree on introducing amendments into the actual list of federal standards and rules.

Gosatomnadzor entrusted the development of standard documentation to its Research and Technical Center for Nuclear and Radiation Safety (RTC NRS) performing technical support of the regulatory authority.

Results, Problems and Prospects of Standard Documentation Development

During 1994-2003 were developed:

- FSR – 63
- SG – 26
- MD – 27

including standard documents concerning floating Nuclear Power Installations (NPIs) and their related supporting infrastructure:

- FSR – 4
- SG – 4
- MD – 7.

In 2004 some FSR, which more detailed elaboration had been justified previously, were developed further.

On July 01, 2003 new Federal Law (FL) "On the Technical Regulation" came into effect regulating the interfaces in the course of:

- development, acceptance, application and fulfillment of **mandatory** requirements to production, processes of production, running, storage, shipment, sale and disposal;
- development, acceptance, application and fulfillment of the requirements to production, processes of production, running, storage, shipment, sale and disposal, as well as to execution of works/rendering services **on a voluntary basis**; and
- conformance evaluation**.**

This FL also determines the rights and obligations of the parties which interfaces are regulated by the FL.

As said in Clause 3 of Article 4 of the FL, Russian federal executive authorities have a right to issue only **acts of recommendation** in the technical regulation area**,** safe for the cases specified by Article 5 of the FL.

According to Article 7 of the FL, in compliance with the **minimum-necessary requirements** for safety, including: -safety of emissions; and -nuclear and radiation safety, the relevant technical regulations are determined and established.

These technical regulations are to be developed and implemented during the coming seven years. Before their coming into effect previous Gosatomnadzor's standard and technical documents (FSR, SG and MD) are to be in force.

From the FL "On the Technical Regulation" it follows that the previous federal standards and rules establishing safety requirements are to be revised, some safety requirements being introduced into technical regulations, the others being transformed into recommendations and introduced into the safety guides.

The new hierarchy of standard documentation on nuclear and radiation safety may be represented as follows:

- general technical regulation on nuclear and radiation safety;
- special technical regulations;
- safety guides and conditions of license validity; and
- national standards.

The main results of preparatory work on development of technical regulations are summarized below:

1. A list of technical regulations to be immediately developed has been established.
2. ToRs and concepts have been developed.
3. An analysis of the Russian laws and standard acts in force has been performed.
4. A working group has been established comprising specialists of Gosatomnadzor, Rosatom and RF Ministry for Health.
5. Mandatory requirements for development of technical regulations have been determined.
6. Prototypes of technical regulations have been prepared.

Development of Standard Documents Related to Safe Management of Damaged SNF

In real practice one often has to cope with the situations requiring special regulatory documents. As such example, let us consider management of non-standard SNF stored at Floating Service Vessel (FSV) "*Lepse*" owned by the Murmansk Shipping Company (MSC).

In 1962 FSV "*Lepse*" was reequipped into a maintenance vessel used in operations on reloading of nuclear reactors of the Russian icebreaker fleet.

FSV "*Lepse*" has: storage for Spent Fuel Assemblies (SFAs), tanks for Liquid Radioactive Waste (LRW) and working area for process operations with reactor equipment. Since 1981 FSV "*Lepse*" has been only used for storage of SFAs, RW, fittings and gears.

SFA storage actually housing 639 SFAs is the major source of nuclear and radiation hazard at the FSV. As the result of protracted (over 40 years) storage some SFAs have became partially deformed giving presently no way of their extracting using standard technologies.

The integral activity of SFAs makes up 2.5×10^{16} Bq (680 000 Ci). High gamma dose rates are recorded in rooms of SFA storage and in the adjacent rooms wherein surface radionuclide contamination varies within $(0.25–8.33)\times10^{8}$ Bq/m^2 or $(1.5–50)\times10^{5}$ decay/min·cm^2.

In compliance with the rules of the Russian Marine Register and taking into account potential hazard of the FSV, in 1999 "*Lepse*" was repaired and examined in dock at "Nerpa" shipyard. FSV hull wear was estimated at < 30%.

As expected, after the performed in-dock overhaul safe riding of "*Lepse*" will continue over the next 10 years until the following examination.

To reduce dose commitment for personnel, the Environmental Foundation "Bellona" (Norway) supplied the MSC with module building constructions to arrange therein - at 50-m distance from "*Lepse*" board - workplaces for the FSV watch service and specialists performing radiation-hazardous operations.

On the Norwegian Government's initiative in 1994 the problem of FSV "*Lepse*" management was included into the European Commission's plans.

In 1999–2001 a "Regulatory "*Lepse*" Project" was implemented under the international support. As the result of the project implementation, Gosatomnadzor - supported by the Norwegian Radiation Protection Authority - developed the following documents:

- Two SGs:
  - "Requirements on a Safety Analysis Report when Unloading SFAs during Implementation of FSV "*Lepse*" Complex Decommissioning Project";
  - "Requirements on a Quality Assurance Program when Unloading SFAs during Implementation of FSV "*Lepse*" Complex Decommissioning Project";

  and
- MD:
  - "Requirements on a Set and Content of Documents on Nuclear and Radiation Safety Analysis to Be Provided by the Running Entity and Organizations Performing Works and Rendering Services to the Running Organization in Order to Obtain a License of Gosatomnadzor while Implementing FSV "*Lepse*" Complex Decommissioning Project".

The above documents were developed on basis of Russian legislative and standard documentation taking into account the IAEA's recommendations and the current international practice.

SG: "REQUIREMENTS ON A SAFETY ANALYSIS REPORT WHEN UNLOADING SFAS DURING IMPLEMENTATION OF FSV “*LEPSE*” COMPLEX DECOMMISSIONING PROJECT" (SG-016-01)

The Guides comprise the Gosatomnadzor’s requirements on nuclear and radiation safety analysis report during SNF unloading from FSV “*Lepse*” storages.

Based on the data provided by such reports, Gosatomnadzor considers sufficiency of safety justifications during defueling of FSV “*Lepse*” storages to exclude violations of nuclear and radiation safety requirements and overexposure of the FSV’s personnel, population and environment under both normal and off-normal operating conditions.

Safe defueling of “*Lepse*” storages is complicated by inoperable condition of their systems and equipment, some SFAs being deformed. The requirements to special SFA-unloading equipment are also listed in the Guides.

The Guides specifies the requirements on:

– general description of the object;
– nuclear and radiation safety with consideration for the object specificity;
– systems and safety elements;
– SFA unloading installation and unloading procedures;
– safety analysis during SFA unloading;
– training of personnel;
– work management during SFA unloading; and
– emergency preparedness.

SG "REQUIREMENTS ON A QUALITY ASSURANCE (QA) PROGRAM WHEN UNLOADING SFAS DURING IMPLEMENTATION OF FSV “*LEPSE*” COMPLEX DECOMMISSIONING PROJECT” (SG-017-01)

The Guides comprise the Gosatomnadzor’s recommendations on developing both general and special QA programs during defueling of FSV “*Lepse*” storages when implementing the FSV complex decommissioning project. The project provides for individual unloading of SFAs, including damaged ones, using special equipment.

The SG’s recommendations also allow the running entity (and the organizations performing works and rendering services to the running entity) to take into account the requirements of the in-force regulation on compulsory certification of safety-ensuring equipment and articles of home and foreign manufacture while developing their own QA programs.

The Guides comprise the requirements on:

– structure of "QA Program" for the project implementation;

- functions of the running entity (and organizations performing works and rendering services to the running entity) during “QA Program” development and their related responsibilities;
- “QA program” contents;
- “QA program” implementation during execution of the Project; and
- certification of the used equipment, articles and technologies.

The SG annexes comprise recommendations on development of all sections of the required “QA Program”.

MD: “REQUIREMENTS ON A SET AND CONTENT OF DOCUMENTS ON NUCLEAR AND RADIATION SAFETY ANALYSIS TO BE PROVIDED BY THE RUNNING ENTITY AND ORGANIZATIONS PERFORMING WORKS AND RENDERING SERVICES TO THE RUNNING ORGANIZATION ON ORDER TO OBTAIN A LICENSE OF GOSATOMNADZOR WHILE IMPLEMENTING FSV “*LEPSE*” COMPLEX DECOMMISSIONING PROJECT” (MD-06-20-2001)

The MD determines the set and the contents of documents justifying nuclear and radiation safety and covering all activities related to defueling of FSV “*Lepse*” storages as well as other works during implementation of FSV “*Lepse*” complex decommissioning project.

These Requirements are mandatory for any entity-legal person (“the applicant”) submitting a license application to Gosatomnadzor for the following activities:

- designing of SFA unloading equipment;
- making of SFA unloading equipment;
- construction of a pad for temporary storage of nuclear materials (transportation casks (TUK) with SFAs);
- running of the pad for temporary storage of nuclear materials and nuclear material handling (TUK with SFAs);
- SFA management during transportation (SFA unloading from FSV “*Lepse*”); and
- RW management during transportation (RW unloading from FSV “*Lepse*”).

Proposals on Cooperation for Development of Standard Documents

RTC NRC of Gosatomnadzor of Russia has collected a wide experience of developing standard documents (including international ones) and is ready for participation in works on standard and regulatory support of nuclear and radiation safety issues under different projects, including non-standard ones, wherein possible solutions of technical tasks put by have not been supported yet by appropriate standard documents.

LONG-TERM SAFE STORAGE OF SPENT NUCLEAR FUEL FROM SHIP POWER UNITS IN UNDERGROUND STORAGE FACILITY IN THE NORTH-WEST REGION OF RUSSIA

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Abstract

The paper provides an assessment of the problem, which presents the management of spent nuclear fuel (SNF) from ship nuclear power units (NPU) in the north-west region of Russia and suggested is a concept of solving the problem. The concept is based on the utilization of the regional underground storage facility for long-term storage of defective, damaged and other kinds of SNF that can not be reprocessed.

Presented are the results of the study of operational safety of such SNF storage, including the nuclear and radiation safety, removal of residual heat and protective properties of the rock massif in an emergency scenario.

1. Introduction

In the variety of problems, related to the radiation safety of the north-west region of Russia and the neighboring countries (Norway, Finland and Sweden) a special place is taken by the ship NPU SNF management problem. In order to give the reader some insight into the importance of the considered problem, we just note, that the overall amount of SNF accumulated at special facilities of the Navy and ice-breaker fleet in Murmansk and Arkhangelsk regions totals to over 200 reactor cores (RC), including up to 20% of the problem type fuel: defective, damaged as well as uranium-zirconium alloy-based and uranium-beryllium alloy-based spent fuels, which Russian industry currently has neither potential nor adequate techniques to reprocess.

For solving the problem of management of those SNF types in the present day situation, the Mining Institute of the Kola Science Centre of the Russian Academy of Sciences has put forward a new SNF management concept, based on building and utilization of a regional underground storage facility, which could receive the problem

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types of SF for safe long-term (up to 70 years) storage. The basic aspects of the concept have been presented in the corresponding sections of this paper.

2. The concept of management of non-processible SNF from ship NPU in Russia's European North

Among the problems, related to the management of SNF from ship NPUs in the north of Russia's European area, an ever-increasing importance is that of the problem of SNF long-term storage, which is still not admitted for reprocessing by the industry, mainly, due to the absence of required engineering facilities at "Mayak" industrial enterprise (IE). According to the data of waste future arisings assessments made by the Russian and foreign experts [1, 2] the amount of non-processible (NP) SNF to be accumulated in the region by 2020, may be equal to over 40 RC of transport reactor facilities. Depending on the physical condition, the contents of fuel composition and features of its management, the NP SNF can be divided into several groups:

- spent fuel assemblies (SFA), containing fuel composition based on uranium-zirconium alloy. This type of fuel has been used in reactors of nuclear-powered vessels of the civil/merchant fleet. At the moment, all those SFAs are being stored aboard "Lotta" floating maintenance base (FMB), in canisters up to 5 SFA in each. The total number of this SNF is equivalent to 13 RC of civil nuclear-powered ships [1];
- SFAs, the composition of which includes intermetallic compound UBe13 dispersed in beryllium matrix. This type of fuel has been used in reactors of nuclear-powered submarines (NPS) with liquid-metal coolant (LMC). Currently, the largest part of uranium-beryllium fuel is being stored at the coastal maintenance base (CMB) in Gremikha village in the spent removable parts (SRP). Each of SRP is a block structure that includes a RC with the plunged control rods, a lateral beryllium reflector and the upper plug of biological shield. A certain amount of SNF of that type is kept in decommissioned settled NP submarine reactors' cores. After SNF of all NPSs is unloaded, there will arise at least 9 RC of LMC reactors at the CMB;
- SFAs with non-processible fuel composition, which will be unloaded from reactors of special deep-diving NPSs of the Northern Navy. Presumably, the total amount of this SNF is equivalent to 9 RC of such reactors [1];
- defective SNF. This type of SNF includes SFAs that had structural damage (swelling, contortion, partial loss of containment property etc.) in the process of NPU operation, during long-term storage or while carrying out transportation and technological operations with SNF. Independently of the reactor type and design features of SFAs, defective assemblies are not allowed for reprocessing by the industry. At present, defective SFAs of the civil fleet are stored at "Lotta" and "Lepse" FMB. All SFAs of that type are stored at storage facilities for SNF, which are located at the CMB in Andreyeva Bay and Gremikha village. The total number of such SNF can be estimated to be approximately 2 RC of NPS reactors [1]. Taking into consideration the SNF storage conditions at the CMB, one may assume that the quantity of defective SNF at the CMB may exceed the above number. So,

- for instance, the paper [2] notes, that according to data of the Contact Expert Group of IAEA, the number of damaged canisters, stored at the CMB in Andreyeva Bay amounts to 300. The damage to canisters, used as containers for SNF, may result in damaging SFAs, the number of which may be roughly equal to 9 RC of NPS reactors;
- SNF from damaged NPS reactors. This kind of SNF includes cores of reactors damaged as a result of accidents or undue operation. By the end of 1990-ies there were two damaged RC of NPS reactors in the Northern Navy [1].

It should be noted that almost all NP SNF and a considerable part of reprocessible fuel are currently stored at facilities of the Northern Navy and Murmansk Shipping Company. Most of them are either in emergency condition or do not meet up-to-date safety requirements in the context of the long-term storage of nuclear and radioactive material.

During the last years, the Minatom of Russia has been making considerable efforts, aimed at improvement of the SNF management practices and, in particular, at securing the safety of SNF storage. Aimed at this, a concept of SNF management has been developed, which implies storing SNF in containers for up to 50 years, until the necessary facilities are built. This concept is based on utilization of dual-purpose metal-concrete containers (MBK), meant for transportation, storage and/or disposal of the spent fuel from transport reactors. According to the concept, it is proposed to build open storage sites for temporary storage of SNF containers at some specialized enterprises of the region [3].

According to Russia's strategy, all SNF (including NP SNF) should be taken away from the region. At the same time, limited capabilities of the Russian Minatom and its dependence on the foreign financial aid stipulate some uncertainty as to the terms of implementation of the SNF management concept in the region, including the problem of SNF long-term storage. The current state of SNF management infrastructure requires to solve, first of all, such top-priority tasks as SNF unloading from decommissioned NPS and transportation of the regional SNF for reprocessing. These problems can be solved during the nearest 15-20 years, provided that there is necessary technical-organizational and financial support. The biggest uncertainty is related to the NP SNF. At present, the crucial question is that of choosing the approaches and method of utilization of such SNF: reprocessing or disposal [4].

To satisfy the existing requirements for ecological safety in the field of management of the exposed fuel it seems expedient to consider a variant of NP SNF long-term storage in an underground storage facility, placed in geological formations of the region. In the report delivered by the IAEA Contact Expert Group [3] the storage of ship SNF in special facilities is considered as a possible variant under certain development of the SNF management scenario in Russia.

In general, the storing of NP SNF underground may be carried out according to different variants, which are considered in detail in the section, dealing with structural and lay-out diagrams of the underground storage. We should note briefly, that the considered variants envisage the dry method of SNF storage. According to one variant it is proposed to place SNF mainly in containers, which, to some extent is the development of the concept of container storage. In such case, there might be needed about 210 containers of MBK-40 type, each containing 7 canisters with SFAs. Additionally, there must be envisaged the storage of LMC reactors SRPs amounting to 9 pieces. According to another variant, it was planned to store SFAs in canisters, which could be placed in special built-in structures of the underground storage facility. Conceptually, this variant corresponds to the pattern of SNF dry storage in separate canisters (boxes). According to this variant, there might be needed to place around 1900 canisters with SFAs in the underground module of the storage facility.

The idea to use specific properties of stable geological formations is the basis of conceptual approaches, applied in Russian and international practices to comply with the guaranteed safety level of radiation hazardous facilities [5, 6, 7 etc.]. Among them, in particular, there are SNF and radioactive waste (RW) storage facilities, for which it is typical to contain considerable amount of long-lived radiotoxic materials. The results of numerous investigations carried out in various countries show, that placement of such facilities in underground storages allows to ensure both high level of their protection against outside natural or technogenic impacts and environment and population protection under various, even hardly probable, inside accidents.

It should be noted, that the problem of isolation of the long-lived RW, accumulated in the region, may become in the future one of the most significant problems of ensuring radiation safety. So, the paper [7] shows, that during operation time and decommissioning of the reactor facility of the Kola NPP and nuclear-powered ships there may be accumulated over 60,000 t of long-lived radioactive wastes in the region. The forecasted radiological feature of such wastes stipulates the objective necessity of their long-term isolation from the biosphere in an underground storage facility, placed in stable geological formations at the depth of 100 m and more. So, building such an underground SNF storage may be regarded as a component of the complex problem of improving the regional environmental and population radiation safety, by means of having advantage of the unique properties of regional geological formations.

3. Study of radionuclide composition of SNF from ship NPU

A significant decay cooling time (10 years and longer) is typical for NP SNF accumulated in the region and, thus, fission fragments, with less than 1 year period, have now been decomposed down to an unimportant level of activity. At present, safety of SNF management at different stages are determined by the long-lived radionuclides.

The calculation method of SNF isotope composition is based on application of reactors' computer codes RITM and KRATER, which suggest taking into consideration the burnup fuel [8, 9]. They both describe the change in time of nuclei concentration from the composition of the initial loading (^{234}U, ^{235}U and ^{238}U) and determine the content in fuel of another 22 long-lived actinides and fission fragments, playing an important role in assessments of safety of SNF storage: ^{79}Se, ^{85}Kr, ^{90}Sr, ^{99}Tc, ^{129}I, ^{137}Cs, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}Cm etc.

Reactor cores of ship reactors being of small size are noted for the considerable degree of the nuclear fuel burnup, the large reserve of activity at the start of campaign, which requires taking into consideration the influence of neutrons on the flow and their spectral distribution (thus, on the isotope composition of the fuel) of neutron absorbing materials, that are used for activity compensation and spatial profiling of the capacity. It is necessary as well to consider the applied profiling of energy release through uneven distribution of fuel by the RC volume. So, to determine the isotope composition of ship reactor SNF seems to be the most complex neutron-physical task. In order to solve it, the NPU reactor data have been systemized, followed by development of simplified, though reliable (robust), mathematical models [10].

Robust models of ship NPU reactors use a cylinder model for geometric representation of reactor cores with different nuclear-physical properties. The heterogeneous structure of RC layers is considered using submodels of a reactor cell (fuel, cladding, moderator, control rods (CR)) and microcells (the burnup absorber (BA) of neutrons).

To find the density of neutron flow in the fuel, information is required on: the capacity, energy output, and operation mode of RC; their dimensions; material composition of RC and the reflector; the geometry and number of fuel rods and fuel assemblies (FA); the BA and CR; the coolant type and its thermodynamic parameters.

For mathematical description of fuel burnup and fission products' accumulation using KRATER software package three basic types of NPU are chosen:

– NPUs of OK-900 type with uranium-zirconium fuel in thermal-neutron pressurized water tank reactors;
– NPUs with intermediate reactors, Pb-Bi eutectic cooled and
– NPUs of the 1st and 2nd generation NPS.

The basic physical and engineering characteristics of ship NPU reactors, used as original data for building robust models, that is: nominal and operation capacity (N_t, N_{op}), energy output ε, fuel and coolant types etc., have been found on the basis of the analysis of the published data [10] (see Table 1).

The same table displays other parameters of some RC, being parameters of robust models and identified according to reactor programs. In Table 1, models' parameters are marked with ⊕. These are RC dimensions, uranium load and enrichment, the number of SFAs, methods of profiling the capacity and compensation of reactivity etc.

A number of important parameters, such as: uranium-235 RC loading, which provides the set energy output of RC; RC dimensions, type and characteristics of heterogeneity etc. have been estimated according to intermediate multiversion neutron-physical calculations of reactors under the known values of N_{op}, ε and RC lifetime $T_k = \varepsilon / N_{op}$.

The results of investigations of fuel cycles in robust models of ship NPU reactors are illustrated in Fig.1, 2 and presented in Table 2 by the mass of basic actinides and fission fragments. Besides, Table 2 presents activities of ^{90}Sr and ^{137}Cs, determining the radiation potential of SNF.

A number of nuclei, important in SNF management are not taken into consideration by the KRATER program. In particular, these are spontaneously fissionable ^{244}Cm, gamma-emitters ^{134}Cs, ^{154}Eu, α-active ^{238}Pu. These nuclei were identified by depletion cycle equations' solution algorithm and it has been found, that in RC of OK-900A NPU reactor there are accumulated by the end of campaign ^{244}Cm – 11.7 g, ^{134}Cs – 344 g, ^{154}Eu - 69 g, ^{238}Pu – 400 g.

Basing on the results of depletion cycle calculations it is anticipated, that:

- the group of defective SF of the 1st and 2nd generation NPS water-moderated reactors may contain 170 kg of ^{235}U and 6.3 kg of ^{239}Pu. The total activity of defective SNF for 2010 is 0.27 mln. Ci;
- the group of non-processible uranium-beryllium fuel contained in 9 spent removable parts may contain up to 1,100 kg of ^{235}U and 13 kg ^{239}Pu. The total activity of ^{90}Sr, ^{137}Cs and daughter products of their decay makes 1.41 and 1.09 mln. Ci for the years 2000 and 2010 respectively;
- the group of non-processible SNF on the basis of uranium-zirconium alloy, formed as a result of operation of OK-900 and OK-900A NPU reactors of nuclear-powered ice-breakers from 1970 to 2001 and stored on board "Lotta" FMB, includes up to 750 kg ^{225}U and about 30 kg of ^{239}Pu, whereas the accumulated ^{90}Sr, ^{137}Cs together with ^{90}Y and ^{137m}Ba have activity of 9.67 and 7.65 mln. Ci for the years 2000 and 2010 respectively.

TABLE 1. Table 1 Basic parameters of RC in robust models of ship NPU reactors

Parameters	**Nuclear-powered ship, NPU type**				
	1-st generation NPS, BM-A	2-d generation NPS, BM-4	NPS with LMC, 705 & 705K projects	"Arctica", "Russia" etc. Ice-breakers	
				OK-900	OK-900A
N_{op}, MW	21	26.7	50	65	75
ε, GW · day	12.5	17	24-30⊕	84.6	97.3
Dimensions of RC D×H, cm ⊕	83×90	98×90	76×75	114×100	118×95
Nuclear fuel	UO_2+Al	UO_2+Al	UBe_{13} intermetallide	U-Zr alloy	U-Zr alloy
Fuel rods, mm ⊕	6.1×0.45	6.1×0.45	12×0.6	5.7×0.75	5.7×0.75
Uranium load, kg/ load of ^{235}U, kg	262	350	185 ⊕	370/150 ⊕	333/200
Enrichment by ^{235}U,%	21	20	89 ⊕	36/45⊕	45/75⊕
Number of SFA (or) fuel rods	180	250	(2004) ⊕	241	241
BA mass, g ⊕; homogeneous (heterogeneous)	85	120	—	133 (930)	150 (1150)
Reflector	Steel-water	Steel-water	Beryllium	Steel-water	Steel-water

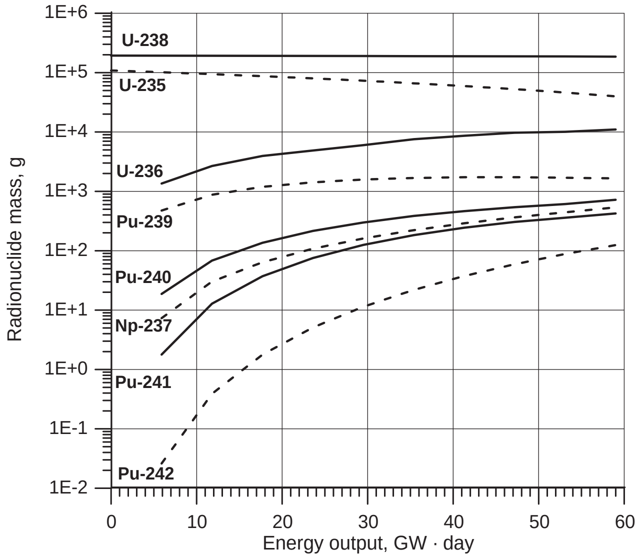

Figure 1. The burnup and generation of actinides in the robust model of a NPU reactor of OK-900 type, with the uranium loaded into the RC 302 kg, enrichment by ^{235}U 36% and U-Zr fuel.

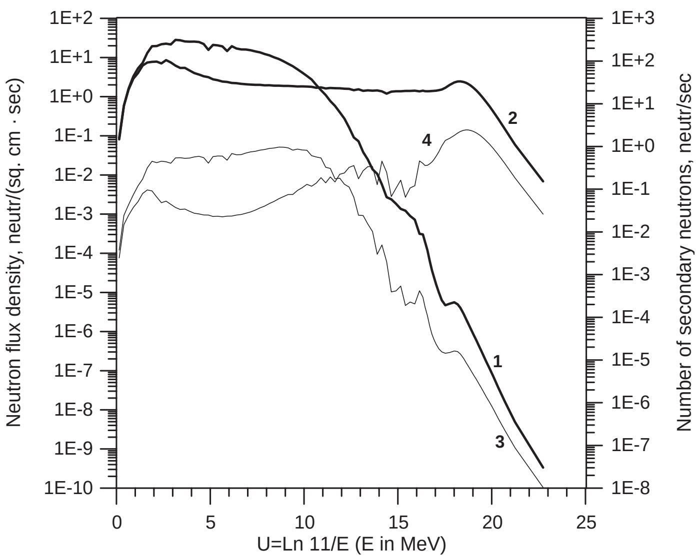

Figure 2. The spectrum of neutrons flux (1,2) and the number of secondary neutrons (3, 4) in reactor lattice of the RC of ship NPU reactors, RITM code. 1, 3 – NPS of 705 project; 2, 4 - NPU of the OK-900A.

The total nuclear potential of non-processible SNF is estimated at 2,300 kg of the mass of ^{235}U and 60 kg of ^{239}Pu, whereas radionuclides ^{90}Sr and ^{137}Cs along with products of their decay are characterized by the activity amounting to approximately 9.5 mln. Ci for 2010.

TABLE 2. Isotope composition of SNF from NPU by the end of lifetime with different energy output of the RC [10]

Radionuclide	**NPU type, energy output, GW · day**						
	OK-900	OK-900	OK-900A	OK-900A	NPS I	NPS II	NPS-705 [11]
	38	45.3	78	97	12.5	17	
	Actinoids and fission fragments masses, kg						
^{235}U	42.38	50.36	57.59	82	40.03	49.77	123.6
^{236}U	9.27	12.25	16.39	22.27	2.78	3.76	11.56
^{237}Np	0.429	0.638	0.948	1.36	0.074	0.114	0.93
^{238}U	164.8	205.2	209.21	125.8	204.4	259.8	16.56
^{239}Pu	1.80	2.268	2.429	2.158	1.424	1.934	1.77
^{240}Pu	0.601	0.785	0.914	0.729	0.226	0.324	0.1115
^{241}Pu	0.377	0.547	0.692	0.668	0.098	0.157	0.0252
^{242}Pu	0.0945	0.153	0.238	0.210	0.00793	0.0154	0.0104
^{241}Am	0.00868	0.0168	0.0245	0.0256	0.0020	0.0033	0.035
^{85}Kr	0.0428	0.0542	0.0704	0.0887	0.0151	0.0204	0.01145
^{90}Sr	0.895	1.265	1.662	2.107	0.269	0.364	0.413
^{99}Tc	1.124	1.445	1.881	2.329	0.321	0.4339	0.746
^{137}Cs	1.659	2.144	2.822	3.521	0.458	0.6226	0.684
^{151}Sm	0.00921	0.0114	0.0133	0.0186	0.00635	0.00822	0.0412
Activity of ^{90}Sr, ^{137}Cs, Ci							
^{90}Sr	$1.369 \cdot 10^5$	$1.758 \cdot 10^5$	$2.309 \cdot 10^5$	$2.92 \cdot 10^5$	$3.80 \cdot 10^4$	$5.14 \cdot 10^4$	$5.76 \cdot 10^4$
^{137}Cs	$1.437 \cdot 10^5$	$1.857 \cdot 10^5$	$2.444 \cdot 10^5$	$3.06 \cdot 10^5$	$3.97 \cdot 10^4$	$5.39 \cdot 10^4$	$5.91 \cdot 10^4$

4. Assessments of the safety of SNF storage

According to Russian normative documents and recommendations by the IAEA, when developing the project and scientific evaluation study of SNF storage facility safety the following issues, being the main ones, should be considered:

– residual heat removal;
– ensuring the subcriticality of the storage system;
– providing the radiation protection for staff and population.

4.1. RESIDUAL HEAT REMOVAL

One of the main factors, that determine the thermal regime of the storage facility and the efficiency of the system of residual heat removal, is the value of residual heat rate (RHR) of SNF. Characteristic of ship NPUs is the great variety of isotope compositions, which depend on the burnup degree and the fuel cooling time. That is why, along with the above factors, an integral value of residual heat in the storage is stipulated by the quantity of SNF of various types, sent in for long-term storage. Based on the results of ship NPU reactor cores' RHE calculations it was found, that the integral value of residual heat at the storage may amount to 41 kW by 2010. The various types of SNF contribute differently: uranium-zirconium fuel of ice-breaker fleet makes up 19.3 kW; SNF from NPS reactors with LMC - 4.4 kW; defective SNF of ice-breaker fleet - 4.1 kW; defective SNF of the Navy - 13.5 kW (taking into account the assumed quantity of defective SFAs at the storage facility in Andreyeva Bay).

The goal of investigations of the SNF storage thermal regime was to study the possibility of removal of residual heat by means of natural processes: thermal conduction and natural convection. It is assumed that the certain temperature of air in the underground module is supported by means of the ventilation system. These investigations also served the task of finding the optimum layout of SNF packages from the point of view of thermal safety of the storage. In calculations FFM code was used. This code had been developed by one of the authors of this paper (A.V. Naoumov) and it enables to realize numerical solution of the equation of non-stationary thermal conduction in three-dimensional geometry. FFM code was successfully applied, in particular, when modeling the heat transfer processes within the framework of scientific validation of thermal safety during long-term isolation of the Chernobyl NPP emergency unit [12].

For conditions of SNF storage in separate canisters there had been considered different variants of canisters layout in steel tubes, placed in a grid pattern inside a built-in reinforced concrete structure: single-level canister placement; double-level canister placement inside one tube; placement in one tube of one canister, with SFAs placed in double-levels in each canister. The last variant is based on the technology of separation of fuel parts of SFAs, which is regarded by Russian experts as a promising pattern of ship NPU SF management from the point of view of reduction of the storage volume [13]. Tube spacings varied between 0.5 and 1 m.

There were studied two types of SNF canisters storage: in steel tubes, placed in the air medium ("CASCADE" type) and in steel tubes, placed in a concrete body inside a built-in structure. The modeling of radioactive decay heat removal was implemented based on the initial integral value of RHR of NP SNF by 2010.

In case of "CASCADE" type storage, residual heat removal is carried out, basically, by means of natural convection and radiation heat transfer. The implemented assessments have showed that the calculated maximum temperature in the storage facility (fuel zone) did not exceed 100 ^{0}C with the tube spacings being 50 cm. At the same time, the temperature of the interior surface of the built-in structure is 45-50^0 C. The temperature of the basic bulk of materials in the storage, as estimated, does not exceed 40 ^{0}C.

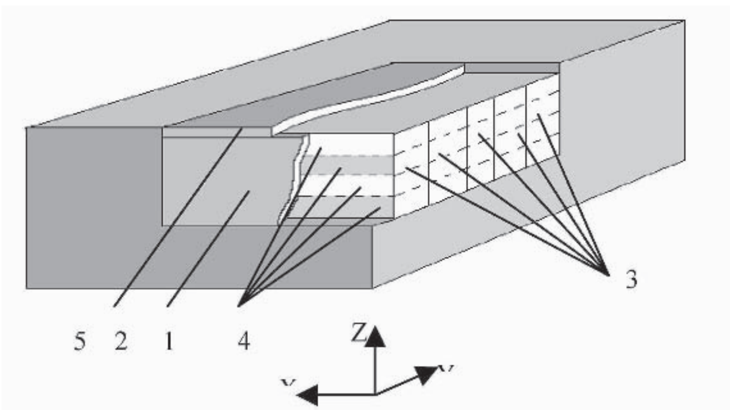

Figure 3. A model representation of the underground module to be used as a storage of canisters in tubes of concrete body.

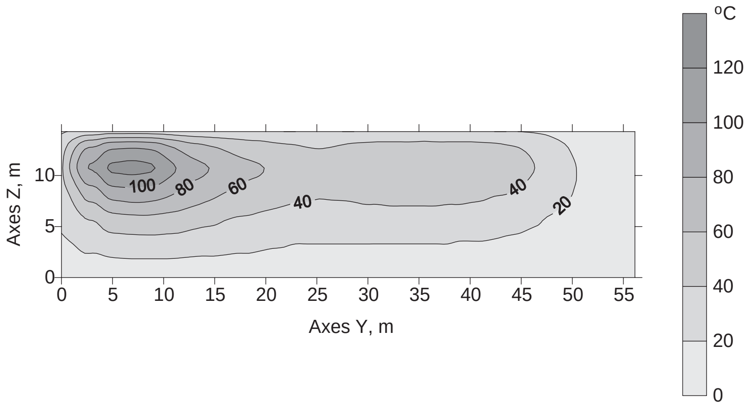

Figure 4. Temperature field in the plane X = 0 two years later (T_{max}=126 ^{0}C)

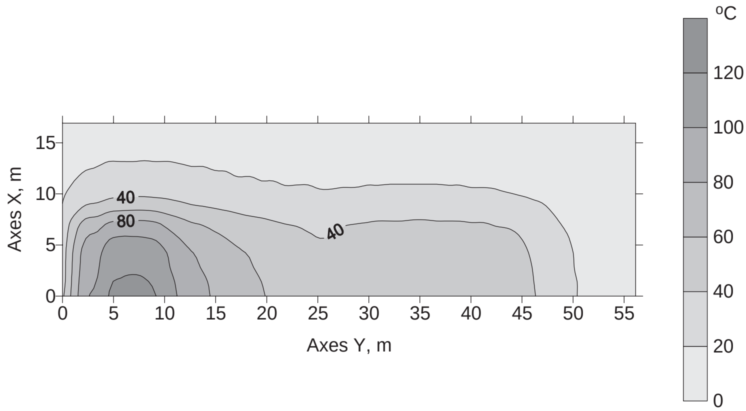

Figure 5. Temperature field in the plane Z=10.6 m two years later (T_{max} = 126 ^{0}C)

In more detail, the results of thermal calculations are presented for the variant of canisters storage inside a concrete body, for which less intensive heat removal is typical. The main mechanism of heat dissipation is thermal conduction of the concrete body and rock. Fig.3 presents a calculated model, that includes a built-in structure, formed by lateral walls (1) and upper shield (2); zones (sections) of placement of SNF different types (3), each of which is divided in layers (4), modeling fuel and end parts of SFAs at their double-level placement in canisters. The built-in structure is from three sides (X+, Y+, Z-) surrounded by the rock massif (5). Geometrically, this model is characterized by the following parameters: height 14.3; width (directed along X axis) approximately 17 m; length (directed along Y axis) approximately 56 m. In all directions 10 m thick peripheral areas are within the rock massif. The model used the condition of symmetry, to which boundary condition of the second kind corresponds with X=0. At the upper boundary (Z+) and lateral surface (Y+0) a condition of the third kind is used with ambient air temperature 15 ^{0}C. At other boundaries of the calculated area the condition of the first kind is used with the temperature of surrounding massif being 7 ^{0}C.

As a result of the calculations, it was found that when tubes are placed in a concrete body with 50 cm spacings the maximum predictable temperature in the storage (inside canisters and the adjoining concrete body layers) may reach 200 ^{0}C. Fig. 4 and 5 show the results of calculations for the variant with 72 cm spacings. In those calculations the following values of the heat transfer coefficient were used: $\alpha_{Y=0} = 3$ W/(м^2·K); $\alpha_{Z+} = 6$ W/(м^2·K). In this variant the following order of SNF placement in the module zones (sections) was accepted: uranium-zirconium SNF of the ice-breaker fleet, defective SNF of ice-breaker fleet, SNF of NPS reactors with LMC, defective SNF of the Navy. Fig. 4 and 5 illustrate temperature fields in cross-sections, to which maximum temperature values (126 ^{0}C) correspond after 2 years of storing SNF. One can see from the figures, that the heat patch with the temperature of 100 ^{0}C and higher is localized within a limited area of the storage facility. The maximum temperature within a local area of the interior surface of the built-in structure does not exceed 60 ^{0}C. The outside surface of the structure is heated up to 30-40 ^{0}C within a limited area. At the same time the thermal condition of the largest part of the materials is characterized by the temperature below 60 ^{0}C.

While making a short comment of the results of the accomplished calculations, we can note, that within different variants of SNF canisters placement inside the tubes of the concrete body, the predictable duration of the heating stage is 600-700 days. The maximum temperature in the storage and the thermal condition of materials and structures of the underground module depend significantly on the tube spacing. The calculations have showed that favorable thermal regime for the storage facility may be achieved with the spacing of about 70 cm.

When stored in containers, the residual heat removal is done, mainly, by means of the natural convection and radiation heat transfer. To find the maximum temperature of the container materials there was carried out the heat transfer process modeling inside a single container in three dimension geometry. Calculations were done using FFM code. At the same time, the heat transfer through emission inside the container was taken into consideration by the value of the efficient thermal conductivity. It was assumed in calculations, that there were 70 SFAs (with double-level placement of SFAs in canisters) loaded in the container, the total residual heat from which made about 770 W. The calculation results showed that the thermal condition of the container is characterized by the following parameters: SFAs canister zone maximum temperature being 80 ^{0}C; the maximum temperature of the container outside surface - around 50 ^{0}C; the heating time made approximately 60 days.

On the whole, the results of the accomplished calculations enable us to conclude, that the removal of residual heat can be ensured, while no application of the forced cooling of SNF packages is needed.

4.2. NUCLEAR SAFETY ISSUES

Under SNF storage in MBK-40 containers, which have the shield of heavy concrete 0.4 m thick, it is the least probable (<0.01) that neutrons, generated in the fuel (under spontaneous fission of nuclei $^{242,\ 244}$Cm, ^{238,240}Pu) could escape from the container. So, neutron multiplying is totally determined by the properties of a single container. MBK-40 container had been designed for both transportation and long-term storage of SNF meeting the prescribed requirements for nuclear safety. On the contrary, the largest part of neutrons, generated in canisters, can escape from them and depending on the medium in which canisters are stored (water, air, concrete), the neutron multiplying of various scale may take place within the system of canister storage. This fact is determined by both the small transversal canister size and the small thickness of canister shell and it can clear up the interest, shown in studying nuclear safety with regard to canister type of SNF storage.

A dry type storage facility can be presented as a structure in an underground shaft, where, over a periodical lattice there are placed, so called, storage pits (SP). In each SP, one or several airtight SNF canisters are placed vertically. In our case, a storage pit is an airtight steel tube with ∅=230-240 mm, in which either 2 canisters of the currently used design, are place vertically or one modernized canister, however, with two fuel parts, that fit its height.

Different type SNF storage is implemented in storage sections, meant for the corresponding type of SP to be placed. For storage of 1,900 SNF canisters, 950 SP are needed.

According to Russian standards, the nuclear safety (NS) should be ensured by the limiting of the spacings between SPs (a) in storage sections. At the same time, effective neutron multiplication factor should not exceed $K_{eff} = 0.95$. When assessing the NS, one has to take into consideration as well the consequences of natural and man-made impacts, such as floods, explosive shock wave etc. Hereby, we consider the ways of ensuring the safety in two types of SNF dry storage: the first one - when SPs are placed in a concrete body and the second in the air medium inside a built-in concrete structure ("CASCADE" type). The main parameters, which determine the SP storage spacings are the amount, isotope composition and enrichment of SNF (see Table 3).

The most persuasive proofs of NS storage can be obtained with a conservative approach, when the real lattice of SP is replaced with an infinite one, while such highly neutron absorbing fission products as ^{154}Eu, ^{151}Sm, ^{157}Gd etc. are not taken into consideration in the isotope composition.

Neutron multiplying parameters for large SP placement spacing had been studied using "RITM" reactor code, in which the neutron transfer equation is solved by the method of successive collisions. The composition of the canister (R=11 cm) fuel zone is represented by the uranium-zirconium SNF with enrichment of 56.7% by ^{235}U.

From the results presented in Fig.6 it follows that for normal operation conditions (dry state, solid curves 1 and 2) a significant subcriticality is ensured at any spacing values. The largest value $K_{inf} = 0.88$ is observed for "CASCADE" type storage. The breeding properties of the medium in the "Concrete body" type storage facility vary considerably depending on the distance between SPs and, for instance, with $a \geq 50$ cm, high subcriticality $K_{inf} \leq 0.42$ is ensured.

TABLE 3. Content of fissile isotopes of SNF in NPU robust models (in average by RC)

SNF type	Type of NPU	Content of fissile isotopes, g					Enrichment, %
		^{235}U	^{236}U	^{238}U	^{239}Pu	^{241}Pu	
Defective	VM/A	156	15.4	794	5.5	0.54	16.1
Defective	VM-4	140	20.9	743	5.4	0.63	15.5
U-Zr alloy	OK-900A	340	92.4	522	9.0	2.8	35.6
U-Zr alloy*	OK-900A	550	91.2	328	8.3	2.0	56.7
UBe_{13}	Project 705	534	46.0	73.6	4.0÷6.0	0.20	81.7

* from the high enrichment zone

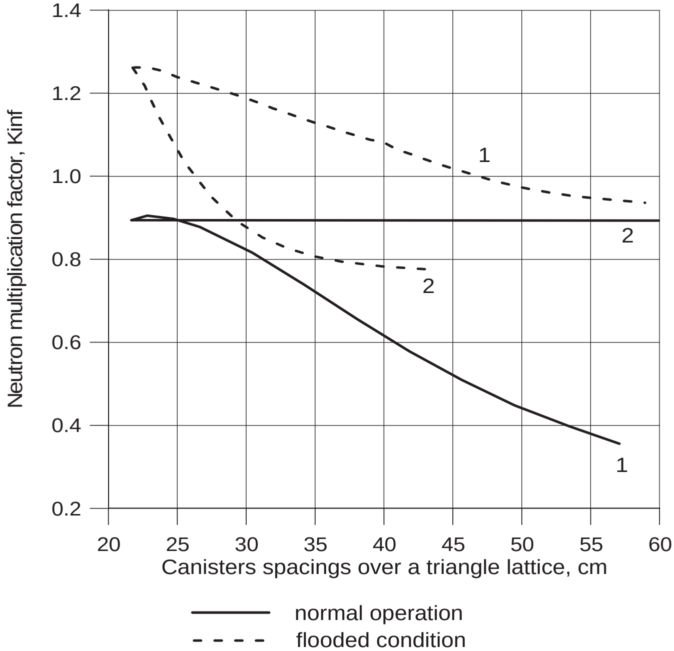

Figure 6. Characteristics of nuclear safety, when storing SF canisters in steel tubes, placed in the module of underground storage: 1 - inside a concrete body; 2 - in the air

For emergency situations, involving the flooding of the underground SNF storage facility, and the seal failure, with canisters flooded, subcriticality is achieved only for a certain range of SNF canister placement spacing values. So, when storing SNF in a concrete body, the safety $K_{inf} \leq 0.95$ is ensured with grid spacings a≥55 cm. When canisters are stored in the air medium the safe spacing does not exceed 28 cm (dotted curves in Fig.6). Such big difference in the density of SP placement within the storage module, which ensures the nuclear safety for the considered types of SF storage, is explained by the more efficient moderation and absorption of neutrons by water compared to concrete.

Thus, as a result of investigation there were found such SNF underground storage designs, which ensure the nuclear safety by means of the storage inherent safety properties. To conclude, we should note that SNF storage in the air medium allows for a denser and, thus, a more economical design of storage pits, whereas storage in a concrete body is characterized by higher resistance to exterior dynamic impacts, as in such case the possibility of displacement and deformation of canisters and SFAs is excluded.

4.3. RADIATION SAFETY

Dose loads on the staff from ionizing radiation impacts are one of the basic factors, determining the level of the storage radiation safety. The purpose of studies was to assess the efficiency of the built-in structure in the underground module of the storage as an engineering shielding barrier. As a criterion of radiation safety for personnel the permissible effective dose rate was considered. For standard conditions adopted by the Russian normative document NRS-99 [14] the value of permissible dose rate for personnel makes up ~ 0.01 mSv/h. As the subject of research, there has been considered a section of the built-in structure, containing tube block with canisters for uranium-zirconium SNF of OK-900A type NPU cores with the minimum cooling time of SNF being 10 years. The main sources of gamma-radiation in SF composition are radionuclides ^{134}Cs, ^{137}Cs and ^{154}Eu. As a neutron source, there was considered ^{244}Cm, which determines for over 97% of contribution into the total neutron intensity in a packing. Radiation potential of SF in the section is characterized by the total activity of gamma-emitters of about $2 \cdot 10^6$ Ci and the intensity of spontaneous fission neutron source of about $5 \cdot 10^9$ neutr/sec.

According to the results of ionizing radiation field characteristics, it was found that for "CASCADE" type canister storage variant the gamma-radiation dose rate from the mixture of fission products on the interior surface of the built-in structure constitutes about $5 \cdot 10^4$ mSv/h, whereas the dose rate from spontaneous fission neutrons made 12 mSv/h. To estimate the efficiency of the built-in structure, from the point of view of the staff protection from radiation, there was carried out a numerical modeling of the process of neutron and gamma radiation passage through the shielding of regular concrete with density of 2 g/cm^3. The results of calculations are given in Fig.7. The presented data show, that numerical experiments found the law of moderation of the dose rate as exponential with relaxation length being 13.7 g/cm^2 for gamma radiation and 24.7 g/cm^2 for neutrons of spontaneous fission.

On the whole, calculation results show that concrete has efficient protective properties compared to the considered sources of ionizing radiation. These properties allow us to recommend the thickness of the protective barrier of the built-in structure to be 1 m with the concrete density 2.2 g/cm^3.

4.4. STUDY OF PROTECTIVE PROPERTIES OF THE SURROUNDING GEOLOGICAL MEDIUM

It is assumed, that as a result of emergency impact of man-induced or natural character, an SF storage facility is not unloaded after its operation time is over, but is converted into a facility of permanent uncontrolled disposal of SNF, while the interior space is filled with concrete or bentonite. Over the years, there will start an uncontrolled ingress of water into the storage, the air, that got inside in the process of excavation and exploitation of the storage will be expelled. There are reasons to assume that in the long run (after several hundred years [15]), practically natural character of ground water flow and its geochemical parameters will be restored.

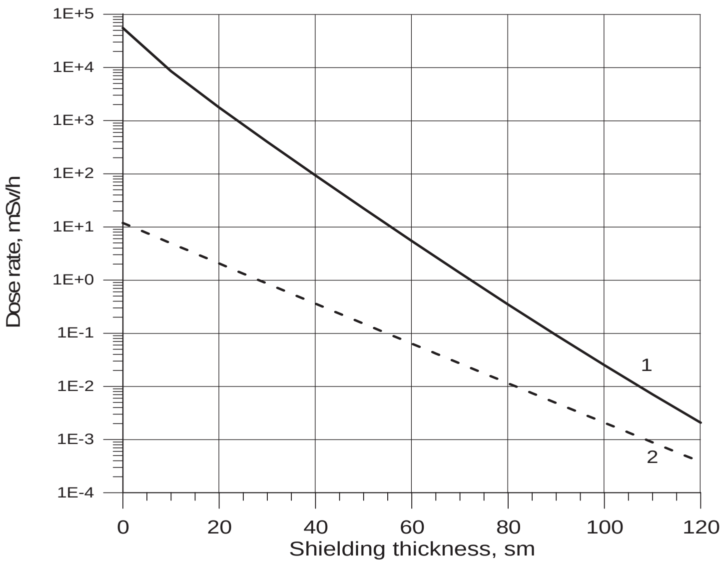

Figure 7. The attenuation of dose rate of gamma-radiation (1) and neutrons (2) by the concrete shielding.

A task had been set to study the migration of long-lived and the most toxic radionuclides (the time period is 10,000 years) in the surrounding geological medium and to find the potential of protective properties of the rock granite massif.

To prepare the initial information on geochemical parameters of radionuclides, there were analyzed reports by Swedish, Finnish, Japanese, Swiss and American experts, where they had presented investigation materials of migration parameters of radionuclides in granites. The generalization of this analysis is done in the monograph [10], where the conclusion is made about the possibility of assuming as a basis the results of Scandinavian experts' studies, because of the similar geochemistry of the geological environment.

There was studied the migration of radionuclides, first divided into three groups, according to the value of distribution coefficient ($R = 1 + \frac{K_d \rho}{\varepsilon}$; $\rho = 2{,}700$ kg/m^3 - rock density; $\varepsilon = 0.005$ - rock porosity): non-sorbing ($1 \le R < 100$), slightly sorbing ($100 \le R < 10^3$) and highly sorbing ($R \ge 10^3$) (see table 4).

All non-sorbing radionuclides are characterized by their high solubility, so for them the model of instant dissolving is true and the most long-lived one of this group - isotope ^{129}I and, additionally, isotope ^{3}H, being of interest from the point of view of tritium problem, were considered as standard migrants. Among slightly sorbing radionuclides, isotope ^{79}Se is of interest, and it has a negligible distribution coefficient in rock and a long half-life. For that isotope the model of solubility limit is true. Of the large group of radionuclides characterized by high adsorption properties, the migration of the following radionuclides was studied: of isotopes ^{90}Sr and ^{137}Cs, for which levels of solubility and activity are most high, which, however, differ by an order in their adsorption properties, and plutonium isotopes - ^{238}Pu and ^{239}Pu, which differ in their half-life and initial activity in the storage. For those istopes the model of solubility limit is true.

Table 4 offers the information on the above radionuclides, which was used in calculations and analysis of the results (half-life periods, levels of initial activity, yearly ingress limits (YIL), intervention levels (IL), migration parameters).

Hydrogeological characteristics of potential sites of the region, assumed depth of the storage placement, as well as possibilities of the PORFLOW program code [16], using which the study was accomplished, allowed us to make up a conceptual representation of the migration model.

Fig. 8 presents vertical cross-section of the migration model. The underground water flow and radioactivity spread from the storage is considered in a three-layer model. The model layers have different coefficients of conductivity and porosity. For the layer of placement of the nuclear material storage in the hydrogeological task, condition of zero flow is used [10].

TABLE 4. Parameters used in calculations and analysis of the results

Isotope (group)	**Half-life, years**	**Initial activity, Ci**	**YIL, Bq/year [14]**	**IL, Bq/kg [14]**	**Solubility, M**	**Distribution coefficient, m^3/kg**	
						concrete [10]	rock
^{129}I (1)	$1.57 \cdot 10^7$	0.21	$5.3 \cdot 10^3$	1.30	high	0	0
^3H (1)	12.33	3,000	$2.1 \cdot 10^7$	7,700	high	0	0
^{79}Se (2)	$6.5 \cdot 10^4$	15	$3.6 \cdot 10^4$	48.00	$1 \cdot 10^{-7}$	0.0003	0.0005
^{90}Sr (3)	29.1	510,000	$1.3 \cdot 10^4$	5.00	$1 \cdot 10^{-3}$	0.0018	0.005
^{137}Cs (3)	30.14	540,000	$7.7 \cdot 10^4$	11.00	high	0.0028	0.05
^{238}Pu (3)	87.74	10,500	$2.5 \cdot 10^3$	0.60	$2 \cdot 10^{-8}$	4.3	0.05
^{239}Pu (3)	$2.41 \cdot 10^4$	495	$2.4 \cdot 10^3$	0.56	$2 \cdot 10^{-8}$	4.3	0.05

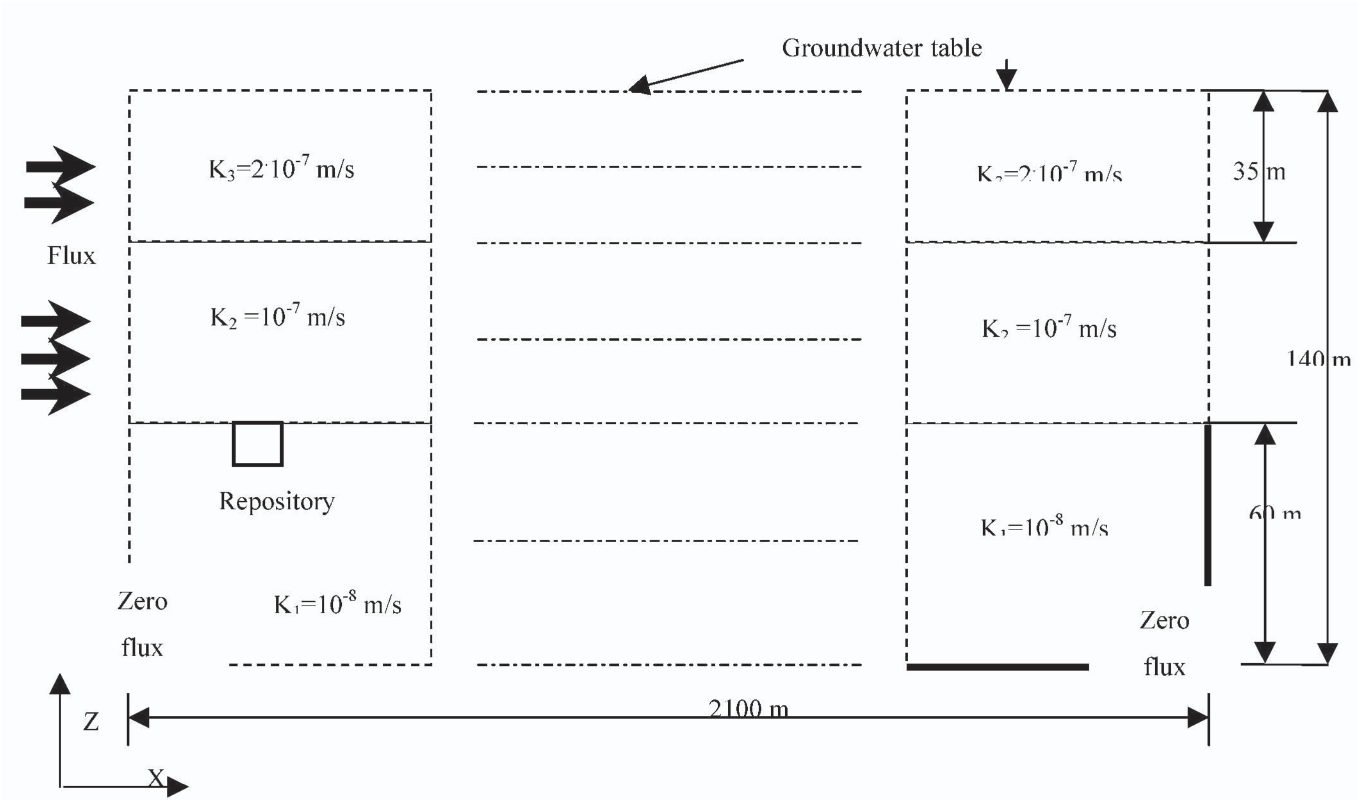

Figure 8. Vertical cross-section of migration model.

Two values of hydraulic gradient (along X axis) are considered – 0.01 and 0.05 m/m for a better assessment of both, the situation (0.01 m/m) typical for sites and a more critical one, from the point of view of the storage safety (0.05 m/m). The whole spatial field of modeling is a rectangular parallelepiped with dimensions 2,100x400x140 m along axes X, Y, and Z, respectively. For the storage area, the hydraulic conductivity coefficient of concrete B6 - $K=10^{-10}$ m/s. A weak water-conducting area stipulates the deviation from the homogeneity of the velocity field, where the storage is placed. As boundary conditions in a migration task, there was used the condition of zero flow, except the left boundary along X axis (pure water ingress).

The analysis of the results was done by the following items: ground water pollution areas (in IL and YIL/M units, Bq/kg) for a certain estimated time; dynamics of ground water pollution in units IL at control places, located at different distances from the storage and at different depths from the surface; dynamics of radioactivity release rate from the storage (Bq/year); areas of the surrounding rock with adsorbed α- and β-activity above the level of LRW.

We present hereby some investigation results. Non-sorbing activity (^{129}I, ^{3}H) spreads as a passive admixture. In the time course, those radionuclides leave almost entirely the storage area, including by means of radioactive decay (^{3}H). Tritium, due to its short half-life, even at the nearest control place (240 m from the storage), reaches the level of $5{\cdot}10^{-3}$ IL only in the variant of maximum hydraulic gradient.

The analysis of results of calculations of the 3-d group radionuclides' adsorbed activity demosntrated that a considerable part of activity of all the studied radionuclides without exception stays within the storage facility. So, for ^{90}Sr and ^{37}Cs radionuclides, having the highest activity level, the activity release from the storage over the 300-year period after the nuclear emergency does not exceed $1.5{\cdot}10^{-5}$ and $2.0{\cdot}10^{-6}$ of the original activity of those radionuclides, respectively. The activity, released from the storage is adsorbed, basically, down the flow and the areas with the activity, adsorbed one the rock are clearly seen. Fig.9 presents spatial distributions of the adsorbed activity for α - emitter ^{239}Pu at the maximum hydraulic gradient for some 10,000 years after the emergency. The share of the released activity of that long-lived radionuclide, for example, 5,000 years after the emergency makes only 0.3%.

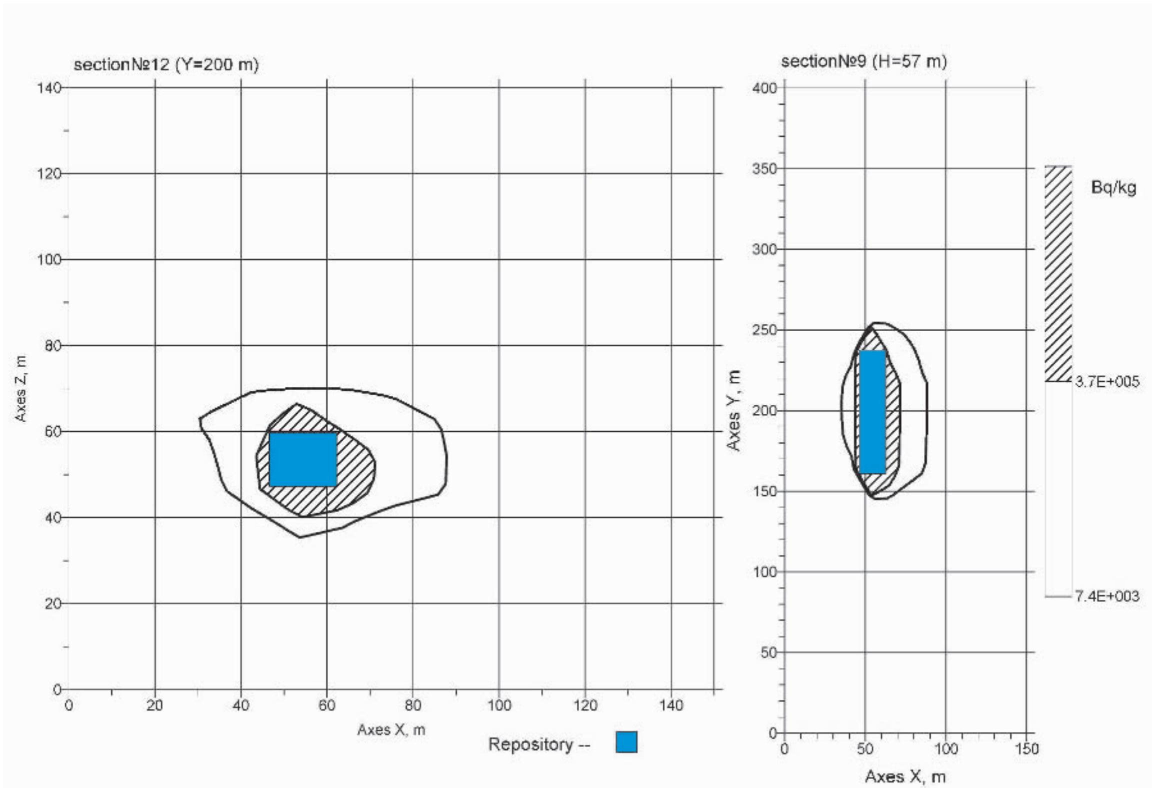

Figure 9. Distribution in the host rock of the adsorbed activity of ^{239}Pu for 10,000 years with hydraulic gradient 0.05.

The implemented studies allowed us to assess the potential spatial area of adsorbed α- and β-activity in the surrounding massif. The high sorption properties of rocks of the Scandinavian shield in the conditions of the reducing redox-potential provide insignificant activity release from the storage area for a large group of radiologically hazardous radionuclides.

It is shown that as a result of the sorption mechanism of moderation of radionuclides transfer and radioactive decay, ensured are high shielding and protective properties of surrounding rocks in prevention of the man's habitat pollution. Thus, taking into consideration the adopted model representations one may state, that converting an underground facility of a long-term SF storing as a result of emergency impact into a new form, that is, into a facility of uncontrolled disposal of non-processible SNF and RW, mainly, confirms the capability of rock massif to perform a shielding and protective functions for prevention of the man's habitat pollution. The carried out studies demonstrated high properties of granite geological formations of the Kola Peninsula as a containment of long-lived SNF activity.

5. Conceptual general layout of the SNF storage facility

The concept of SNF management adopted in Russia, as far as the SNF containerization engineering solutions are concerned, envisages the application of a radiation-shielded container and canisters, which can be used in SNF storage and transportation. In case of long-term storage it is suggested to build near-surface storage facility in geological formation at a depth of about 100-150 m, and to use dry package storage and multi-barrier isolation system.

The MBK-40 container is a metal-concrete radiation-protective dual-purpose container, the design of which makes it possible to use it in SNF transportation and storage.

When a canister is used as a storage packing, an extra protection is necessary in handling the packages. To transport SNF within the territory of the storage facility and delivering it into the near-surface module, a special protective transport container is used.

The basic variants are module designs which are to be used for SNF storage of in metal-concrete containers or in a built-in reinforced concrete structure. These designs allow the SNF packages to be effectively placed, which will result in reduction of mining operations volume, in mine workings maintenance cost lowering. These designs also provide much wider possibilities to control and manage the processes occurring in the rock mass and barriers both during construction and operation of the storage facility.

The module for SNF storage in reinforced concrete containers is a chamber-type working with a cross-section of 155 m^2. To deliver containers, a transport way of 39 m^2 in section and 30 m long is built from the end of the module. At the opposite end of themodule, there is a ventilation connection of 18 m^2 in section and 20 m long. The module is divided into a receiving area and a storage area. The schematic layout of the module is given in Fig.10 by a reinforced concrete partition wall.

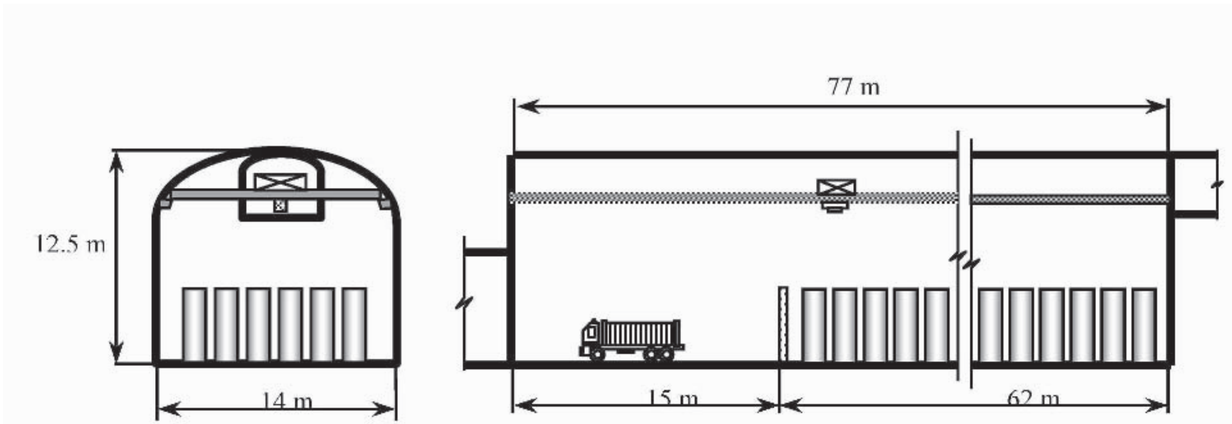

Figure 10. The layout of the module to be used as storage for containers

Reinforced concrete containers are placed in the bottom of the module, vertically, one by one from top to bottom. The distance between containers is 0.3 m, which is stipulated by the task of heat elimination from the outside surface and application of crane equipment in disposing containers. The basic (geometric) dimensions of the module are as follows: length is 77 m, width - 14 m, height - 12.5 m. The total volume of the module is 11,935 m^3. The module capacity is 210 containers of MBK type and 9 SRP in protective containers.

The module for SNF to be disposed of "CASCADE" canisters is a chamber working of 244 m^2 in cross-section and has an additional built-in structure. Fig.11, as an example, offers a module for canisters storage

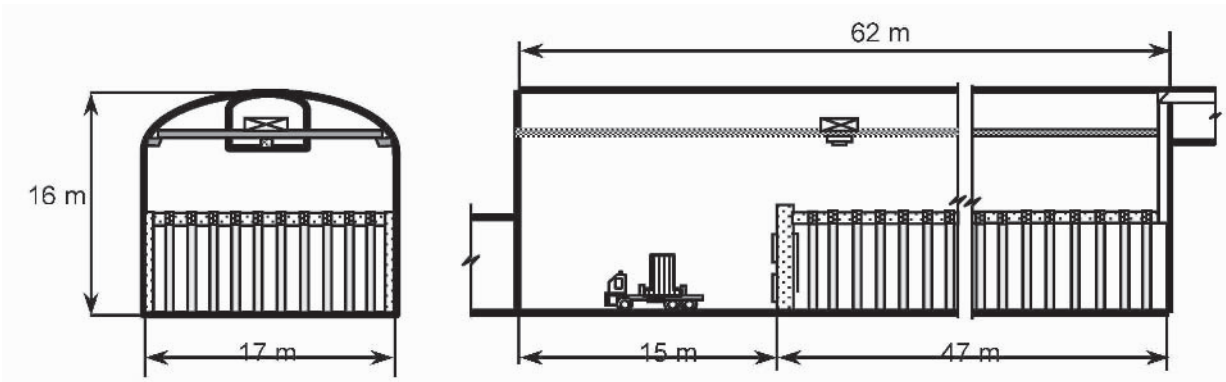

Figure 11. The layout of the module to be used as storage for canisters

The built-in structure consists of reinforced concrete walls and a ceiling. The ceiling has metal tubes-pits built-in into which canisters are disposed. The pits footing is sealed and these are closed with a sealed plug on the top. Each pit receives two canisters, placed from top to bottom. The heat elimination from the outside surface of pits may be ensured as a result of compulsory air feed and by means of natural convection. The spent air is drained off via pipes from the opposite end of the module through the ventilation connector. The basic (geometric) dimensions of the module are: the length is 62 m, width - 17 m, height - 16 m. The total volume of the module is 15,370 m^3. The holding capacity of the module is 1,900 canisters.

The additional module is designed for low-activity waste storage, which is produced in the course of the storage facility operation. It is a chamber working of 83 m^2 in cross- section. Waste in received for the storage in metallic drums of 0.6 or 1.6 m in diameter.

In addition to modules for SNF and operational RW, the storage facility includes a number of main and auxiliary workings. The main-access-working access ramp is designed for the following operations: to deliver containers from surface to underground, transport the staff, deliver empty transportation containers from the storage facility to the surface, mining equipment traffic, deliver broken rock mass to the surface feed the ventilation stream, and arrange the main communications (cables and pipes). It is necessary to deliver packages and personal differently in time and space mainly in order to ensure safety.

The module level floor follows a ring pattern formed by transport and ventilation-and-assembly galleries. Along with the main workings there are a number of auxiliary chamber workings for accommodation of services and stationary equipment located on the module level floor.

It should be noted that the structural-and-design lay-out envisages setting up of an extended near surface infrastructure to ensure near surface storage facility operation.

The preliminary technical-and-economic assessment has been performed for the two variants of the near surface SNF storage facility configuration: variant 1 - SNF is placed into canisters within a built-in reinforced concrete structure; variant 2 - SNF is placed in metal concrete containers. The variants suggested are characterized by the following values:

Designed volume units:		Designed volume of the storage facility:	
canisters	1,900	for canisters, cub.m.	109,230
containers	210 and 9 SRP	for containers, cub.m.	106,035

The construction and operation costs for the basic objects of the SNF storage facility are summarized in Table 5.

TABLE 5. Construction and exploitation costs for the SNF storage facility

Object	**Costs, mln.rbls**		**Costs, mln.$US***	
	Variant 1	Variant 2	Variant 1	Variant 2
Construction				
Surface facility	540	540	17	17
Near surface facility	1,380	1,103	44	35
Total	1,920	1,643	61	52
Equipment (incl. installation)	241	232	8	7
TOTAL	2,161	1,875	69	59
Operation if SNF and RW disposed	67	25	2	1

Note: - The Central Bank of Russia exchange rate 31.7 rbls for 1 $US as of Nov.10, 2002*

6. Site selection for the SNF underground storage facility in the North-West region of Russia

With a number of guidelines approved by the international community concerning site selection for long-term storage and storage of SNF, a type methodology used in making a comparison between the sites as well in making a choice of an optimal one has not been developed so far.

The analysis of the worldwide experience has shown that in various countries the approaches used in the solution of the problem under discussion differ substantially from each other, however, two main trends [17] can be distinguished here.

The first one amounts to making a preliminary decision to operate any known site and to check its being adequate. Another one needs sequential fulfillment of some obligatory procedures:

- to determine the criteria of the site adequacy; whether to exclude them or prefer:
- to list potential sites;
- to determine the parameters assessing each site both from the technical and social-economic viewpoint;
- assess the preliminary chosen sites by means of a multifactor analysis and site ranking;
- approve the most preferable site.

The experience gained by many countries shows that both trends are efficient and can yield good results in some cases the first one being able to require much lower costs.

The Mining Institute considered an intermediate variant, the number of "attractive" sites was limited, these being compared in compact form.

The ideas listed above are valid for both the surface and near surface variant of a SNF storage facility. Nevertheless, it is quite obvious that depending on the placement of the storage facility on the surface, in near surface or deep geological formations, the significance and the importance of some parameters characterizing the site will change substantially. In particular, if placed near surface, of great importance will be technical factors if placed on surface, the social ones, for instance, consent of the population living within the territory in which the approved site is located. However, with any variant, the ecological safety and economical acceptability play the key part in the long run.

The safety of a near surface SNF storage facility and its economical acceptability depend on many factors which amounted to three groups. Group I characterizes the natural conditions of the site, which are able to effect the storage facility operation. Group II determines the safety of the designed facility under its normal loading and operation. Group III addresses the issues of economical and social assessment. The preliminary chosen sites have been studied in terms of probability and the degree of impact of both natural and man-made effect under a normal scenario of the events development and under emergency.

A joint placement with other radiation hazardous or nuclear objects allows construction and operation cost to be substantially reduced. For example, building in the same site a radioactive waste storage facility and a long-term SNF storage facility enables use for both objects not only one and the same transport infrastructure but in part an underground one (access galleries, laboratories etc.).

Based on the studies carried out by the Mining Institute in joint co-operation with some West European partners and Russian organizations, a number of potential sites have been chosen, including:

– near the settlement of Dalny Zelentsy (Murmansk region);
– near the of Kuzreka river (Murmansk region);
– on the Shapochka mountain (Arkhangelsk region);
– the Saida-Bay (Murmansk region).

The above sites ranking was carried out taking into account the criteria concerning the natural conditions, the effect on the fauna and vegetation in the storage facility operation, the population residential areas and its density in the nearest territory, as well as economical indices. In addition, a correction condition was introduced, which was connected with the reliability of the information available for each site. The site ranking results are presented in Table 6.

TABLE 6. The site ranking score

Indices	**Sites**			
	Dalny Zelentsy	Kuzreka	Saida-Bay	Shapochka
Optimal assessment score	86	71	78	88
Optimal assessment ranking	2	4	3	1
Optimal assessment score, reliable information taken into account	88	71	80	69
Site ranking, reliable information	1	3	2	4

Thus, the site near the settlement of Dalny Zelentsy, the Barents sea coast, has been recommended as the most suitable one for a near surface SNF storage facility.

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SPENT NUCLEAR FUEL STORAGE

PROBLEMS OF REMEDIATION OF CONTAMINATED FACILITIES, TERRESTRIAL AND AQUATIC SYSTEMS

RADIATION RISK DURING WATERBORNE STORAGE OF NON-DEFUELED REACTOR COMPARTMENT UNITS

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1. Introduction

So far the procedure of complex decommissioning of Russian Nuclear Submarines (NS) has been limited to: -containerization of high-active equipment of Reactor Installation (RI) inside Reactor Compartment (RC) and -temporary waterborne storage of cut out RC as a component of multi-compartment units. Works on making up of one-compartment Reactor Units (RU) for storage on a solid basement are actually at the initial phase. Radioecological hazard of waterborne storage of large-in-number NS withdrawn from service in the Navy and cut out RC increases significantly if their reactors still house Spent Nuclear Fuel (SNF).

Under normal waterborne storage conditions of NS and RU only outer surfaces of their strong hulls are in contact with marine water. But in a case of emergency due to buoyancy failure the following two scenarios could develop:

- destruction of only strong hull and marine water ingress therein; and
- destruction of both strong hull and reactor installation and marine water ingress inside RI.

2. Marine Water Contamination in Case of RU Sinking and Destruction of Non-defueled RI

In a case that strong hull of an afloat-stored RC is destroyed but its RI remains intact, the following main sources of radionuclide release are prevailing: corrosion products of the reactor vessel and, to a lesser extent, those of fragments of metal-water shielding tank. But if both strong hull and RI are destroyed, marine water enters inside RI. Because the actually used process technology does not provide for re-sealing of reactor upper head after NS defueling [1], in case of RI destruction marine water comes into

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contact with inner surfaces of reactor elements. The release rate of fission products and actinoids into marine water reaches $5 \cdot 10^9$ Bq/h [2].

The release rate of radionuclides generated inside RC into outboard water depends on the extent of seal failure of its elements, i.e., on the number and size of openings (holes), as well as on specific hydrological conditions within waterborne storage areas (tidal phenomena and ejection caused by undercurrents). Radionuclide spreading over water area is principally determined by local peculiarities of tidal phenomena and wind-effected currents.

The dynamics of generation of the radioactive contamination within bays housing afloat-stored RU can be described as follows. For some time - determined by the intensity of turbulent mixing, water circulation velocity in bays depending on tidal and wind-effected phenomena and the specificity of RU arrangement in individual bays - the radioactivity remains within the bay limits. Next, the radioactivity reaches the bay narrow entrance and spreads over the surrounding water areas. The average value of volumetric activity increases and reaches some equilibrium value. If water mixing is active, volumetric activity within the contamination plume becomes identical to the average-over-bay values several hundred meters from the source of contamination. Under conditions of active mixing of bay waters due to tidal and wind-effected phenomena, the coefficients of vertical and horizontal diffusion reach 10^2–10^3 cm^2/s and 10^4–10^5 cm^2/s, respectively.

Sea bottom contamination depends on the rate of principal radionuclide deposition. In estimates the deposition coefficient 10^{-5} cm/s can be taken.

Below the results of a calculation of fission product and transuranic element concentrations in reactor core are presented for a standard-type operation of a second-generation NS provided that the NS power resource was completely run-out that contributed to an additional conservatism in estimates. The release rate of fission products and transuranic elements is given in Table 1. From the table data it follows that the rate of corrosion product release is 2 to 3 times less as compared to that of fission products. The conservative estimate of 10^{-2} $year^{-1}$ [3] was taken in the calculations as the release constant.

TABLE 1. Release rate of fission products and transuranic elements, Bq/h

Duration of storage after dismantlement, year	Duration of NS afloat storage before dismantlement, year					
	5		10		15	
	FP*)	TE**)	FP	TE	FP	TE
0	$8.8 \cdot 10^9$	$7.5 \cdot 10^8$	$6.4 \cdot 10^9$	$6.0 \cdot 10^8$	$5.6 \cdot 10^9$	$4.9 \cdot 10^8$
5	$6.4 \cdot 10^9$	$6.0 \cdot 10^8$	$5.6 \cdot 10^9$	$4.9 \cdot 10^8$	$4.9 \cdot 10^9$	$4.0 \cdot 10^8$
10	$5.6 \cdot 10^9$	$4.9 \cdot 10^8$	$4.9 \cdot 10^9$	$4.0 \cdot 10^8$	$4.3 \cdot 10^9$	$3.2 \cdot 10^8$
15	$4.9 \cdot 10^9$	$4.0 \cdot 10^8$	$4.3 \cdot 10^9$	$3.2 \cdot 10^8$	$3.9 \cdot 10^9$	$2.7 \cdot 10^8$

* - FP – fission products
** - TE – transuranic elements.

The calculation results for equilibrium activity of fission products and transuranic elements in the bay waters are demonstrated in Table 2.

TABLE 2. Equilibrium specific radionuclide activities of fission products in bay waters, Bq/kg

A) Fission Products (FP) and Corrosion Products (CP)

Duration of storage after dismantlement, year	Duration of NS afloat storage before dismantlement, year													
	5							10						
	^{90}Sr	^{90}Y	^{134}Cs	^{137}Cs	^{147}Pm	^{151}Sm	^{60}Co	^{90}Sr	^{90}Y	^{134}Cs	^{137}Cs	^{147}Pm	^{151}Sm	^{60}Co
0	2.26	2.26	0.34	2.43	1.49	6.10^{-2}	$5.6 \cdot 10^{-3}$	2.0	2.0	$6.4.10^{-2}$	2.17	0,40	$5.7.10^{-2}$	$2.92 \cdot 10^{-3}$
5	2.0	2.0	$6.4.10^{-2}$	2.17	0.40	$5.7.10^{-2}$	$2.92 \cdot 10^{-3}$	1.76	1.76	$1.2.10^{-2}$	1.93	0,11	$5.5.10^{-2}$	$1.53 \cdot 10^{-3}$
10	1.76	1.76	$1.2.10^{-2}$	1.93	0.11	$5.5.10^{-2}$	$1.53 \cdot 10^{-3}$	1.55	1.55	$2.2.10^{-3}$	1.71	$2,8.10^{-2}$	$5.2.10^{-2}$	$7.78 \cdot 10^{-4}$
15	1.55	1.55	$2.2.10^{-3}$	1.71	$2.8.10^{-2}$	$5.2.10^{-2}$	---	1.37	1.37	$4.1.10^{-4}$	1.53	$7,5.10^{-3}$	$5.0.10^{-2}$	---

Duration of storage after dismantlement, year	Duration of NS afloat storage before dismantlement, year					
	15					
	^{90}Sr	^{90}Y	^{134}Cs	^{137}Cs	^{147}Pm	^{151}Sm
0	1.76	1.76	$1.2.10^{-2}$	1.93	0.11	$5.5.10^{-2}$
5	1.55	1.55	$2.2.10^{-3}$	1.71	$2.8.10^{-2}$	$5.2.10^{-2}$
10	1.37	1.37	$4.1.10^{-4}$	1.53	$7.5.10^{-3}$	$5.0.10^{-2}$
15	1.22	1.22	$7.7.10^{-5}$	1.37	$2.0.10^{-3}$	$4.8.10^{-2}$

B) Transuranic Elements (TE)

Duration of storage after dismantlement, year	Duration of NS afloat storage before dismantlement, year									
	5					10				
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{241}Am	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{241}Am
0	$2.1/10^{-2}$	$1.43.10^{-2}$	$2.7.10^{-1}$	$7.05.10^{-1}$	$8.7.10^{-3}$	$2.01.10^{-2}$	$1.43.10^{-2}$	$2.65.10^{-1}$	$5.52.10^{-1}$	$1.36.10^{-2}$
5	$2.01.10^{-2}$	$1.43.10^{-2}$	$2.65.10^{-3}$	$5.52.10^{-1}$	$1.36.10^{-2}$	$1.92.10^{-2}$	$1.43.10^{-2}$	$1.2.10^{-2}$	$4.35.10^{-1}$	$1.75.10^{-2}$
10	$1.92,10^{-2}$	$1.43.10^{-2}$	$2.65.10^{-3}$	$4.35.10^{-1}$	$1.75.10^{-2}$	$1.86.10^{-2}$	$1.41.10^{-2}$	$2.65.10^{-3}$	$3.41.10^{-1}$	$2.04.10^{-2}$
15	$1.86.10^{-2}$	$1.41.10^{-2}$	$2.65.10^{-3}$	$3.41.10^{-1}$	$2.04.10^{-2}$	$1.78.10^{-2}$	$1.41.10^{-2}$	$2.65.10^{-3}$	$2.68.10^{-1}$	$2.26.10^{-2}$

Duration of storage after dismantlement, year	Duration of NS afloat storage before dismantlement, year				
	15				
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{241}Am
0	$1.92.10^{-2}$	$1.43.10^{-2}$	$2.65.10^{-3}$	$4.35.10^{-1}$	$1.75.10^{-2}$
5	$1.86.10^{-2}$	$1.41.10^{-2}$	$2.65.10^{-3}$	$3.41.10^{-1}$	$2.04.10^{-2}$
10	$1.78.10^{-2}$	$1.41.10^{-2}$	$2.65.10^{-3}$	$2.68.10^{-1}$	$2.26.10^{-2}$
15	$1.71.10^{-2}$	$1.41.10\text{-}^{2}$	$2.65.10^{-3}$	$2.11.10^{-1}$	$2.43.10^{-2}$

3. Occupational Exposure within Waterborne Storage Areas in a Case of Emergency Radionuclide Release into Marine Environment

Calculation method. When dealing with bays and estuaries wherein the decommissioned NS and RU are stored afloat, some exposure paths typical for open sea are unimportant. E.g., there is no ingestion intake for lack of sea fishery and nonuse of both the bay waters and coastal areas for recreational purposes. In such areas the following exposure paths are the most important [4]:

1. Inhalation of suspended aerosol particles, marine aerosols (drop fraction) and steam;
2. External exposure during works on piers, floating dock structures, floating facilities, afloat stored RU, etc.;
3. External exposure from water during diving operations at a distance of several meters from sea bottom (e. g., by ship board) when the exposure from bottom sediments can be neglected;
4. External exposure during underwater operations close to or immediately on sea bottom. (In such a case one has to do with an additional external exposure source due to possible roiling of bottom sediments. Concentrations of re-suspended particles vary within a wide range depending on peculiarities of sea bottom structure and bottom sediment generation. One should also keep in mind that when concentrations of suspended particles exceed 10-20 mg/l, visibility could become very low to conduct any diving operations).

The most accurate estimates of population exposure due to radioactive contamination of marine water can be performed through direct dose calculations via different exposure paths. Evaluation calculations can be done in compliance with the Methodic Guides "Estimate of the impacts of radiation-hazardous operations performed by nuclear shipbuilding enterprises on the environment and population " [5].

The values of exposure dose rate via different exposure paths due to radionuclides concentrated in marine water are given in Article [4] wherein the results of individual dose calculations for different exposure paths are presented.

When calculating the inhalation intake, dose coefficients and respiration intensity data (8100 m^3/year) for adults given in RSS-99 [6] were used. For lack of detailed data on relative concentrations of components in the inhaled air for specific NS and RC storage and basing areas, the appropriate recommendations of IAEA [7] were applied.

The calculations of external exposure for works on piers, vessels, platforms, etc. were performed with due regard for the natural shielding phenomenon using the following IAEA's shielding factors: 0.2 for gamma-exposure and 0 for beta-particles [7].

The exposure from sea bottom for every radionuclide was calculated with consideration for the coefficient k_d of element distribution between water and bottom sediments equal to the ratio between specific concentrations of the element in question in bottom sediments and water [8]. The dose rate was calculated at different distances from sea bottom under the assumption that the latter represented an infinite plane isotropic-emitting layer of a finite width. The contribution of scattered-in-water gamma radiation to dose rate value was also taken into account in calculations.

Using the Methodic Guides [5] trough an assigned value a_i for volumetric activity of i radionuclide in water and a specified human exposure time t_j via path j the radiation dose D_{ij} through this path due to water contamination by the indicated nuclide can be calculated via the following formulae:

$$D_{ij} = a_i p_{ij} t_j$$

where: p_{ij} – exposure dose rate for j path of radiation impact, the volumetric activity of i radionuclide in marine water being equal to 1 Bq/m^3.

The integral dose of radiation impact D_i for different exposure paths can be determined through the formulae:

$$D_i = a_i \sum_j p_{ij} t_j \, .$$

Calculation results. From a variety of options presented in Table 2 the following two ones were chosen for subsequent calculations:

1. *Most conservative option*: NS waterborne storage time prior to dismantlement is 5 years, storage time after dismantlement is 0 years;
2. *Type option*: NS waterborne storage time prior to dismantlement is 10 years, storage time after dismantlement is also 10 years.

The results of dose rate calculations for different paths of radiation impact for the above 2 options are given in Tables 3A and 3B, respectively.

TABLE 3. Dose rates for different exposure paths, Sv/s

A) Time of NS waterborne storage prior to dismantlement is 5 years, time of storage after dismantlement is 0 years

Radionuclide	Inhalation intake caused by marine aerosols	External exposure					
		During works on piers, vessels, platforms, etc.	During diving operations				
			From water	Distance from sea bottom			From roiled bottom sediments with 10 mg/l concentration
				0.5 m	1.0 m	1.5 m	
Sr-90	$1.39 \cdot 10^{-15}$	–	–	–	–	–	–
Y-90	$2.25 \cdot 10^{-15}$	$1.08\cdot10^{-19}$	$1.08.10^{-18}$	$7.39\cdot10^{-28}$	–	–	$1.09\cdot10^{-16}$
Cs-134	$5.95\cdot10^{-17}$	$1.03\cdot10^{-14}$	$1.03\cdot10^{-13}$	$1.20\cdot10^{-12}$	$3.24.10^{-14}$	$8.77\cdot10^{-16}$	$3.10\cdot10^{-15}$
Cs-137	$3.01\cdot10^{-16}$	$2.79\cdot10^{-14}$	$2.79\cdot10^{-13}$	$3.11\cdot10^{-12}$	$8.11\cdot10^{-14}$	$1.99\cdot10^{-15}$	$8.35\cdot10^{-14}$
Pm-147	$1.05\cdot10^{-15}$	$1.48\cdot10^{-19}$	$1.48\cdot10^{-18}$	$2.80\cdot10^{-15}$	$4.62\cdot10^{-18}$	$6.11\cdot10^{-21}$	$2.97\cdot10^{-17}$
Sm-151	$3.46\cdot10^{-17}$	$4.01\cdot10^{-18}$	$4.01\cdot10^{-17}$	$2.28\cdot10^{-18}$	$1.98\cdot10^{-25}$	$2.01\cdot10^{-32}$	$7.99\cdot10^{-16}$
Co-60	$6.20\cdot10^{-18}$	$3.07\cdot10^{-16}$	$3.07\cdot10^{-15}$	$2.62\cdot10^{-12}$	$1.14\cdot10^{-13}$	$5.11\cdot10^{-15}$	$5.64\cdot10^{-14}$
Sum: FP+CP	$\boldsymbol{5.00\cdot10^{-15}}$	$\boldsymbol{3.85\cdot10^{-14}}$	$\boldsymbol{3.85\cdot10^{-13}}$	$\boldsymbol{6.93\cdot10^{-12}}$	$\boldsymbol{2.28\cdot10^{-13}}$	$\boldsymbol{7.98\cdot10^{-15}}$	$\boldsymbol{6.88\cdot10^{-14}}$
Pu-238	$1.68\cdot10^{-12}$	$2.39\cdot10^{-20}$	$2.39\cdot10^{-18}$	$4.73\cdot10^{-17}$	$1.93\cdot10^{-19}$	$4.10\cdot10^{-21}$	$2.39\cdot10^{-17}$
Pu-239	$6.85\cdot10^{-15}$	$3.13\cdot10^{-21}$	$3.13\cdot10^{-19}$	$3.42\cdot10^{-17}$	$5.13\cdot10^{-18}$	$2.29\cdot10^{-21}$	$3.13\cdot10^{-18}$
Pu-240	$1.29\cdot10^{-13}$	$3.45\cdot10^{-20}$	$3.46\cdot10^{-18}$	$8.42\cdot10^{-17}$	$1.93\cdot10^{-19}$	$1.76\cdot10^{-21}$	$3.46\cdot10^{-17}$
Pu-241	$3.60\cdot10^{-15}$	$4.14\cdot10^{-19}$	$4.14\cdot10^{-17}$	$3.66\cdot10^{-15}$	$5.99\cdot10^{-18}$	$7.97\cdot10^{-21}$	$4.15\cdot10^{-16}$
Am-241	$5.20\cdot10^{-15}$	$7.44\cdot10^{-19}$	$7.44\cdot10^{-17}$	$1.46\cdot10^{-14}$	$8.53\cdot10^{-18}$	$1.10\cdot10^{-19}$	$1.49\cdot10^{-14}$
Sum for TE	$\boldsymbol{1.81\cdot10^{-12}}$	$\boldsymbol{1.22\cdot10^{-18}}$	$\boldsymbol{1.22\cdot10^{-17}}$	$\boldsymbol{1.83\cdot10^{-14}}$	$\boldsymbol{2.00\cdot10^{-17}}$	$\boldsymbol{1.26\cdot10^{-19}}$	$\boldsymbol{1.50\cdot10^{-14}}$
Total	$\mathbf{1.81\cdot10^{-12}}$	$\mathbf{3.85\cdot10^{-14}}$	$\mathbf{3.85\cdot10^{-13}}$	$\mathbf{6.95\cdot10^{-12}}$	$\mathbf{2.28\cdot10^{-13}}$	$\mathbf{7.98\cdot10^{-15}}$	$\mathbf{8.38\cdot10^{-14}}$

B) Time of NS waterborne storage prior to dismantlement is 10 years, time of storage after dismantlement is 10 years

Radionuclide	Inhalation intake caused by marine aerosols	External exposure					
		During works on piers, vessels, platforms, etc.	During diving operations				
			From water	Distance from sea bottom			From roiled bottom sediments with 10 mg/l concentration
				0.5 m	1.0 m	1.5 m	
Sr-90	$9.55.10^{-16}$	–	–	–	–	–	–
Y-90	$1.55 \cdot 10^{-15}$	$7.44 \cdot 10^{-20}$	$7.44 \cdot 10^{-19}$	$5.07 \cdot 10^{-28}$	–	–	$7.46 \cdot 10^{-18}$
Cs-134	$3.85 \cdot 10^{-19}$	$6.69 \cdot 10^{-17}$	$6.69 \cdot 10^{-16}$	$7.74 \cdot 10^{-15}$	$2.10 \cdot 10^{-16}$	$5.68 \cdot 10^{-18}$	$2.01 \cdot 10^{-18}$
Cs-137	$2.12 \cdot 10^{-16}$	$1.96 \cdot 10^{-14}$	$1.96 \cdot 10^{-13}$	$2.19 \cdot 10^{-12}$	$5.71 \cdot 10^{-14}$	$1.40 \cdot 10^{-15}$	$5.88 \cdot 10^{-16}$
Pm-147	$1.97 \cdot 10^{-17}$	$2.78 \cdot 10^{-21}$	$2.78 \cdot 10^{-19}$	$5.26.10^{-17}$	$8.68 \cdot 10^{-20}$	$1.15 \cdot 10^{-22}$	$5.57 \cdot 10^{-20}$
Sm-151	$2.95 \cdot 10^{-17}$	$3.42 \cdot 10^{-18}$	$3.42 \cdot 10^{-17}$	$1.94 \cdot 10^{-18}$	$1.69 \cdot 10^{-25}$	$1.71 \cdot 10^{-32}$	$6.82 \cdot 10^{-17}$
Co-60	$8.63 \cdot 10^{-19}$	$2.14.10^{-17}$	$2.14 \cdot 10^{-16}$	$3.66 \cdot 10^{-13}$	$1.58 \cdot 10^{-14}$	$7.11 \cdot 10^{-16}$	$7.88 \cdot 10^{-17}$
Sum: FP+CP	***$2.72 \cdot 10^{-15}$***	***$1.96 \cdot 10^{-14}$***	***$1.96 \cdot 10^{-13}$***	***$2.53 \cdot 10^{-12}$***	***$7.31 \cdot 10^{-14}$***	***$2.11 \cdot 10^{-15}$***	***$7.35 \cdot 10^{-16}$***
Pu-238	$8.91.10^{-15}$	$2.12 \cdot 10^{-21}$	$2.12 \cdot 10^{-19}$	$4.19 \cdot 10^{-18}$	$1.71 \cdot 10^{-20}$	$3.63 \cdot 10^{-22}$	$2.12 \cdot 10^{-19}$
Pu-239	$6.75 \cdot 10^{-15}$	$3.09 \cdot 10^{-21}$	$3.09 \cdot 10^{-19}$	$3.37 \cdot 10^{-17}$	$2.42 \cdot 10^{-19}$	$2.26 \cdot 10^{-21}$	$3.09 \cdot 10^{-19}$
Pu-240	$1.27 \cdot 10^{-15}$	$3.39 \cdot 10^{-22}$	$3.39 \cdot 10^{-20}$	$8.27 \cdot 10^{-19}$	$1.90 \cdot 10^{-21}$	$1.73 \cdot 10^{-23}$	$3.39 \cdot 10^{-20}$
Pu-241	$1.74 \cdot 10^{-15}$	$2.00 \cdot 10^{-19}$	$2.00 \cdot 10^{-17}$	$1.77 \cdot 10^{-15}$	$2.90 \cdot 10^{-18}$	$3.85 \cdot 10^{-21}$	$2.01 \cdot 10^{-17}$
Am-241	$1.22 \cdot 10^{-14}$	$1.74 \cdot 10^{-18}$	$1.74 \cdot 10^{-16}$	$3.43 \cdot 10^{-14}$	$2.00 \cdot 10^{-17}$	$2.57 \cdot 10^{-19}$	$3.49 \cdot 10^{-15}$
Sum for TE	***$3.09 \cdot 10^{-14}$***	***$1.94 \cdot 10^{-18}$***	***$1.94 \cdot 10^{-16}$***	***$3.61 \cdot 10^{-14}$***	***$2.29 \cdot 10^{-17}$***	***$2.60 \cdot 10^{-19}$***	***$3.51 \cdot 10^{-15}$***
Total	**$3.36 \cdot 10^{-14}$**	**$1.96 \cdot 10^{-14}$**	**$1.96 \cdot 10^{-13}$**	**$2.60 \cdot 10^{-12}$**	**$7.31 \cdot 10^{-14}$**	**$2.11 \cdot 10^{-15}$**	**$4.25 \cdot 10^{-15}$**

Due to a variety of combinations of the above-considered principal exposure paths the calculated exposure time over one year is not identical for different professional groups working at the water area. Thus the following three professional groups can be distinguished [4]:

1. *Group #1* includes people to whom inhalation of radionuclides with aerosol particles, drop-like marine aerosols and steam represents the main exposure path. These are workers performing different operations at open-air industrial areas close to water surface. The members of this critical group are exposed to radiation during work as well as when staying in open air outside the production site but within the trace formed by wind-transferred marine aerosols. The integral exposure time for the members of Group #1 is conservatively estimated at 3500 hours per year.

2. For *Group #2* the exposure proceeds via a combination of two principal radiation exposure paths: inhalation radionuclide intake with marine aerosols and external exposure from water surface. This group mainly includes workers of piers, floating dock structures, other floating facilities, etc. contacting directly with the bay (estuary) water. For the members of Group #2 inhalation exposure time is identical to that of Group #1, i.e. 3500 hours; the duration of external exposure from water is taken equal to 1900 hours per year.

3. *Group #3* consists of divers with an additional radiation exposure path – that is external exposure from water and sea bottom during underwater operations. The duration of every diving operation depends on the specificity of individual task to be accomplished. As conservative estimate, the value of 600 hours per year was accepted for the members of Group #3. During underwater operations divers inhale decontaminated air; consequently, the duration of marine aerosol inhalation intake can be estimated at 2900 hours per year. The time of external exposure from water surface for the members of Group #3 should be also corrected for diving duration. Therefore the "planned" external exposure time over one year is taken equal to 1000 hours. It is accepted that underwater operations are performed at 1-m distance from sea bottom; consequently, the integral effective dose takes account of external exposure from both sea bottom and roiled bottom sediments.

The durations of radiation impact via different exposure paths for three professional groups are presented in Table 4; the results of effective dose calculations are demonstrated in Table 5.

TABLE 4. Duration of radiation impact for different professional groups over one year, hour

Exposure path	Group #1	Group #2	Group #3
Inhalation intake from marine aerosols	3500	3500	2900
External exposure from water surface	-	1900	1000
External exposure during diving operations	-	-	600

TABLE 5. Effective dose (Sv) via different exposure paths

Professional group	Inhalation intake from marine aerosols	External exposure				Sum
		During works on piers, ships, platforms, etc.	During diving operations			
			From water	From sea bottom	From roiled bottom sediments	
Time of NS waterborne storage prior to dismantlement 5 years, time of storage after dismantlement 0 years						
#1	$2.28.10^{-5}$	-	-	-	-	$2.28.10^{-5}$
#2	$2.28.10^{-5}$	$2.61.10^{-7}$	-	-	-	$2.31.10^{-5}$
#3	$1.85\cdot10^{-5}$	$1.38\cdot10^{-7}$	$8.25\cdot10^{-7}$	$4.92.10^{-7}$	$1.81.10^{-7}$	$2.01\cdot10^{-5}$
Time of NS waterborne storage prior to dismantlement 10 years, time of storage after dismantlement 10 years						
#1	$4.23.10^{-7}$	-	-	-	-	$4.23.10^{-7}$
#2	$4.23.10^{-7}$	$1.34.10^{-7}$	-	-	-	$5.57.10^{-7}$
#3	$3.49.10^{-7}$	$7.06.10^{-8}$	$4.23.10^{-7}$	$1.59.10^{-7}$	$9.27.10^{-9}$	$1.09.10^{-6}$

An analysis of the calculation results shows that, if using the conservative approach (time of NS waterborne storage prior to dismantlement 5 years, that of storage after dismantlement 0 years), under the taken assumptions effective exposure doses would be approximately the same for all professional groups equaling ~ 20 μSv/year. The main contribution to the exposure dose would be due to inhalation intake of radionuclides with marine aerosols.

In the "type option" case the members of Group #3 (divers) would be the most affected: the effective dose ~ 1 μSv/year, the main contribution being due to exposure from water.

According to Reference [9], the probability of sinking for RU during waterborne storage within a temporary storage center does not exceed 1.10^{-7} 1/year. Using the values of coefficients for individual lifelong risk and damage due to occupational exposure equaling 0.056 1/Sv [6] and 0.8 man.year / man.Sv [10], respectively, one may obtain the values of individual annual risk and damage for a case of RU sinking accompanied by destruction of both strong hull and reactor installation (see Table 6).

TABLE 6. Radiation risk for different professional groups during works at water areas of RU waterborne storage

Professional group	Effective dose, Sv	Individual fatal risk, 1/year	Individual annual damage in lost years of life, years/year
Time of NS waterborne storage prior to dismantlement 5 years, time of storage after dismantlement 0 years			
#1	$2{,}28.10^{-5}$	$1.28.10^{-6}$	$1.82.10^{-5}$
#2	$2.31.10^{-5}$	$1.29.10^{-6}$	$1.83.10^{-5}$
#3	$2.01\cdot10^{-5}$	$1.10.10^{-6}$	$1.58.10^{-5}$
Time of NS waterborne storage prior to dismantlement 10 years, time of storage after dismantlement 10 years			
#1	$4.23.10^{-7}$	$2.37.10^{-8}$	$3.38.10^{-7}$
#2	$5.57.10^{-7}$	$3.11.10^{-8}$	$4.46.10^{-7}$
#3	$1.09.10^{-6}$	$6.11.10^{-8}$	$8.70.10^{-7}$

4. Conclusion

The performed analysis has demonstrated that in a case of reactor compartment unit sinking accompanied by destruction of both strong hull and reactor installation the annual individual radiation risk due to work at the water area would not exceed $1\cdot10^{-6}$ corresponding to the level of "negligible risk" according to the Russian Radiation Safety Standards (RSS-99). Individual annual damage expressed in "lost years of life" would not exceed $2\cdot10^{-5}$ years provided that the time of NS waterborne storage prior to dismantlement would make up 5 years and after dismantlement 0 years (i.e., RU sinking would occur immediately after making up). If applying a more realistic option (10 years of waterborne storage prior to and after dismantlement), the value of this parameter would not exceed $1\cdot10^{-6}$ years.

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ACTUAL STATUS AND TOP-PRIORITY PROPOSALS ON REHABILITATION OF A RADIATION-HAZARDOUS FACILITY AT GREMIKHA CMB

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Temporary Storage Facility (TSF) for Spent Nuclear Fuel (SNF) and Radioactive Waste (RW) in Gremikha was designed and constructed in the early 1960s. Since then many SNF and RW have been accumulated therein.

In the course of TSF running shielding barriers of some storage facilities degraded and partly lost their functions. As a result, radionuclides contaminated soils, aquatic systems, buildings and constructions generating thereby secondary sources of radioactive contamination requiring localization and elimination.

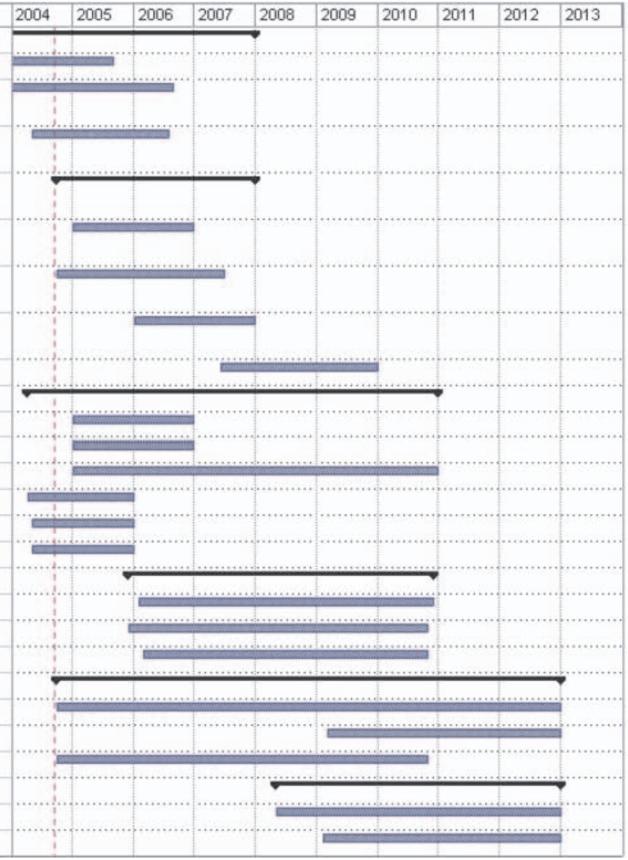
Baffling complexity of solution of nuclear, radiation and environmental safety challenges at the TSF in question is further aggravated by poor condition of its infrastructure giving presently no way of performing radiation-hazardous operations on TSF rehabilitation.

Fundamental decisions on Gremikha TSF rehabilitation with consideration for its geographical location, amount of accumulated SNF and RW, their storage conditions, structure and technical status of supporting infrastructure are determined by "The Concept of Environmental Rehabilitation of Coastal Maintenance Bases in the Northwest Russia" [1].

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 317–337.

TABLE 1. Main types of work on Gremikha TSF rehabilitation and expected terms of their implementation

ID	Work type
1	**1. Urgent preparative works**
2	1.1. Providing infrastructure for works with "Alpha"-class NSs
3	1.2. Ensuring radiation safety during works
4	1.3. Performing IERS of open pad, buildings and constructions
5	**1.4. Design proposals on open pad rehabilitation, performing FS**
6	1.4.1. Management of SNF and high-active SRW at SRW TSF
7	1.4.2. Development of transport and management flowsheet and equipment procurement; repair of buildings and roads.
8	1.4.3. Displacement of SRW and SNF in containers within TSF
9	**2. Rehabilitation of SRW TSF**
10	**3. Design proposals on CMB rehabilitation; performing FS**
11	3.1. Subproject. SRU management
12	3.2. Subproject. Management of SNF of VVER
13	3.3. Subproject. SRW management
14	3.4. Subproject. LRW management
15	**4. Physical protection project**
16	**5. Radiation monitoring project**
17	**6. Management of SNF (both LMC and VVER)**
18	6.1. Unloading of SRU of "Alpha"-class NS
19	6.2. SRU management
20	6.3. Management of SNF of VVER
21	**7. RW management**
22	7.1. SRW management
23	7.2. LRW management
24	7.3. Buildings and constructions
25	**8. Rehabilitation**
26	8.1. Rehabilitation of the object territory
27	8.2. Rehabilitation of the object aquatic systems



From the above table it follows that major works in different areas are to be performed in Gremikha including: -management of SNF of Nuclear Submarines (NSs) with Liquid-Metal Coolant (LMC) reactors and VVER; -management of Solid and Liquid Radioactive Waste (SRW and LRW); -removal of SNF and Spent Removable Units (SRUs) to “Mayak” for reprocessing; -rehabilitation of buildings, constructions, terrestrial and aquatic systems.

This paper addresses the problems and considers proposals on solution of top-priority tasks related to management of SNF (VVER) and rehabilitation of the SRW TSF open pad.

Gremikha Coastal Maintenance Base (CMB) is located on the Barents Sea coast about 400 km from Murmansk-city. So far Gremikha has not been connected with the “mainland” by terrestrial transport communications. People, foodstuffs, building and repair materials are delivered to Gremikha only by sea or by air (helicopter). This is the most important negative factor complicating work execution at the radiation-hazardous CMB. The CMB is situated on coast of Chervianaya Bay (Iogan’ka Roadstead strait of Sviatonosskiy Bay of the Barents Sea) 1.5-2 km by land from dwelling houses of Ostrovnoy-town. CMB total area is 14.98 ha; the “technical area” equals 6.4 ha. The perimeter length makes up: 0.68 km overland and 0.655 km along the coastline. Relief of the CMB site is billowy, the relative height amplitude reaching 25 m.

Previously CMB Gremikha was a naval object used, in addition to the Northern Fleet basing purposes, to resolve the following tasks:

- reload water-cooled (VVER-type) reactors;
- receive, store and forward for reprocessing Spent Fuel Assemblies (SFAs) of the first-generation NSs;
- reload LMC NS reactors and store SRUs of those reactors;
- receive, store and issue for disposal/processing LRW and SRW.

By now the CMB has been transferred under Rosatom’s jurisdiction for preparative-for-rehabilitation works and is actually known under the name of Branch #2 of Federal State Unitary Enterprise (FSUE) “SevRAO”. The former CMB comprises: 32 process buildings and constructions, 20 of them being within the “technical area”. Warehouses are located within the administrative and utility area. General layout of main buildings and constructions of “SevRAO” Branch #2 is demonstrated in Fig.1; maps of the surrounding water areas are given in Fig. 2 (a and b).

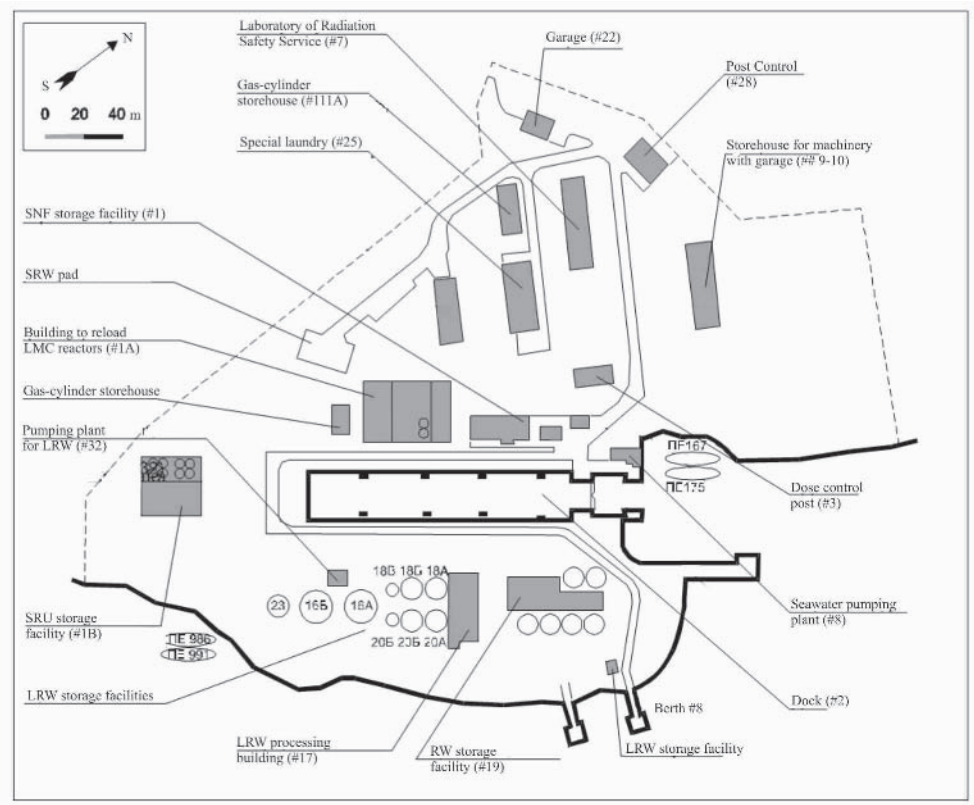


Figure 1. Layout of main buildings and constructions of "SevRAO" Branch #2

a)

b)

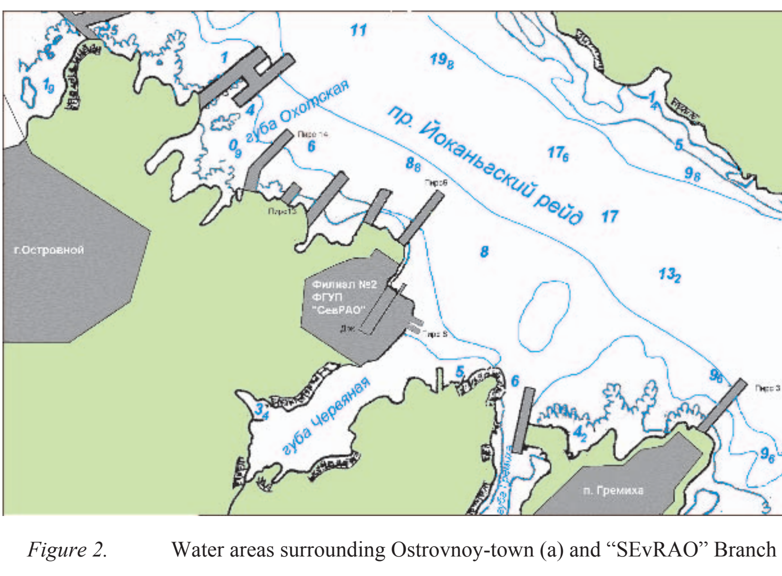


Figure 2. Water areas surrounding Ostrovnoy-town (a) and "SEvRAO" Branch #2 (b)

The first-generation NSs were equipped with "BM-A"-type water-cooled thermal neutron reactors. Today it is very difficult to identify either the type or power production of cores unloaded from the first-generation NSs and stored in Gremikha.

Nine types of cores with different characteristics (power resource, fuel loading parameters, fuel enrichment) were developed for first-generation reactor installations. Any of them could have been unloaded and stored at Gremikha.

Two cores had 6-% fuel enrichment, the remaining cores 21 % enrichment. Mean fuel load therein (^{235}U) was equal to 46.8 kg. According to specifications, maximal fuel burnup made up 20 % of the initial fuel load on average.

The core assembly was of channel type. Schematic diagram of Fuel Assembly (FA) is demonstrated in Fig. 3. A canister-type FA consists of two main components: an insert with fuel element (active part) and a tube-insert (gripping part).

Fuel elements with stainless steel claddings, 0.27-0.3 mm in width and 1000/900 mm in length were used in FAs.

SFAs of reactor installations of the first-generation NSs were temporarily stored at two areas of the CMB: at a special storage facility (Building #1) and outside storage facility in containers (types #6 and #11) at the open-air SRW TSF.

SFA storage in Building #1

Building #1 – storage facility of VVER SFAs - was built in 1962 (designer – Research Institute for Power Technology – Russian abbreviation: VNIPIET), the design storage capacity equaling 1440 SFAs. The storage facility was constructed as 4 independent ponds with a common process hall. SFAs were stored in Building #1 using a standard ("wet") mode – SFA hanging at special built-in holders. Thanks to siding

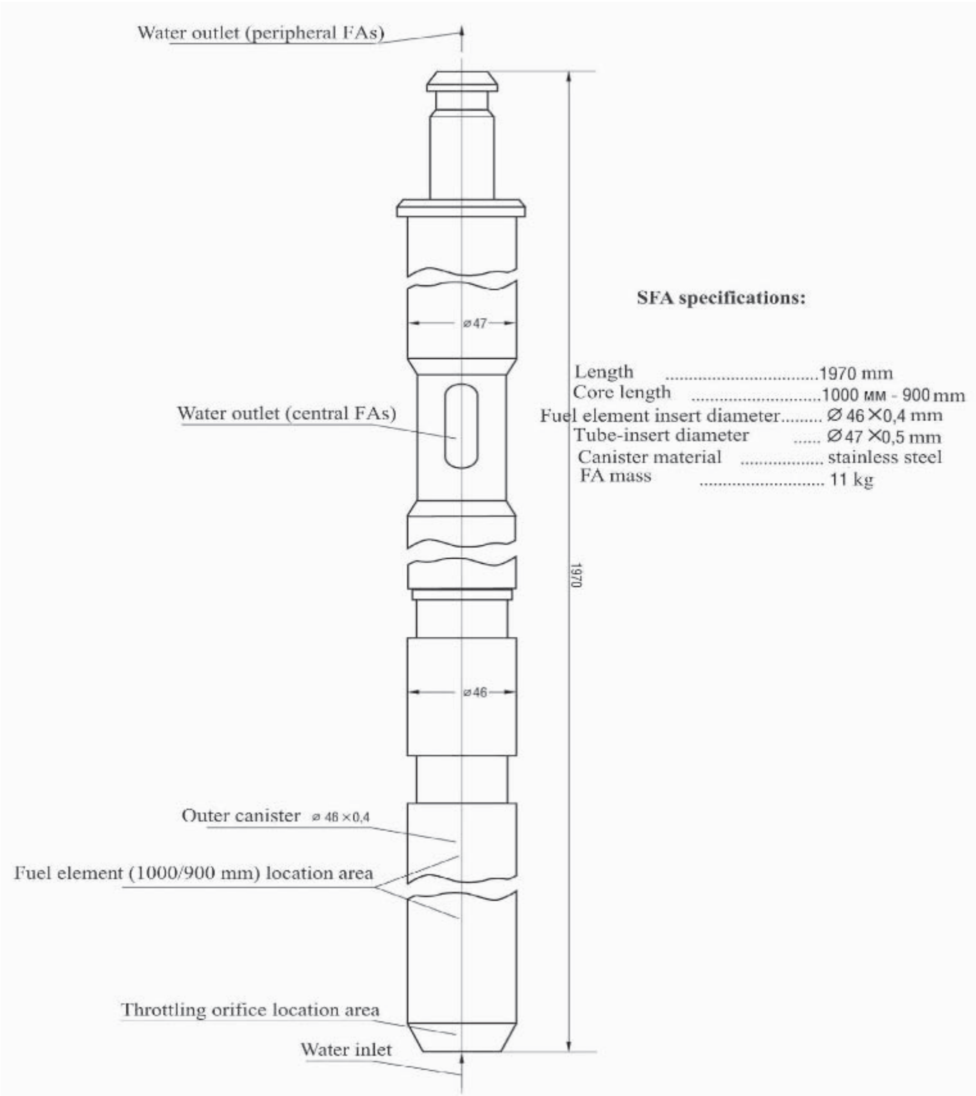


Figure 3. FA Schematic diagram

with a wall of dry dock (SD-10) used to reload NS reactors, SFAs were placed into storage facility directly in the course of core-reloading operations.

In 1984 a leak was revealed in pond # 1. After that SFAs were unloaded from three ponds and placed into pond #2, whereas ponds ##1, 3 and 4 were dried up. Before 1999 in pond #2 106 SFAs were stored, which - due to considerable deformations and damage - could not have been placed using standard procedure into shrouds, type #22, for subsequent forwarding to “Mayak” for reprocessing.

In 1999 all SFAs were unloaded from pond #2 and placed (without observation of design requirements) into shrouds, type #22. A part of unloaded SFAs was considerably deformed and bended. Prior to loading into shrouds, type #22, some SFAs were unbended that resulted in their damage. When loading individual SFAs into shrouds, force was applied. At present all shrouds with SFAs (altogether 16 shrouds, type #22) are located in the intake chambers of ponds in Building #1. Since 1986 running of Building #1 has been forbidden. The available therein SFA transport and management equipment is obsolete and does not comply with the present-day safety requirements. SFA fragments located on Pond #2 bottom were gathered (no design requirements being observed) and loaded into a leaded container, which was next moved to the open-air SRW TSF. There is no information on SFA fragments stored therein, as well as on fragment number, fuel spillage, etc. To obtain input information, examination of the container is necessary.

SFA storage at SRW TSF

At the open-air SRW TSF 107 containers, type #6, and 9 containers, type #11, are stored housing approximately 800 SFAs (Fig. 4). Schematic layout of containers, types #6 and #11, with SFAs and containers with high-active SRW at the open pad is demonstrated in Fig. 5.

In containers type #6 SFAs are stored without shrouds. A special holder is introduced into the inner cavity of container type #6 providing for fixed arrangement of SFAs with a specified spacing. The holder represents a steel cylinder insert with seven openings 60×4 mm for tubes. The generated seven cells (52 mm in inner diameter) are used to house SFAs of the first-generation reactors. On the outside all tubes of the holder are covered by a common casing 219 mm in diameter. The holder is rigidly fixed inside the container case. SFAs are easily placed into the holder cells with no additional fixing. The holder should provide for free water sink from the inner container cavity via bottom openings.

Figure 4. General view of SRW TSF

Container of type #11 was designed for storage and transportation of one shroud #22/#22M comprising seven SFAa. This is a welded construction, which consists of an upper casing and a bottom casing, cylinder tube part, bottom and plug. SFAs are placed into seven tubes 60×3 mm. Shroud is sealed with a plug using a fluoroplastic gasket.

According to the present-day standard requirements, the obsolete-type containers, types #6 and #11, are unsuitable for SFA shipment; at the same time the up-to-date transport-reloading equipment is unadapted to operations with SFAs stored in obsolete-type containers. Consequently, prior to forwarding such SFAs to “Mayak”, development of non-standard equipment and special transport-management flowsheets is necessary.

One more circumstance requiring special management of SFAs is due to their damaged condition. In compliance with the requirements of Russian State Standard #95.957-93 "Spent Fuel Assemblies of Naval Nuclear Reactors. General Delivery Requirements", SFAs should be easily moved in shrouds under their own weight and have no damage of gripping device (head). If SFA does not comply with such conditions, it is considered as “damaged” and cannot be forwarded to “Mayak”. According to the available information, a considerable proportion of SFAs stored at SRW TSF is damaged due to the following:

- bending lengthwise exceeds the admissible value resulting in no possibility of SFA easily placing into shroud cells;
- loss in structural shape: swelling, mechanical damage of a part of SFAs;
- spillage of fuel composition leading to secondary material generation of due to fuel-environment (air, water, ice) interactions; sometimes an amorphous substance is generated in water comprising fuel and destroyed elements of SFA construction.

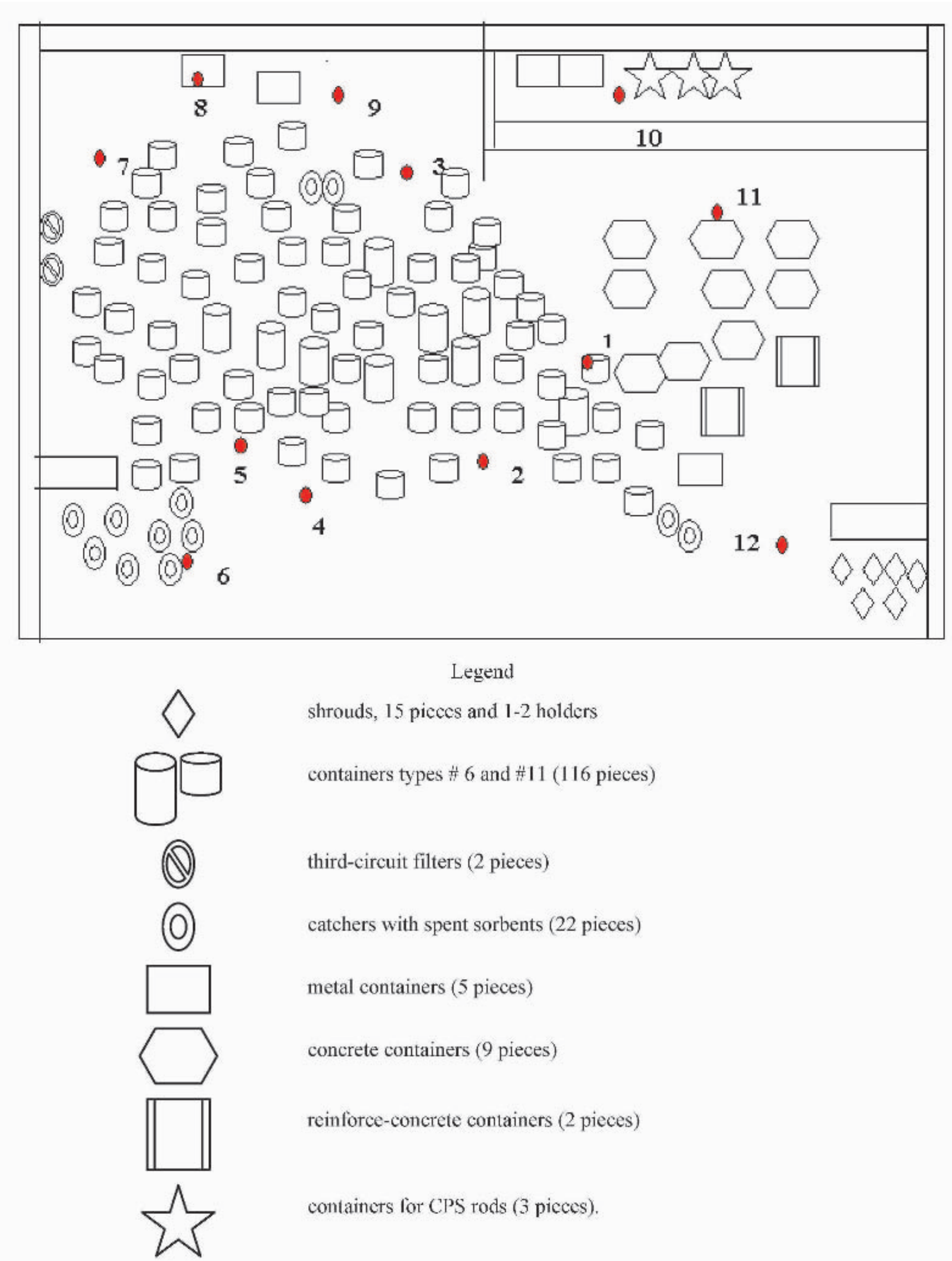


Figure 5. Arrangement of containers with SFAs and SRW at SRW TSF

Data on the degree of SFA damage in containers, type #6 and type #11, as well as in shrouds located in intake sockets of Building #1 ponds are demonstrated in tables 2 and 3.

TABLE 2. Damaged SFAs stored in containers types #6 and #11

No	**SFA defect**	**SFA number**	**Comment**
1	Active part break	46	-
2	Swelling	7	-
3	SFA condition to be clarified	13 · 7 = 91	in 13 containers type #6
4	SFA fragments	?	one container
5	Gripping part deformation	5	-
6	**Total**	149	-

TABLE 3. Condition of 106 SFAs in intake sockets of ponds in Building #1

No	SFA condition in shrouds type # 22	SFA number	Loading characteristic
1	No SFA deformation	31	Easily
2	Before loading into shroud SFA deformation varied from 5° to 75°; unbent (when unbending, a crack was generated)	70	SFA did not entered easily into shroud cell
		3	SFA was hammered into shroud cell
3	Active part fragment was loaded	2	Easily
4	Total	106	-

The actual SFA storage conditions at SRW TSF in containers types #6 and #11 should be considered as extremely adverse.

There is water in some containers types #6 and #11 housing SFAs, which freezes in the cold season. The generating ice squeezes SFAs out of containers that results in additional non-controlled SFA deformation. Due to deformations break and destruction of SFA gripping part, as well as swelling and destroying of SFA canister are possible potentially resulting in impossibility of SFA extracting from containers using standard equipment. Presence of water inside containers aggravates the nuclear safety problems, especially in case of containers type #11. There were some cases when 11 (and not 7) SFAs were loaded without shroud into containers type #11. There is a risk of local critical mass generation in such-type containers in case of SFA destruction, presence of water and fuel-water redistributions.

In the course of on-pad storage no cases of local critical mass generation have been recorded. However, when shipping containers with SFAs, a variety of external impacts are possible (blows, inclinations, turnovers) potentially leading to fuel redistribution. Such a situation could result in increase in neutron multiplication factor if sufficient water amount were available inside container. Criticality estimates performed by specialists of RRC FEI (IPPE) [6] and RRC KI [7] under conservative assumptions confirmed the possibility of developing such a phenomenon. Subcriticality of any fuel configurations inside dried up container is rather high; thus containers housing 7 and more SFAs must be unwatered.

Analysis of non-standard storage conditions of SFAs in containers types #6 and #11 at the open pad has revealed a number of problem issues which solution could allow full-scale implementation of the top-priority SNF management projects. Unfortunately, so far no direct solutions of the problem issues have been developed yet. The following additional investigations are necessary:

Water (LRW) management in containers types # 6 and #11

– development of methods of express water (LRW) analysis in containers types #6 and #11 to obtain analysis results over water column height;
– analysis of methods and designing of an installation to perform unwatering and drying up of containers with observance of nuclear, radiation and environmental safety.

Management of SFAs and SFA fuel elements

– selection of methods and procedures of studying SFA condition when performing inventory and flaw detection after protracted storage in non-standard conditions;
– analysis and selection of methods to estimate fuel condition in fuel elements (when performing fuel diagnostics throughout SFA height) stored in containers types #6 and #11 in case of major shielding effect of shielding-wall width. Development of investigation methods and appropriate software to support such works;
– development of express-analysis method for SFA spillage to determine concentration of actinides (U, Pu, Am), Sr^{90} and Cs^{137}; and
– analysis of methods and development of procedures for removal of fuelspillage from containers and shrouds and next its transformation into a form suitable for forwarding for reprocessing; subsequent decontamination of containers.

First and foremost, improvement of environmental situation at the former CMB is connected with establishment of safe conditions for storage and management of SNF and high-active SRW at SRW TSF.

RW stored at SRW TSF are not only a high-active source of external γ-radiation, but also a source of radionuclide migrations via ground beyond the pad boundaries and thus a source of potential contamination of marine ecosystems. Comparisons of the results of investigations performed in 1999 [2] and in 2003 [5] revealed an increase in surface of contaminated areas around SRW TSF, not only containers with SFAs but also those with SRW being sources of radioactive contamination.

Under the present-day situation immediate works are necessary to prevent both further radionuclide spreading via ground from high-active SRW and non-controlled deformation of SFAs in containers caused by weather effects. These challenges are proposed to be resolved through transferring containers with SFAs and those with high-active SRW from SRW TSF under special “shelters” within the former CMB site, the nuclear and radiation safety requirements being observed. Specific locations of such “shelters” should be identified in the course of Integrated Engineering and Radiation Surveys (IERS), thorough examination and estimates of engineering peculiarities of different CMB buildings.

Transportation of both containers with SRW and those of types #6 and #11 containing water and partly damaged SFAs/fuel spillage from SRW TSF to a “shelter” is a potentially nuclear and radiation hazardous operation. Consequently, prior to perform container unwatering, one needs detailed development of process technology for container transportation in compliance with appropriate safety requirements, develop and manufacture special transport facility (or reequip the existing transport facilities), and develop the safest shipment route (possibly, construct a new road).

Thus priority measures should be focused on solution of the challenge of safe temporary storage of containers with SFA can be formulated as follows:

1. Water removal from containers

Water removal from containers with SFAs and their storage in dried up condition is the most urgent task. Its solution would: -slow down further non-controlled destruction of SFAs; -decrease the risk of fuel composition spillage; and -exclude the danger of local critical mass generation.

2. Transferring containers with SFAs inside a temporary shelter

Construction of a shelter (with a roof) above SRW TSF is inexpedient (or impossible) due to the following circumstances:

– strong winds (typical wind speed 20-25 m/s, maximal wind speed 30 m/s giving no way of constructing a lightweight covering);
– establishment of a lightweight covering above SRW TSF would complicate

transport-reloading operations with containers types #6 and #11 to be performed using jib cranes; and

- building of a permanent construction with installation of an inner bridge crane would be an extremely difficult task due to: -considerable radiation contamination level; -rock base and -difficulties when using machinery because of rock precipice on both sides of the pad.

Thus in-site transfer of containers types #6 and #11 from the open pad under a shelter represents the most acceptable and rapidly realizable solution of this challenge.

Construction of such a shelter would eliminate weather effects on containers and would create appropriate conditions for their better identification for purposes of developing a safe project of container transfer to another facility.

To establish such shelters, Building #10 (for containers with SNF) and Building #17 (for containers with RW) could be used.

However before using those buildings for purposes of temporary storage of containers with SNF and RW, their reconstruction (with restoration of some supporting systems) would be necessary.

To remove containers from open pads, appropriate transport and management solutions should be elaborated, and special transportation equipment should be developed and manufactured.

All works should be performed in compliance with safety requirements of the Russian Radiation Safety Standards (Russian abbreviation: NRB-99) and Main Sanitary Nuclear Safety Regulations (Russian abbreviation: OSPORB-99).

To determine applicability of special standard base when performing design work on transferring SNF and RW into temporary shelters, development of a special guiding document is necessary to identify the status of work (SNF displacement) and determine a list of requirements for temporary shelter of containers with SNF and RW.

A schematic diagram and a list of top-priority works on managing SNF and RW at SRW TSF are demonstrated in Fig. 6.

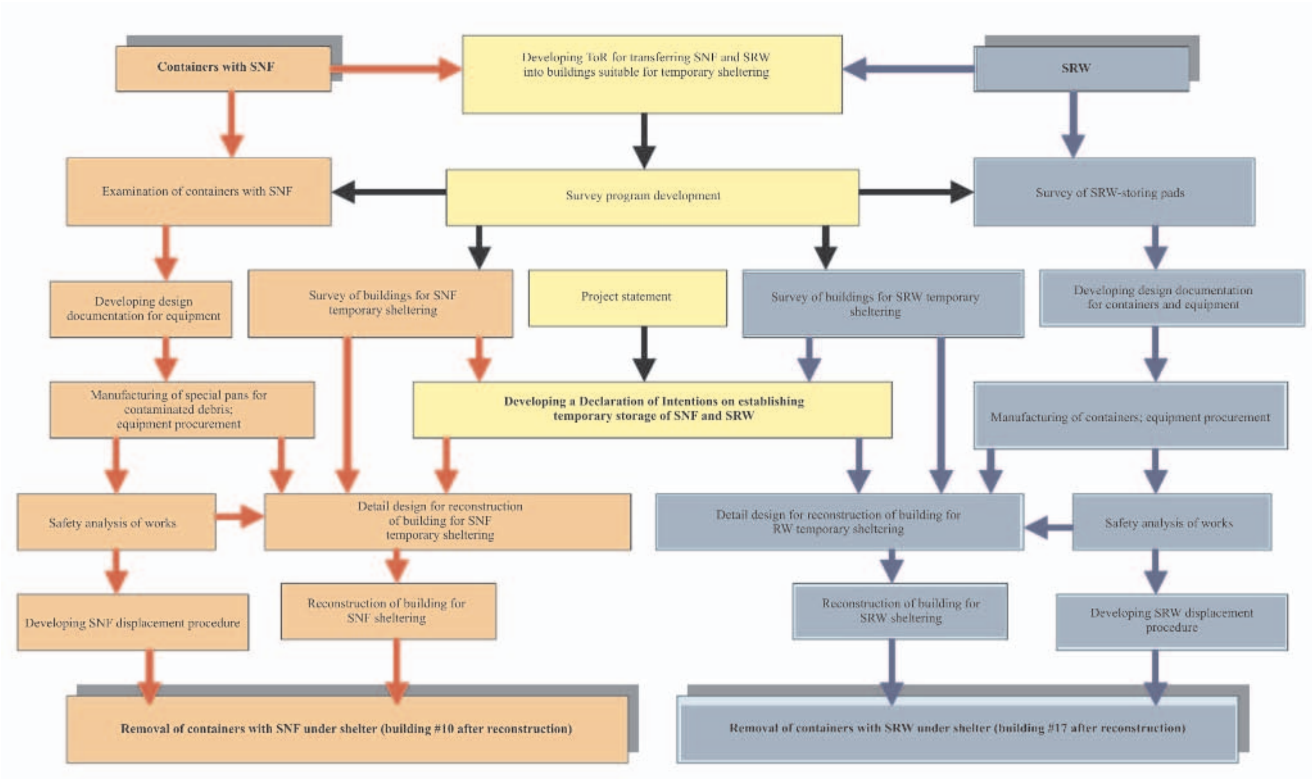


Figure 6. Functional diagram and top-priority works on SNF & RW management necessary to initiate environmental rehabilitation of SRW TSF in Gremikha

3. Establishing a transshipment station for SFA reloading into “ChT” shrouds

It is only by sea that SFAs can be actually removed from former CMB in Gremikha. Individual decisions on SFA removal will depend on the selected SFA management option. However independently on the selected option, establishment of a special transshipment station to reload SFAs from containers, types #6, #11, and shrouds type #22 into up-to-date “ChT” shrouds followed by their loading into TK-18 transportation casks accepted for SFA shipment to “Mayak” is an indispensable condition for taking a decision on SNF removal and subsequent forwarding to “Mayak” for reprocessing.

The necessity of establishing a special transshipment station results from: -availability of many damaged SFAs; -potential spillage of fuel composition; and – need of additional working through of standard flowsheet for SFA reloading into up-to-date shrouds (type “ChT”) using standard reloading equipment (02OK-300 reloading container, KB-650B/KB-651 base container, guiding mechanisms and guide cups) in Gremikha conditions [8].

SFA reloading from containers, type #6 and type #11, should be performed at a specially equipped transshipment station. A possible layout of such a station is depicted in Fig. 7.

The following works are to be performed at the transshipment station: SFA inventory, flaw detection, grading and fuel amount/fuel burnup measurement. Such-type transshipment stations could be established in Gremikha (e.g., Building #1a) or elsewhere (e.g., in Andreeva Bay).

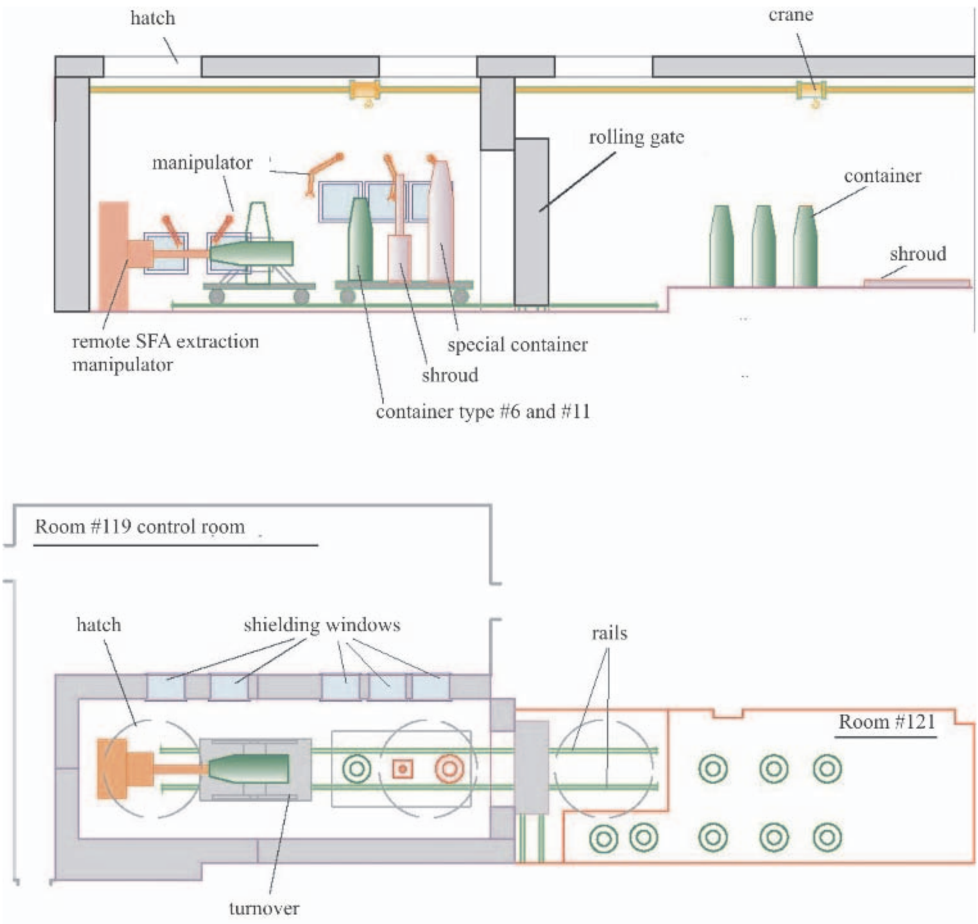


Figure 7. Tentative layout of equipment in shielding chamber to reload SFAs into rooms ##119-121 of Building #1A

Option of SFA reloading at transshipment station arranged on vessel board

Transshipment station is established at a specially-reequipped/newly-built vessel placed, e.g., into SD-10 dry dock. In such a case all inventory and reloading operations are performed at the vessel in question followed by reloading of “ChT”-type containers into TK-18 transportation casks.

Shipment of SFA-housing containers, type #6 and type #11, in transportation casks

Another option is possible providing for shipment of containers, types # 6 and #11, and shrouds, type #22, with SFAs to Andreeva Bay on board of a container ship in special transportation casks for further management of SFA stored therein.

After transportation cask delivery to CMB in Andreeva Bay, SFAs are reloaded from containers types # 6 and #11 to “ChT”-type shrouds followed by one more reloading into TK-18 casks and forwarding to “Mayak”.

This option provides for establishment at CMB in Andreeva Bay of a transshipment station to perform SFA inventory and reloading, if necessary. To perform reloading in Andreeva Bay of SFAs presently stored in Gremikha in containers, types #6 and #11, the issue of allowability of shipping containers with SFA, types #6 and #11, from Gremikha to Andreeva Bay needs resolution.

Option of SFA removal in shrouds, type #22, to “Mayak”

One more option is also possible providing for SFAs stored in shrouds type #22 in Building #1 to be directly forwarded to “Mayak”. In such a case the SFA removal flowsheet could be described as follows:

- perform repair (including in-dock overhaul) of one Floating Service Vessel (FSV) design 326m (PM-124) presently used to transship SNF of retired NS onto board of FSV design 2020 at “Zvezdochka” shipyard water area (Severodvinsk); obtain authorization for use of PM-124 to receive shrouds with SNF in dock SD-10 and transfer them on board of FSV, design 2020, at Gremikha CMB water area;
- tow (after overhaul) PM-124 to Gremikha CMB water area;
- unload shrouds, type #22, from blind areas of Building #1 using reloading container KB-651 and prepare them for shipment on board of PM-124;
- place FSV PM-124 into dock SD-10 and prepare the FSV for receival of shrouds, type #22, with SNF;
- using bridge crane of dock SD-10 (or other loading machine) and by means of base container KB-651 transfer 16 shrouds with SNF to PM-124 storage facility, storage capacity of the latter being 80 shrouds;
- arrange FSV, design 2020 (or “*Lotta*” FSV/”*Imandra*” FSV) close to dock SD-10;
- remove PM-124 from dock SD-10 and moor it to FSV, design 2020;
- transfer – by means of FSV design 2020 – shrouds with SNF from PM-124 storage facility on board of FSV design 2020;
- transport shrouds with SNF to “Atomflot” berth, place them into TK-18 (TK-108/1)-type casks and forward to “Mayak”.

It would be also expedient to consider <u>an option of using a berth at the CMB water area.</u> If such option were applied, transport-management SNF removal flowsheet could be described as follows. First FSV design 2020 (or FSV “Lotta”) is moored on Berth #9. Next prepared shrouds with SNF, type #22, are delivered in KB-651 base container by special transport facility to Berth #9 and transferred on board of the FSV. After that the FSV transfers them to "Atomflot" berth.

To implement the above option, one needs to: -examine Berth #9; -determine the possibilities of its running for SNF transshipment purposes; -select pick-and-place devices for KB-651-type container handling and –prepare a route within the CMB site for SNF transfer to Berth #9.

Option of SFA mothballing in containers types #6 and #11

There are grounds to believe that ~ 900 SFAs stored presently in non-standard conditions at Gremikha CMB, as well as about 50% of all SFAs are damaged. About 10% of SFA fuel elements could be damaged too. It is expected that the extent of damage of both SFAs and fuel elements is different. The following types of defects are possible:

– bending lengthwise exceeds the admissible value: attempts on unbending result in SFA cracking;
– individual fragments of SFAs and fuel elements could be different in length and could lose their structural shape; and
– spillage of fuel composition is possible with the generation of secondary materials when interacting with the environment (air and water).

Taking into account that some containers stored at SRW TSF were affected by atmospheric precipitations over many years due to lack/faulty sealing of their lids, ice impacts on SFAs could have resulted in loss in their structural shape and even in generation of amorphous mass comprising fuel and destroyed elements of SFA construction.

If fuel spillage or amorphous mass were available in containers, development of special procedures would be necessary for extracting, gathering and further handling of such conglomerate. It is possible that such procedures would be very sophisticated, expensive and long lasting. Moreover, works with spillage would inevitably result in considerable radioactive contamination, and thus container decontamination would become an extremely difficult task. It is reasonable to expect that the quantity of such containers would be relatively minor (mainly, containers types #11 with SFAs stored therein without shrouds). In such a situation it would be expedient to consider the possibility of fuel mothballing in containers (in the form of spillage or "mechanically non-extractable" fuel), e.g., via filling with furfurol for long-term storage, next transferring such "mothballed" containers into "SRW" category and finally determining an area for their ultimate disposal.

There is presently no way of determining the most advantageous option. Thus in the feasibility study basic technology of every considered option (and, possibly, of other options) should be elaborated in detail. One needs to identify the following issues: necessary equipment, scope of construction, assembly and repair works, cost of every option (wherever possible) and criteria to be applied when selecting an optimal option. Based on such investigations, an optimal option for managing SFAs stored at TSF in Gremikha will be selected along with an alternative for SFA removal for reprocessing.

Top-priority proposals on SNF Management

In compliance with the "CMB Rehabilitation Concept", the ultimate goal of SNF management consists in SNF removal from TSF to "Mayak" for reprocessing. This would exclude both the necessity of performing subsequent nuclear hazard assessment at the site and diminish the radiation hazard of all other works at TSF. However, as follows from the above-said, for that purpose the following tasks should be resolved (or conditions be created):

- Establish safe conditions of SNF storage at TSF through excluding, first and foremost, SFA-water contact via removing water from containers types #6 and #11. Establish a sheltered area for containers and transfer them from SRW TSF open pad;
- Perform full SFA inventory and flaw detection;
- Determine (by piece):
  - SFAs that could be shipped to "Mayak" after reloading into shrouds type "ChT" and casks "TK-18";
  - considerably damaged (swollen and partly destroyed) SFAs which, prior to shipping to "Mayak", would require reloading into "ChT" transport shrouds;
  - SFAs with destroyed fuel elements, including nuclear fuel spillage, for which special handling technologies would be necessary (dissolving in chemical solutions or monolithization for "everlasting" storage);
- Perform works on SFA reloading from containers, types #6 and #11, to "ChT"-type shrouds. Select the most appropriate technology, nuclear, radiation and environmental safety being ensured;
- Select SNF conditioning technologies before shipping to "Mayak"/transient-storage station;
- Develop transport and management flowsheet for SNF removal to "Mayak"/transient-storage station;
- Work through standard and technical documentation related to SFA management;
- Select appropriate technologies for handling of damaged SFAs and possible fuel spillages in containers, types #6 and #11, and shrouds, type #22; and
- Restore necessary infrastructure at TSF to support SFA handling operations.

There is presently no way of performing a high-quality analysis of different options of VVER SNF management due to lack of necessary initial data on actual condition of containers with SFAs, state of buildings and the radiation situation in the areas of possible works.

As a top-priority urgent measure supporting initiation of preparative works on VVER SNF management, one needs creating appropriate conditions for safe carrying out and control over radiation-hazardous works and normal hygiene and sanitary conditions for personnel (functioning of: decontamination room, radiation safety service and dose control system and availability of individual protectants for personnel). Next, when the above conditions are fulfilled, conducting of a detailed Integrated Engineering and Radiation Survey (IERS) of SFA container storage areas (SRW pad, Building #1) and of their future sheltering areas (Buildings #10 and #17) is necessary. IERS will allow: -obtaining factual data on radiation levels at the areas of planned works and -determining appropriate measures of occupational protection when handling containers with SFAs at their sheltering areas; -identify technical condition of containers including their strapping units and that of buildings #10 and #17. Such-type information is necessary to obtain initial data on development of an appropriate technology for transferring containers with SFAs and high-active SRW to sheltered areas.

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COMPLEX RADIOLOGICAL SURVEY OF TERRESTRIAL AND AQUATIC SYSTEMS IN THE VICINITY OF NUCLEAR SUBMARINE WATERBORNE STORAGE CENTERS AND DISMANTLING ENTERPRISES

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Complex radiological survey of terrestrial and aquatic systems in the vicinity (beyond buffer areas) of nuclear submarine waterborne storage centers and dismantling enterprises is performed on a regular basis by radiometric subdivisions of regional departments of Russian Federal Service for Hydrometeorology and Environmental Monitoring (RosHydromet) under methodic supervision of Scientific and Production Association (SPA) "Typhoon" within the framework of Federal Target Programs funded from the Russian Federation (RF) Budget. Information on radiation monitoring networks in the concerned Russian regions (Arkhangelsk, Murmansk and Kamchatka regions and Primorskiy kray) is summarized in Table 1.

TABLE 1. Structure of Roshydtomet stationary radiation monitoring networks in Arkhangelsk, Murmansk and Kamchatka regions and Primorskiy kray

Region	Type of monitoring, number of stations						
	Dose rate	Atmospheric depositions	Air concentrations	Tritium		^{90}Sr	
				Deposi-tions	rivers	rivers	seas
Arkhangelsk region*	6	2	2	1	1	2	5
Murmansk region	35	9	3	1	-	-	1
Kamchatka region	4	4	1	1	-	-	1
Primorskiy kray**	31	10	1	-	-	1	-

* - in addition, annual control is carried out over radioactive substances in bottom sediments close to Severodvinsk-town in Dvinskoy Bay of the White Sea.

** - in addition are performed: route surveys of the adjacent-to-Chazhma-Bay territory and expedition surveying of Peter the Great Bay.

The results of regular monitoring of the radiation situation in the above regions are generalized in Table 2.

For comparison purposes Table 2 also comprises generalized-over-RF-territory radiation situation data. Comparison of data on volumetric activities of radioactive substances in air, freshwater bodies and sea water between the above regions and averaged RF data over 1999-2003 shows that concentrations of radioactive substances

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in the environmental media at the territories adjacent to radiation-hazardous facilities (outside their buffer areas) do not differ from averaged RF levels.

TABLE 2. Generalized data on radioecological situation in Arkhangelsk, Murmansk and Kamchatka regions and Primorskiy kray

100-km area around radiation-hazardous facility	year	VA of radionuclides in air			Radionuclide depositions		VA in surface waters		VA of ^{90}Sr in sea water, mBq/l
		$\Sigma\beta$, 10^{-5} Bq/m^3	^{137}Cs, 10^{-7} Bq/m^3	^{90}Sr, 10^{-7} Bq/m^3	$\Sigma\beta$, Bq/m^2·day	^{137}Cs, Bq/m^2·day	^{90}Sr, mBq/l	^{3}H, Bq/l	
Murmansk region	1999	6.8	0.87	0.17***	0.9	0.49			2.9
	2000	9.6	1.0	0.1	0.9	bdl			3.4
	2001	12.3	13.6	4.8	1.0	bdl			3.4
	2002	7.0	1.3	0.23	1.1	1.05			3.1
	2003	6.7	1.5	0.45	0.8	0.97			3.6
Arkhangelsk region, SevMash PA	1999	5.5	bdl	1.2	1.2		6.2	2.8	6.0
	2000	6.3	4.9	0.9	0.9		5.7	1.9	4.0
	2001	6.3	4.1	1.7	0.7		6.9	2.4	4.2
	2002	5.1	5.3	1.9	0.5	0.54	7.3	2.0	3.6
	2003	4.1	3.8	1.9	0.6	0.48	5.8	2.4	3.2
Kamchatka region	1999				0.7	0.21**			1.9
	2000				0.7				1.7
	2001				0.8				1.9
	2002				0.8	bdl			2.0
	2003				0.8	0.07*			2.1
Primorskiy kray	1999	23.0	3.0	1.0	1.2	0.30	27.3	3.84	1.9
	2000	22.0	2.0	1.0	1.3	0.23	22.6	3.25	2.3
	2001	21.0	4.0	1.0	1.3	0.62	18.3	3.9	2.1
	2002	18.0	6.0	1.0	1.3	0.19	15.6	2.8	2.0
	2003	17.0	3.0	1.0	1.2	0.29	22.0	3.0	2.1
Averaged RF data	1999	18.6	3.4	1.20	1.3	0.46	6.2	3.4	
	2000	17.4	3.9	1.20	1.4	< 0.4	5.9	2.7	
	2001	16.8	3.7	1.33	1.4	< 0.4	6.1	3.2	
	2002	15.9	4.9	1.19	1.4	0.43	4.8	2.7	
	2003	15.9	4.1	1.56*	1.4	0.34	5.5	2.7	

* - three-month data
** - six-month data
*** - nine-month data

VA- volumetric activity

bdl – below detection limit

VA of radionuclides in air are given in: Murmansk region for Murmansk-city; Arkhangelsk region for Severodvinsk; Primorskiy kray for Vladivostok

SPA "Typhoon" obtains additional information in the course of casual expedition surveys in individual regions and thanks to participation in different international projects. As the result of 1992-2002 expedition surveys performed in radiation-hazardous facility location areas in Kola Gulf, the Kara Sea, Novaya Zemlia bays and the Japan Sea their 3D effects on the environment were discovered, and the contamination levels in seawater, bottom sediments and biota were established.

In 1994 and 1998 specialists of Murmansk region Hydrometeorological Service and SPA “Typhoon” surveyed a fragment of Kola Gulf water area adjacent to “Atomflot” enterprise. In 1995-1998 expedition survey of the entire Kola Gulf and Motovskiy Gulf was performed by Murmansk Marine Biology Institute of the Russian Academy of Sciences [1 and 2]. Those surveys revealed radionuclides in both bottom sediments and algae of the studied gulfs originated from the radiation-hazardous facilities located on their coasts.

As an example, data on specific activities of man-caused radionuclides in bottom sediments and algae taken at “Atomflot’ water area in 1998 are demonstrated in Table 3.

TABLE 3. Specific activity of radionuclides in bottom sediments and algae at “Atomflot” enterprise water area, 1998 (Bq/kg, dry weight)

Isotope	^{60}Co	^{137}Cs	^{152}Eu	^{154}Eu	^{155}Eu	^{40}K
Bottom sediments	< 0.4-14.9	8.4 - 630	< 0.3 - 95	< 0.2 -188	1.4 – 5.4	1850 - 3950
Algae (Laminaria)	< 0.5 – 5.1	3.0 - 260	< 0.6 –10.6	< 1.5 –19.2	< 0.2 –4.0	360-600

As follows from Table 3 data, specific activities of man-caused radionuclides vary within 1-2 orders of magnitude evidencing considerable heterogeneity of contamination levels in bottom sediments and algae. Table 3 also comprises specific activity data for ^{40}K – natural-origin isotope. Comparisons show that in 1998 specific activity of man-caused radionuclides was by 1-2 orders of magnitude less as compared to that of ^{40}K.

Control over radionuclide concentrations in bottom sediments within the water area adjacent to “SevMash” enterprise in Severodvinsk has been performed since 1975. Every year specialists of the Russian Northern Hydrometeorological Service take samples of bottom sediments at 10 points 15-30 km from Severodvinsk. Gamma-spectrometric analysis of samples is performed at SPA “Typhoon”. Table 4 comprises data on ^{137}Cs specific activities in bottom sediments over the last 10 years.

TABLE 4. Averaged (over ten sampling points) specific activities of ^{137}Cs in bottom sediments (Bq/kg, dry weight)

Year	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003
Specific activity	12.3	11.2	9.2	7.4	10.9	8.0	8.6	5.9	7.1	3.1

From Table 4 data it follows that specific activity of ^{137}Cs in bottom sediments has a trend of decreasing over time. Under the used method of analysis concentrations of other gamma-emitting isotopes in samples of bottom sediments were below the detection limit.

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ACTUAL STATUS AND PROSPECTS FOR THE DEVELOPMENT OF AUTOMATED RADIOECOLOGICAL MONITORING SYSTEMS AT OBJECTS AND TERRITORIES CONCERNED WITH COMPLEX DECOMMISSIONING OF NUCLEAR VESSELS IN THE FAR EAST RUSSIA

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According to the Guides [1], monitoring of radioactive contamination in the environment is conventionally subdivided into the following two categories: "information survey" and "research survey". Information survey is performed for purposes of: -prompt revealing and preventing releases of man-caused radionuclides to the environment; and –obtaining on-line data necessary to estimate the radioecological situation and radiation exposure of workers and population. Information survey is conducted continuously being the main type of radioecological monitoring in everyday conditions. Estimates are made via comparisons of the measured values of selected parameters with their standardized values.

Monitoring of changes in the radioecological situation around each radiation-hazardous enterprise is the main methodic principle of information survey. Measurements, which frequency and scope depend on location and specific condition of every environmental contamination source, are conducted as follows: monitoring of seawater, aerosols – every day; dry land, atmospheric depositions – every week; drinking water – every month; coastal and marine vegetation, fish, benthos – once a year.

To perform radioecological survey, special itineraries are established, and special sampling points are selected. When selecting sampling areas, the most-probable-radioactive-source-location principle is used, several (sometimes several tens of) control points being arranged.

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The so-called “research survey” is an extended (“in-depth”) variant of environmental monitoring; however, in real conditions it is most often limited to some “extended” information survey. The research survey is conducted once every three years and more rarely, as well as before commissioning of a hazardous facility and during elimination of consequences of radiation accidents. Samples are also taken using the principle of surveying nuclear- and radiation-hazardous objects and next comparing the survey data with some control “background” points.

Though neither “information survey” nor “research survey” are capable of providing sufficient information to generate the entire “radioecological picture” of the studied territory/water area, they are quite sufficient for discovering eventual radioactive contamination issuing from the surveyed objects.

Since 1990 extended radioecological monitoring based on radiation mapping has been implemented into the radioecological survey practice [2]. Completeness of the environmental contamination information has been attained through generation of radiation field maps with simultaneous indication of source-term location areas, boundaries of radioactive substance spreading, identification of radiation-hazardous zones with indication of the most probable man-caused radionuclide transfer paths.

The described approach has, however, some substantial shortcomings, such as: labouriousness, necessity of applying a wide range of measuring, analyzing and processing tools, high skill of specialists and, consequently, expensiveness.

At present several information databases are maintained in the Far East Russia addressing radioecological status of all nuclear- and radiation-hazardous objects, their technical condition and radiation potential. Information is put at disposal of local, regional and federal authorities and the existing branch-wise crisis centers. According to the established procedure, appropriate information is transmitted to public organizations and mass media.

By now some elements of Automated Radiation Monitoring System (ARMS) have been established at Federal State Unitary Enterprise (FSUE) Far East Plant “Zvezda” (FEP “Zvezda”) and at FSUE DalRAO. At the remaining radiation-hazardous objects of the Far East Russia the so-called “static” monitoring is conducted using standard radiation and radioecological survey equipment (mobile and stationary non-automated gauges).

Since 1998 Ministry for Atomic Energy (Minatom, presently Rosatom) of the Russian Federation (RF) has become responsible for solution of the issues related to complex decommissioning of Nuclear Submarines (NSs) and management of Spent Nuclear Fuel (SNF) and Radioactive Waste (RW) in the North-Western Russia and the Far East Russia.

To fulfill the relevant functions, two Federal State Unitary Enterprises (FSUEs) were established by Minatom - SevRAO in the North-Western Russia and DalRAO in the Far East Russia.

Established by a special decree of the RF Government, DalRAO is Rosatom's representative in the Far East Russia and the main executor of works related to: -management of SNF, Solid Radioactive Waste (SRW) and Liquid Radioactive Waste (LRW) produced in the course of complex decommissioning of NSs and Nuclear-Powered Surface Ships (NPSSs); and -environmental rehabilitation of radiation-hazardous objects in the Far East Russia.

From the organizational viewpoint, DalRAO includes the following four territorial subdivisions:

– DalRAO's Directorate in Vladivostok-city
– DalRAO's representative office in Fokino-town, Primorskiy kray
– DalRAO's Branch #1 in Fokino-town, Primorskiy kray and
– DalRAO's Branch #2 in Viliuchinsk-town, Kamchatka region.

The locations of the main DalRAO's facilities are shown in Figs. 1-2.

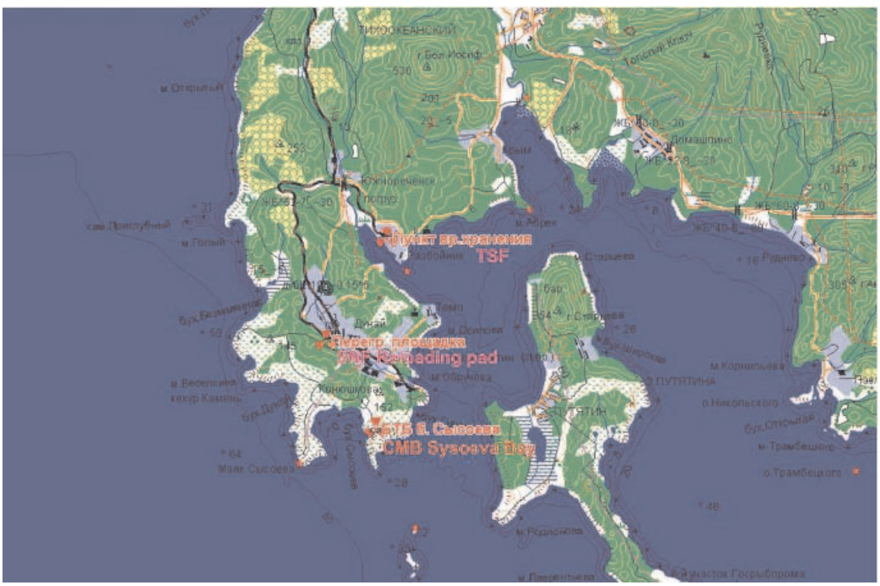

Figure 1. Main facilities of DalRAO's Branch #1, Primorskiy kray

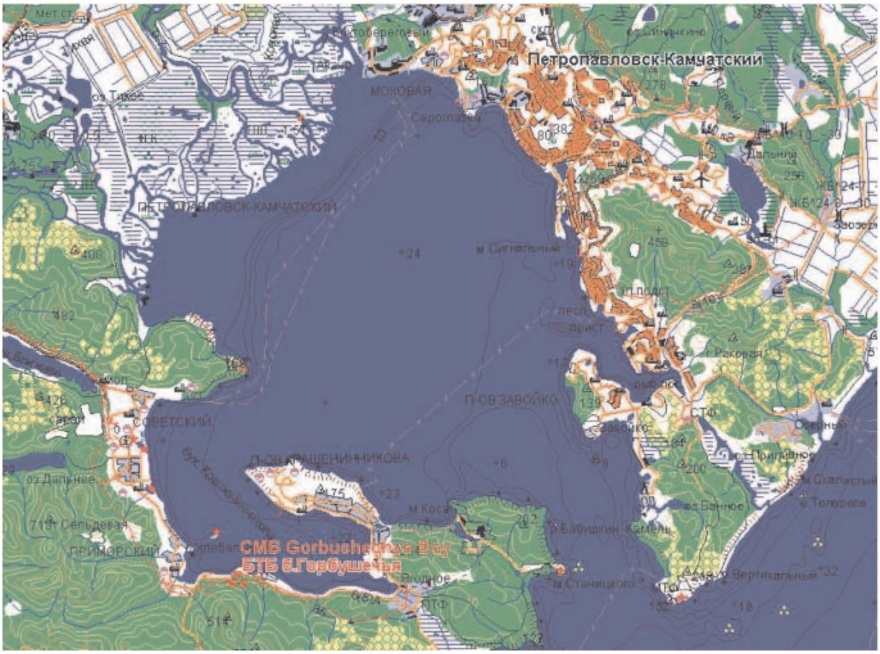

Figure 2. Main objects of DalRAO's Branch #2, Kamchatka region

In 2003 work was initiated on establishment of a DalRAO's Regional Information and Analytical Center for radiation monitoring and Environmental Safety (RIACES) in Vladivostok. The main objectives of RIACES are:

– concentration of information on radiation-hazardous objects concerned with NS complex decommissioning (including their geographical location, general description, current status and radiation potential) and prompt putting of such information at user's disposal;
– receiving, processing and presenting information on radioecological situation within the adjacent territories/water areas with a possibility of applying the results of both already functioning measuring instrumentation of different agencies and newly developed systems;
– simulating radioecological consequences of potential radiation emergencies at the complex decommissioning objects and rendering information support for decision-making on minimization and elimination of their implications; and
– supporting trainings and exercises of workers and agency-level regional units on actions in a case of emergency.

The design of RIACES's - representing, as a matter of fact, a fragment of regional radioecological monitoring and environmental safety system - provides for a possibility of its further extension and build-up.

Within the frames of the first phase of RIACES a software and hardware complex was developed on basis of geoinformation technologies. The complex comprises: -the relevant reference information on complex decommissioning of NSs in the Far East Russia including different-scale digital maps; -information subsystem on each NS subject to complex decommissioning (Reactor Compartment (RC) unit); -information subsystem on waterborne storage centers for NSs and RC units and on-shore radiation-hazardous facilities of DalRAO; and -a databank of reference accidents hypothetically possible in the course of complex decommissioning of NSs (RC units) and management of SNF, LRW and SRW in the Far East Russia.

At that work stage the “pilot” ARMS phase was also established providing for a possibility of visualizing the results at RIACES and comprising several automated dose-rate-measurement channels at facilities of DalRAO Branch #1.

Some fragments of the “pilot” ARMS phase are illustrated in Figs. 3-6.

Figure 3. Exposure dose rate sensor nearby the main building of FSUE DalRAO, Vladivostok-city

Figure 4. Exposure dose rate sensor at the process area of DalRAO (coastal maintenance base in Sysoeva Bay)

Figure 5. Exposure dose rate sensor at DalRAO's temporary storage facility for RC units (Razboinik Bay)

Figure 6. Duty-operator post at DalRAO's temporary storage facility for RC units (Razboinik Bay)

In 2004 specialists of Rosatom's Department for Decommissioning of Nuclear Facilities, DalRAO and Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAS) prepared proposals on establishing a full-scale radioecological monitoring and emergency response system to support management of the taken-out-of-service naval nuclear vessels, SNF and RW in the Far East Russia. These proposals, based on the operational experience of already existing branch-wise and regional systems [3], take account of previous elaborations of different institutions [4 and 5].

The main objectives of this project are:

– prevention and minimization of the consequences of potential radiation accidents at radiation-hazardous facilities and installations concerned with complex decommissioning of nuclear submarines and management of SNF and RW in the Far East Russia;
– supporting decision-making on protection of population and territories, diminishing the consequences of environmental contamination and developing an on-line warning system for the relevant federal and regional structures and services responsible for emergency response activities; and
– informing population and public organizations on the current radioecological situation as well as on potential radiation emergencies.

It is expected that the system under consideration will include: RIACES in Vladivostok and local information and analytical centers in: Primorskiy kray (Fokino-town), Khabarovskiy kray (Sovetskaya Gavan-town) and Kamchatka region (Viliuchinsk-town).

Each local center will unify local ARMS at the following facilities concerned with NS complex decommissioning and SNF and RW management:

– Primorskiy kray – Coastal Maintenance Base (CMB) in Sysoeva Bay, FEP “Zvezda”, Shipyard #30 of RF Ministry of Defense, Temporary Storage Facility (TSF) in Razboinik Bay;
– Khabarovskiy kray – NS TSF in Postovaya Bay; and
– Kamchatka region - CMB in Gorbushechya Bay, North-Eastern Regional Center (NERC) of RF Ministry of Defense, NS TSF in Krasheninnikov Bay;

The planed locations of the future ARMC’s objects are illustrated in Fig.7.

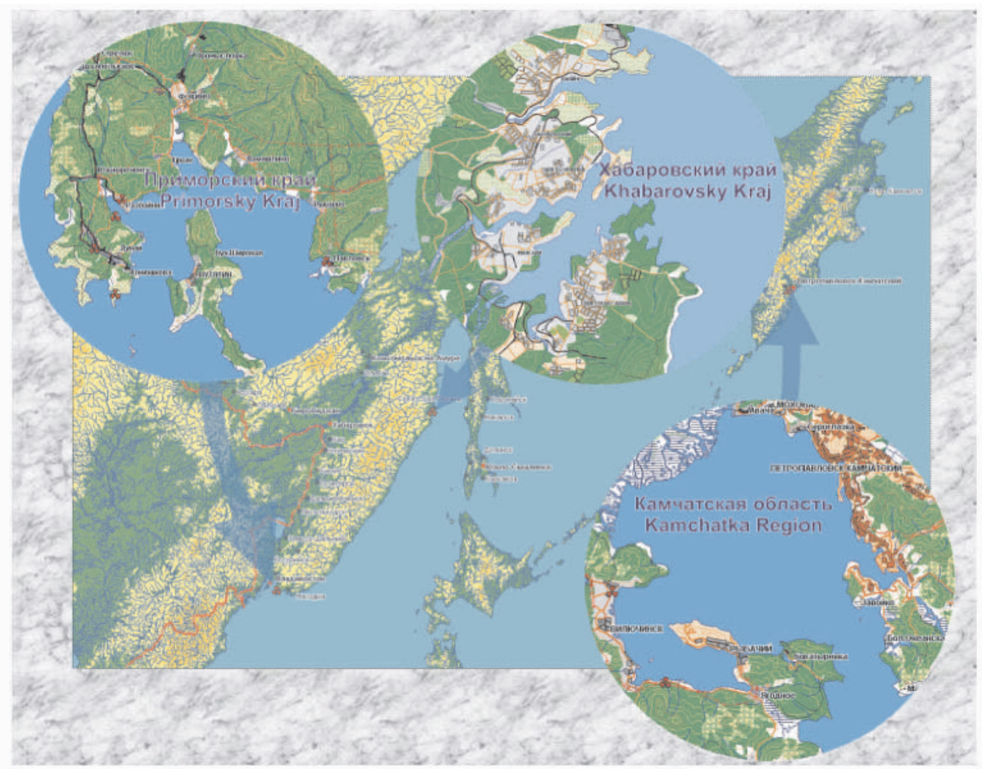

Figure 7. The planned locations of objects of the future ARMC

The ARMS under development will provide for:

– on-line access to information on radiation-hazardous objects, current status of the radiation situation and environment contamination;
– visualization of radiation monitoring data;
– simulations and calculations of radionuclide dispersion in air and water and

estimates of radiation impact on workers, population and environment in a case of radiation emergency;

- support of decision-making on protection of personnel and population in a case of emergency;
- methodic and software support of training and exercises of workers and emergency-rescue units; and
- regular informing of regional authorities and population on environmental risks caused by complex decommissioning of nuclear submarines and management of SNF and RW.

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RADIATION IMPACT ASSESSMENT WHEN DISMANTLING VICTOR CLASS NUCLEAR POWERED SUBMARINES

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The main purpose of the Environmental Impact Assessment (EIA) document is to prepare environmentally safe process procedures for carrying out works in accordance with following tasks:

- The analysis of possible ways of Nuclear Submarine (NS) dismantling. Identification and analysis of potential environmental impact sources within NS dismantling procedures;
- Prediction and thorough study of environmental changes, which may be caused by NS dismantling;
- Prediction and ranking of environmental effects and associated with them social, economical and other consequences of NS dismantling.

EIA purpose is to mitigate environmental negative impact caused by NS dismantling. The basic criteria for assessment of environmental impact caused by chemical and radiation factors of NS dismantling are as follows:

- total hazardous chemicals and radioactive substance releases shall not exceed maximum permissible limits of hazardous chemicals and radioactive materials;
- total hazardous chemicals and radioactive discharge with wastewater shall not exceed established limits;
- industrial (toxic) waste total volume shall not exceed limits specified by limitations on waste disposal;
- production activities and possible emergencies in the process of NS dismantling shall be supported by management measures and safety equipment;
- population safety at the resident area shall be provided in the process of NS dismantling under normal and emergency conditions.

1. Assessment of Radiation Impact on Personnel, Population and Environment under Normal NS Dismantling Process

Figure 1 shows the radiation impact on personnel, population and environment under normal process of NS dismantling.

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 353–360.

The initial data for calculation of nuclear powered plant (NPP) radiation characteristics are the activity values in NPP equipment and process media. To a considerable extent the activity value depends on NPP running regime.

Table 1 gives the results of structural materials calculation activity of NPP and pressure hull of Victor II Submarine.

Calculations indicate that accumulated radioactivity is generally concentrated in reactor internals and reactor vessels. The activity of central caisson and the bottom of metal-water protection tank is less than 2% of gross activity. Pressure hull activity is about 0,04 % of gross radioactivity.

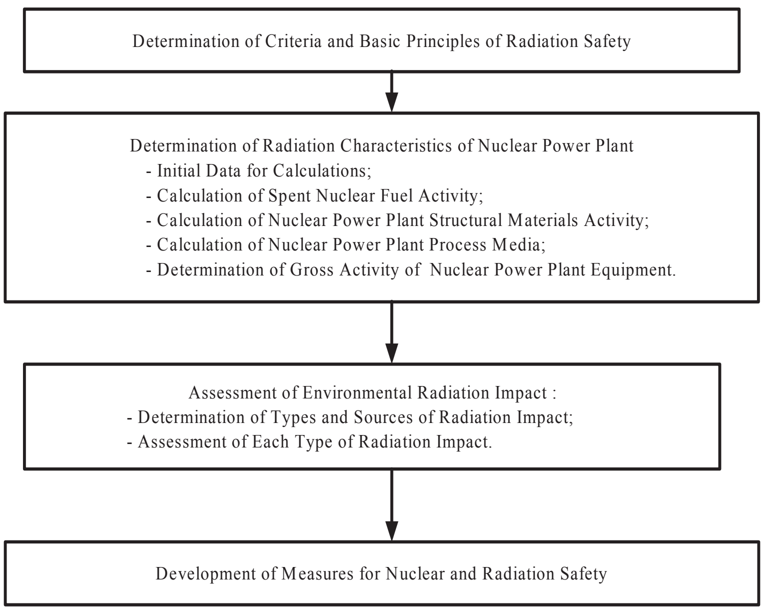

Figure 1. Radiation Impact on Personnel, Population and Environmental under Normal NPS Dismantling Process

TABLE 1. Induced Activity of NPP Structural Materials and NS Pressure Hull in 3-Years Dwell Period

Structural member	Gross activity, Bq
Reactor internals	4.0×10^{15}
Reactor pressure vessel	4.7×10^{14}
Central caisson and bottom of metal and water protection tank	7.3×10^{13}
Pressure hull	1.7×10^{12}

Table 2 gives the calculation results of radiation level by SNF unloading from the drained reactor of Victor- II Submarine.

Personnel dosage due to works with open unloaded reactor will increase to an average of not more than 1 mSv and for entire unloading they will not exceed 2-3 mSv, that amounts to 10-15 % of A-category personnel dose limit specified by NRB-99.

Radionuclide release into the atmosphere under the normal NS dismantling process may occur from the primary circuit or gas system of pressure compensation during the following operations (Table 3).

TABLE 2. Radiation Levels at SNF Unloading from Drained Reactor

Process stage	Dose rate mSv/h	Measurement point
Unloading of resistance thermometer wells	0.4	At the distance of 0.1 mm from thermometer surface
Wells and safety rods trimming	0.4	At the distance of 0.1 m from safety rod well
Reactor cover removal and positioning device installation	2	On bottom face of reactor cover
	1	On open reactor axis at 1.5 m elevation of sliding screen top plate
Works with positioning device	0.02	At 0.5 m elevation from positioning device surface
SNF unloading	<0.001	On container surface

TABLE 3. Radioactive Releases under Normal Process of NS Dismantling

NS dismantling stage	Source of release
Dismounting operations before SNF unloading	Release by high pressure air drop
Removal operations for SNF unloading	Release in SNF unloading operations
SNF unloading	Release in outer and pressure hulls cutting

Radionuclide releases in the process of SNF unloading and outer and pressure hulls cutting (Victor II Submarine) are given in Tables 4 and 5.

TABLE 4. Total Radionuclide Release when Defueling Victor II

Radionuclide	Bow reactor plant Bq	Aft reactor plant Bq	Annual limit Bq/year	
			Personnel	Population
Manganese-54	3800	78000	$1.3 \cdot 10^{7}$	$5.3 \cdot 10^{5}$
Cobalt-60	7700	$2.79 \cdot 10^{5}$	$6.9 \cdot 10^{6}$	$8.3 \cdot 10^{4}$
Cesium-134	2900	–	$2.9 \cdot 10^{6}$	$1.5 \cdot 10^{5}$
Total	14000	$3.57 \cdot 10^{5}$		

TABLE 5. Radionuclide Releases when Cutting Victor II Submarine's Outer and Pressure Hulls

Radionuclide	Release, Bq	Annual limit, Bq/year
Ferrum-55	1900	$5.4 \cdot 10^{7}$
Nickel 59	0.15	$1.5 \cdot 10^{8}$
Nickel-63	17	$4.5 \cdot 10^{5}$
Cobalt-60	170	$6.9 \cdot 10^{5}$

Results of calculations of individual doses for population are given in Table 6.

TABLE 6. Effective Population Exposure Dose Under Normal Condition of Victor II Dismantling, mSv

Distance, km	Dose rate from cloud	Dose rate from inhalation	Dose rate from surface	Total
0.1	5.8×10^{-6}	7.7×10^{-5}	3.3×10^{-3}	3.4×10^{-3}
0.2	4.5×10^{-6}	6.0×10^{-5}	2.6×10^{-3}	2.7×10^{-3}
0.5	1.6×10^{-6}	2.1×10^{-5}	9.0×10^{-4}	9.2×10^{-4}
1	5.4×10^{-7}	6.4×10^{-6}	2.8×10^{-4}	2.9×10^{-4}
2	1.8×10^{-7}	1.8×10^{-6}	7.9×10^{-5}	8.1×10^{-5}
3	9.2×10^{-8}	8.6×10^{-7}	3.7×10^{-5}	3.8×10^{-5}

According to carried out calculations effective annual dose related to radionuclide escape into the environment at normal dismantling procedure, will be 3.4 μSv for population. Maximum effective dose of radiation will not exceed 1.0 μSv for Severodvinsk population at the boundary of sanitary protection area (500 m from radioactive release point). Specified radiation doses are less than annual radiation dose, caused by natural radiation background.

2. Assessment of Radiation Effects in Case of Possible Accidents during NS Dismantlement

List of the design basis accidents is given in Table 7.

List of beyond the design-basis accidents as a result of external effects is given in Table 8.

THE DESIGN -BASIS EMERGENCY SITUATIONS

SNF Container Drop

Made calculations show that maximum effective doses will be: for personnel – 1.3 μSv, for population of Severodvinsk at the boundary of Sanitary Protection Area (500 m from radioactive release place) – 0.12 μSv. It is considerably less than the population dose limit (1 mSv) under normal operational conditions in accordance with Radiation Safety Standards (NRB-99).

TABLE 7. List of the Design Basis Accidents

Initial events	**Assessment of effects**
1 Fall of container with spent nuclear fuel	Assembly destruction with cumulative activity escape of gas products and iodine isotopes.
2 Damage of the assembly in the process of SNF unloading	Damage of assembly and escape of cumulative activity of gas products and iodine isotopes.
3 Fire in the compartment in the process of SNF unloading	Escape of activity in case of water evaporation
4. Unauthorized discharge of coolant of the main coolant circuit in the water area (to 0,5 t)	Escape of activity collected in the coolant

TABLE 8. List of Beyond the Design-Basis Accidents

Emergency situation	Assessment of effects
Spontaneous nuclear reaction	
1 Postulated uncontrolled removal of two lattices in case of reactor filled up with water	Spontaneous nuclear reaction (flashes) with damage of assembly's parts and activity escape
Accidents as a result of external effects	
2 NS flooding in the process of SNF unloading	Activity escape from fuel of unsealed assemblies and as a result of reactor's elements, metal and water shielding tank, and pressure hull corrosion
3 Aircraft fall in the process of SNF unloading	Fire, overheating of assembly in the transloading container with collected activity escape.

Fire in Compartment By SNF Unloading

The calculation of population radiation effects is performed on the basis of different atmosphere stability categories. The estimated maximum annual effective dose for Severodvinsk population at the boundary of the Sanitary Protection Area (500 m from radioactive release point) will be 0.1 μSv. It is considerably less than 1mSv (the population dose limit under normal operation conditions in accordance with Radiation Safety Codes –NRB-99).

Unauthorized High-Pressure Gas Escape by Pressure Release in the Primary Coolant

Calculations show that maximum effective annual radiation doses will be: for personnel – 7.0×10^{-2} mSv, for population of Severodvinsk (500 m from radiation source) – 9.0×10^{-2} mSv.

BEYOND THE DESIGN- BASIS ACCIDENTS

Accident with Spontaneous Nuclear Reaction by SNF Unloading

Such kind of accident should be considered as extremely improbable and postulated.

Gross activity of main radionuclides generated in case of SNR at the moment of flash end will be 1.69×10^{16} Bq, maximum activity after flash will be 1.82×10^{16} Bq. Activity escape into the atmosphere in accordance with radionuclides determining the radiation situation is 1.25×10^{15} Bq.

Effective population exposure doses are given in Table 9.

TABLE 9. Effective Population Exposure Doses in Case of SNR, mSv

Distance, km	Dose rate from cloud	Dose rate from inhalation	Dose rate from surface	Total
0.5	1.9	9.7	26.0	37.0
1	0.54	6.9	18.0	26.0
2	0.12	3.2	8.3	12.0
3	0.051	1.8	4.7	6.5
5	0.017	0.81	2.2	3.0
10	0.0029	0.27	0.70	0.97

Calculation results show that maximum effective annual dose for population of Severodvinsk at the boundary of Sanitary Protection Area (500 meters from radioactive release) will be 37 mSv. So it is not necessary to evacuate people in accordance with NRB-99.

NS Flooding by SNF Unloading

Estimates show that two months of Submarine being flooded is not enough for penetrating corrosion of cladding and activity escape into the atmosphere.

3. Conclusions

1. Maximum gross activity cumulative in NPP of Victor II Submarine will be 1.4×10^{16} Bq (for one reactor) after three years of holding. SNF is the main source of activity value. SNF unloading will lead to threefold reduction of reactor compartment activity.
2. Dose burdens on personnel as a result of gamma radiation during SNF unloading from Victor II Submarine will not exceed 2-3 mSv. It is 10-15% from accepted radiation dose limit for category “A” persons.
3. Maximum annual effective dose for Zvezdochka personnel as a result of radionuclide release to the environment under normal condition of NS dismantling process is 3.4 μSv. Maximum annual effective dose for Severodvinsk population at the boundary of Sanitary Protection Area does not exceed 1.0 μSv. The indicated radiation doses are considerably less than annual dose from natural radiation background.

4. Calculation results of radiation factors analysis show that maximum annual effective radiation dose for population in case of the design–basis accident during Victor II dismantling will not exceed 0.1 mSv. It is considerably less than dose limit for population under normal operational conditions given in Radiation Safety Standards (NRB –99).
5. The most severe radiation effects correspond to the postulated beyond the design-basis accident in case of spontaneous nuclear reaction as a result of one shim lattice removal during dismounting of drives and reactor uncontrolled filling with water. Safety measures, techniques in operation and additional drainage of reactor before SNF unloading exclude the origin of such accident.
6. Calculations show that maximum annual effective population exposure dose at the boundary of Sanitary Protection Area (500 meters from radioactive release point) will be 37 mSv under the hypothetical accident related to spontaneous nuclear reaction. It will not be the necessity to evacuate the population in accordance to Radiation Safety Standards (NRB –99). Collective radiation doses for population will not exceed the annual dose of this region received from natural radiation background.
7. Calculations show that radiation risk for Zvezdochka personnel and Severodvinsk population under probable accidents in the process of Victor II dismantling is considerably less than individual risk limit given in Safety Protection Standards ($5.0\times10^{-5)}$ and negligible risk (1.0×10^{-6}).

WHAT TO DO WITH THE NUCLEAR SUBMARINES WITH DAMAGED CORES?

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Abstract

Some of the early nuclear submarines of Russia suffered accidents, e.g. criticality accidents or LOCAs, whereby the fuel assemblies of the reactor core were damaged and radioactive material released. In the 1960es the reactor compartment of such submarines was disposed of by cutting it out of the submarine and sinking it in the sea. However, since Russia became party to the London Dumping Convention this is no longer permissible. Unfortunately, the alternative, the disposal of these reactor compartments or complete submarines raises very difficult problems. Various approaches have been proposed, and they will be reviewed, but none seems very satisfactory. Since the aim must be to find an approach that will result in an over-all minimum damage to the environment, this raises the question whether some sort of sea disposal would not be the best solution. This question is discussed.

Keywords:

Nuclear submarines, decommissioning, London Dumping Convention, spent fuel, damaged fuel, spent fuel disposal

1. Introduction

The decommissioning of nuclear submarines raises a number of interesting problems, not the least with respect to the handling of the spent fuel of the reactors. This is hardly surprising since the spent fuel contains the major part of the activity of the submarines. There is the problem of possible damage to the spent fuel during handling and how to handle the damaged fuel afterwards. There is the problem of the long-time storage of the spent fuel under conditions that are not always satisfactory. There is the problem of wet versus dry storage of the fuel in large facilities or in transportation casks. And so on. But the most complex problem is undoubtedly the problem of what to do with the submarines, which has suffered a reactor accident whereby fuel assemblies have been damaged. In such cases activity has been released and the fuel can not be removed by

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 361–367.

use of the usually procedure or can not be removed at all. It is this problem that will be considered here.

2. Early Russian Submarines with Damaged Fuel

Some Russian nuclear submarines, in particular those of the first generation, have suffered accidents which has led to fuel damage. Up to the beginning of the 1970es the usual procedure was in such cases that the reactor compartment with the reactor containing the damaged fuel was cut out of the submarine and replaced by a new reactor compartment. The old compartment was dumped close to the Novaya Zemlya in the Kara Sea after being filled with a furfurol-compound to delay and decrease the release of activity to the sea. Two reactor compartments with fuel in one or both reactors were dumped in 1965. In 1972 a submarine reactor with spent fuel, contained in a metal shielding container, was disposed of in the sea close to Novaya Zemlya. Finally, in 1981 a complete nuclear submarine was sunk near Novaya Zemlya. This event ended the dumping in the sea of submarine reactors with damaged fuel [6].

The first three submarine reactors disposed of in this way seems have originated from a Hotel, a November and a Yankee class submarine, all of early designs, and they had all suffered either a criticality or a loss-of-cooling accident. They were all provided with two pressurized water reactors. The submarine was the Project 645 submarine with a November class hull, which was provided with two liquid-metal cooled reactors. It had suffered a loss-of-cooling accident in one of its reactors [4].

3. Later Russian Submarines with Damaged Fuel

After Russian has become party to the Convention of the Prevention of Marine Pollution by Dumping of Waste and other Matter, also called the London Dumping Convention, disposal in the sea of submarines/reactor compartments/reactors with damaged was not the obvious way to deal with the problem. Unfortunately, a few Russian submarines have since then suffered accidents, which involved damage of the fuel. The Northern Fleet seems to have two submarines, which can not be defuelled, an Echo-II class submarine which has suffered a loss-of-coolant accident in 1989 and a Alfa class submarine which has suffered a similar type of accident 1972. The Pacific Fleet seems to have three submarines with damaged cores, two Echo-II class and one Victor-1 class submarines, two of which suffered loss-of-coolant accidents in 1979 and 1985 and one which suffered a criticality accident in 1985. The latter accident is the well-known Chazhma Bay accident [5, 4]. The cores of the five submarines may not be damaged to the same extent, but at least three seem to be heavily damaged and contaminated, and some of submarines are kept floating only with considerable difficulties.

This raises the question: What to do with these submarines with damaged fuel?

4. Proposed Solutions

Various disposal methods for these submarines have been considered [1]. One was to cut the reactor compartments out of the damaged submarines, insert them into sufficiently long sections of the pressure hulls of the missile compartments of ballistic missile submarines and seal the pressure hull sections in both ends. Since the inner diameter of the pressure hull of the missile section is larger then the outer diameter of the reactor compartment of the damaged submarines this would create a sarcophagus around the damaged reactor compartment. Before the reactor compartments were inserted into the sarcophagus the damaged reactors would be filled with furfurol and the space around the reactors with concrete to fix the activity and to provide the necessary radiation shielding. Such a floating sarcophagus could be used for interim storage of the damaged compartment. However, this approach seems to have been abandoned, because its realization would give the personnel involved high radiation doses and because it would only be an interim solution. Sooner or later the reactor compartments would still have to be dismantled.

Another approach involves placing the damaged submarine in a floating dock after the reactor has been properly isolated e.g. with furfurol and towing the floating dock out to a place at sea which has been prepared for the sinking of the dock. After the sinking the dock and the submarine would be filled with concrete and covered with a layer of a sand-gravel mixture the top of which would be covered by a layer of rock to protect the burial place from the effect of weather and human activity. Such an approach, which in essence is burial or entombment at the bottom of the sea will probably reduce the radiation doses to the personnel involved. Its legality depends on the interpretation of the London Dumping Convention.

A third approach discussed by [1] is to excavate a “dry dock” at a proper place at the coast line, sail the damaged submarine properly prepared into the dock and entomb the submarine there under layers of concrete, sand-gravel mixture and hard rock. This method is in many ways similar to the second, but the burial occurs – depending on definitions – on land. This should approach should be in agreement with the London Dumping Convention, but it may be questioned whether entombment in a coastal area, exposed to heavy sea, strong winds and often human activity, is a proper place for an repository. The handling of the submarine is also likely to give significant doses to the personnel.

A fourth approach is discussed by [5]. It involves the possibility of unloading of the core of damaged submarines and dismantling the submarines. Such an approach could easily result in large radiation doses to the personnel involved or become prohibitively expensive if all operations have to be done by remote handling.

None of these approaches seems to be ideal, and they may to a varying degree give significant doses to the personnel involved. So why not think about the unthinkable, sea disposal, even though this approach may seem to be prohibited by the London Convention.

5. Thinking about the Unthinkable

Ignoring for a moment the legal aspects of the disposal of nuclear submarines with damaged cores, the goal should be to find a disposal approach, which will give the overall minimum effect on the environment. This minimum includes the effect of the release of activity to the environment during floating storage and disposal and the effect of the doses received by the navy and shipyard personnel who will have to handle the submarines. These people are part of the environment.

Sea disposal could be carried out in the following way, similar to the approach used during the Soviet period as discussed in section 2. The reactor vessels and the primary circuits are drained for water and filled with a material like furfurol or a low melting point metallic alloy to fix the damaged fuel and the control rod and to decrease the release of activity to the environment. If need be, additional material, e.g. concrete may be added to the reactor compartment to reduce the radiation level around the submarine. At the same time the floatability of the submarine must be ensured. After these preparations the submarine is transported out to the sea to a proper place, where the submarine is disposed of by sinking. Depending on the state of the submarine, it may by towed to the disposal area or it may have to be transported in a floating dock or by some other means.

The disposal area has to be selected in such a way that the disposal of damaged submarines at the site will affect the environment and human activity such as fishing and shipping as little as possible. It has to be placed at an isolated site with little sea current. For the Northern Fleet it would be reasonable to perform the disposal in the Kara Sea e.g. near Novaya Zemlya since a number of reactor compartments and a submarine has already been sunk in this region. It may be more difficult to find a proper disposal area for the damaged submarine of the Pacific Fleet. Japan may not like the use some place in the Sea of Japan as the disposal area and the use of the Sea of Okhotsk would mean a fairly long transportation route, since the damaged submarines are floating at naval bases not far from Vladivostok at the coast of the Sea of Japan. It should also be mentioned that the approach is not without its risks since some of the submarines with damaged cores have buoyancy problems and could sink at unfavorable positions during the transport to the disposal site.

Measurements have been made by a Joint Norwegian-Russian Expert Group of the release of activity at the disposal sites for reactor compartments at Novaya Zemlya. Surface sediment samples at the sea bottom taken close to the dumped reactor compartments and the sunken submarine showed in a few cases activities of up to about 10 kBq/kg, but in most cases the activity of the samples was 100 Bq/kg or lower. At some distance from the dumped objects the level of radionuclides in sediments was close to the range obtained in the open Kara Sea [2]. It may also be mentioned that the activity of fish from the Baltic, measured in Bq/kg is more than a factor of 10 higher than the activity of fish from the Barents Sea. Thus the contamination caused by the dumped reactor compartments has so far been quite limited. Corrosion of the seawater will of course gradually increase the release of fuel and activated materials to the environment, but at the same time the activity will decrease due to radioactive decay

Further the radionuclides seem to be present mostly as particles, not dissolved. This seems to indicate that the contamination will remain local and that the concentration of radionuclides in the seawater will be limited.

These considerations seem to indicate that sea disposal of submarines with damaged cores will give the lowest doses to the personnel involved and that its effect on the sea environment will be quite limited. So sea disposal has the great advantage that it will reduce the certain environmental effects, the personnel doses, while it may increase the more hypothetical, uncertain environmental effects, long term release of small amounts of activity to the sea. This raises the question: Is it not so that sea disposal, carried out in a proper way, will give the lowest overall effect on the environment?

6. The London Dumping Convention

So far it has been assumed that sea disposal is not permitted by the London Convention, which was adopted in November 1972 and came into force in August 1975. The USSR/Russia became party to the convention in January 1976. But what does the Convention really prohibit?

According to Article I of [3] the aim of the convention is "to prevent the pollution of the sea by dumping of waste and other matter that are liable to create hazards to human health, to harm living resources and marine life, to damage amenities or to interfere with other legitimate uses of the sea". Dumping is in Article III defined as "(i) any deliberate disposal at sea of wastes or other matter from vessels, aircraft, platform or other man-made structures at sea" and " (ii) any deliberate disposal at sea of vessels, aircraft, platforms or man-made structures at sea". Article IV states that "the dumping of wastes or other material listed in Annex I is prohibited", and Annex I includes "Radioactive wastes or other radioactive matter". From these Articles it seems obvious that the Convention applies to nuclear submarines.

However, the Convention is not without exceptions. Article V states, that "The provision of Article IV shall not apply .... in any case which constitutes a danger to human life .... if dumping appears to be the only way of averting the threat and if there is every probability that the damage consequent upon such dumping will be less than would otherwise occur. Such dumping shall be so conducted as to minimize the likelihood of damage to human or marine life and shall be reported forthwith to the Organization".

Since the approaches for disposal of damaged submarines discussed in section 3 of this paper constitutes a danger to human life (significant doses to the personnel), since dumping appears to be the only way of averting this threat and since there is as discussed in section 4 every probability that the damage caused by such dumping will be quite limited, it seems quite clear that the London Dumping Convention does not prohibit the dumping of submarines with damaged fuel. However, such dumping must of course be conducted in such a way as to minimize the likelihood of damage to human and marine life and shall be reported to the London Dumping Convention Organization. Before doing so the Party undertaking the dumping shall according to Paragraph 2 of

Article V consult any other country or countries that are likely to be affected by the sinking as well as the Organization. Further the Organization shall after consulting other Parties and international organizations promptly recommend to the Party the most appropriate procedures to adopt. The Party shall follow these recommendations to the maximum extent feasible.

The 1996 Protocol to the London Dumping Convention contain in its Article 8 the same exception as in the Convention [7].

So if the procedure outlined in the London Convention is followed, sea disposal of submarines with damaged cores should not represent a violation of the Convention. It should be emphasized that the exceptions listed in Article V applies only to submarines with damaged cores which constitutes a real danger to human life, not to the ordinary decommissioning of nuclear submarines. This also means that the use of Article V for submarines with damaged cores will have limited consequences and that its use here will in no way undermine the Convention.

7. Conclusions

As discussed above the London Dumping Convention does not seem to prevent the dumping of submarines with damaged fuel provided it in done in a proper way and in consultation with the Convention Organization and the countries likely to be affected by the dumping. A possible approach might be to start the process by the establishment of an international working group to analyze the risks involved in the various approaches, to document that dumping is the safest approach and to outline the most appropriate procedure to be used in connection of the disposal of submarines with damaged cores. After that, the formal procedure of consulting the Organization and other Parties can be started.

8. Personal note

The author of this paper would like to state that this paper contain his personal opinions as a scientist and as a person concerned about the risks involved in the decommissioning of nuclear submarines with damaged fuel. It does not in any way claim to represent the opinion of any official institution or authority in his country.

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ESTABLISHMENT OF AN AUTOMATED DRY STORAGE FACILITY FOR SPENT NUCLEAR FUEL OF DISMANTLED NUCLEAR SUBMARINES AND NUCLEAR-POWERED SURFACE SHIPS IN GEOLOGICAL STRUCTURES OF THE NORTHWEST RUSSIA

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Adequate solution of the challenge of Spent Nuclear Fuel (SNF) management is one of the key goals and objectives of the Russian Federation in the immediate future.

In compliance with the “closed fuel cycle” concept accepted in Russia, SNF unloaded from Nuclear Submarines (NSs) with water-cooled reactors (VVER) as well as SNF of nuclear icebreakers are forwarded to PA “Mayak” for reprocessing, fuel of liquid-metal coolant reactors, zirconium-cladding fuel and damaged Spent Fuel Assemblies (SFAs) being the only exceptions.

NS defueling and SNF shipment for reprocessing are the components of NS complex decommissioning process. By now SNF forwarding for reprocessing has been suspended mainly due to limited reprocessing capacities of Russian nuclear industry enterprises and lack of engineering and financial resources for SNF shipment and arrangement. On the other hand, SNF storage on board of taken-out-of-service afloat-stored NSs as well as within overfull coastal storage facilities does not comply with the present-day safety requirements. Thus today the necessity of establishing an up-to-date interim SNF storage facility is obvious.

At present construction of large-capacity and stable (in geomechanical, hydroengineering and heat-transfer respects) storage facilities represents a topical task.

According to recommendations of the International Agency for Atomic Energy (IAEA), three types of geological formations are suitable for deep underground storage of Radioactive Waste (RW) and SNF:

– igneous and metamorphic rocks;
– clays; and

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– rock salts.

Each of the above geological formations has its virtues and shortcomings. However, when selecting an optimal option, “safety” and “cost efficiency” should be the main criteria.

Based on this postulate, the authors put forward an option of SNF long-term storage in natural geological formations (hills) or artificial pits covered with a thick layer of ground (covering) allowing ensuring safe storage conditions due to natural thick of rocks/ground. This is a cost-efficient option for in such a case construction of resource-demanding storage facilities would not be necessary any more.

After examination of all RW and SNF dry-storage options developed so far in Russia and abroad, the authors have come to the following conclusion: non-containerized horizontal storage of SFAs in shrouds in good natural-convection conditions represents the best variant. The proposed storage concept is an alternative to the container-storage option, and its economic expediency is obvious. By way of example, let us make some simple calculations: the cost of one 40-t container for SNF developed under AMEC Program is $150 000, its service life being 50 years. To defuel one NS, 4 to 12 such-type containers are necessary. According to the statement at CEG IAEA meeting (November 2003) of Mr. Akhunov, Head of Department at Rosatom, of 192 NSs taken out of military service by that time, 100 NSs remained to be dismantled. Accordingly, only the cost of containers to store SNF of 100 to-be-dismantled NSs would require $180 million (or 5400 million rubles - an appreciable sum for the Russian Federation’s Budget).

The proposed engineering solutions would considerably diminish the costs for construction and running of SNF storage facility with no decrease in safety.

The collected so far experience of running of dry storage facilities for SFAs has been evidence of important heat-exchange problems. Because no country has managed to obtain a precise analytical solution of the heat-exchange issues yet, when establishing SFA storage facilities, experimental test benches are widely used.

If SFAs are stored in containers, chambers or reinforce-concrete cells, attaining of uniform natural convection is hardly possible due to high SFA-arrangement density per 1 m^2 (case of vertical storage). Moreover, according to experience of FRAMATOM Company (France), in case of horizontal chamber-type SFA storage forced ventilation is required too. The SNF storage option put forward in this paper would be free of ventilation problems.

The proposed design of SNF storage facility comprises (Fig.1):

– storage facility itself;
– local railway/motor road (similar to SKB in Sweden);
– transport and management equipment including: crane, car truck with distributing container, shielding box and special bed;
– control station; and

– physical protection system and a post control.

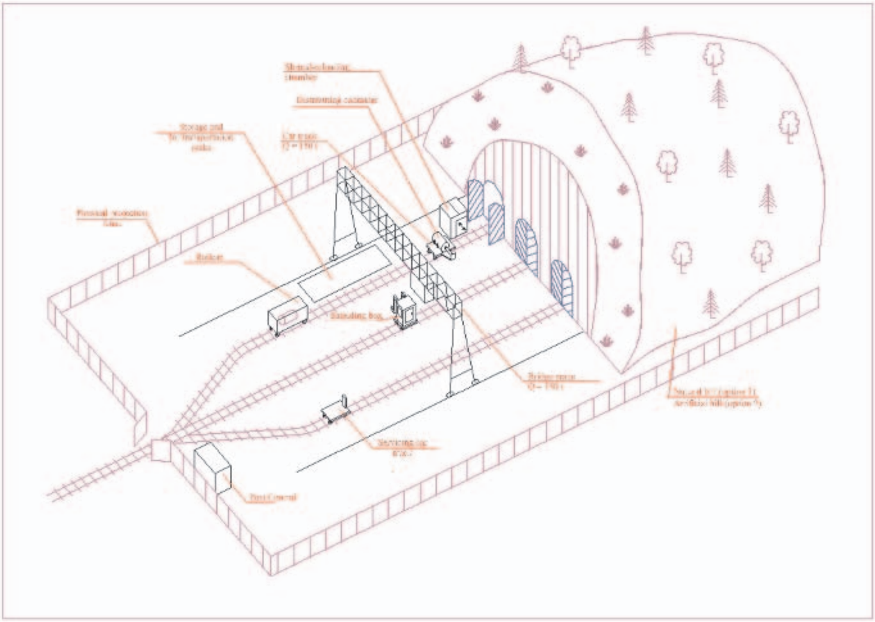

Figure 1. Layout of automated dry storage facility for SNF of dismantled NSs and nuclear-powered surface ships

The storage facility represents a tunnel supported by lightweight reinforced-concrete constructions made on basis of BelgTASM concrete mixes providing for reliable biological shielding.

The basement and vertical bearing plates with special beds for SFA shrouds are mounted on a sand-gravel saddle. Along bearing plates rails are laid to move a car truck loaded with shielding container with SFA-housing shroud.

The car truck comprises a platform with a lift table and a hydraulic pusher to push shroud out of container.

The car truck is also equipped with: a power supply source, an observation system, control equipment including power automatic machinery and a transceiver (another option is also possible of electric-power transmission using trolley or a rail with pulse-code modulation of steering commands).

A special bed is installed at the storage facility entrance to reload shroud from transportation cask to distributing container.

There is also a special shielding box with manipulator to be used in a case of potential emergency or during scheduled preventive works. The shielding box is installed onto a servicing car truck having a possibility of independent travel on rails.

The box has viewing systems, independent drive, transceiver and hitch-mechanism for potential emergency situations.

The storage facility has a labyrinth air pipe with possible access to filters arranged on a spiral path directed to the hilltop. Such a design solution would create natural air convection from the heat source to lower temperature areas.

Permanent temperature constituent inside the hill would make it possible to determine the between-shroud temperature difference. Throttling control in air pipe would allow adjusting the temperature balance.

Transport and management equipment of the storage complex comprises a gantry crane to extract transportation cask (TUK) from railcar and place it inside a reloading chamber. Transfer of the car truck and the shielding box from one rail track to another is also performed using this crane.

The storage facility physical protection system represents a variety of fencings, signal and visual observation equipment, anti-ram devices and equipment of radiation checkout at railcar (trailer) entrance/exit points.

It is expected that the post control will be combined with the facility control room, physical protection system and all running-and-survival systems.

Maintenance staff of the entire storage complex is estimated at 6 people, the repair and support units being called as necessary.

The following storage facility running mode is proposed.

First, special train (trailer) with SFAs enters the storage facility site and undergoes external radiation examination. Next, the train unloading by gantry crane begins. After extraction, transportation cask is placed onto a special reloading pan. Then the transportation cask lid is removed, SFA-containing shroud is extracted and reloaded into the distributing container. After that special gates are opened, and the car truck loaded with the distributing container enters the facility tunnel (Fig.2). SFA-containing shroud is delivered to an automatically selected free space, and after full stop of the car truck reference control by the guiding system is fulfilled. Later on a command for SFA shroud delivery onto storage facility bed is coming. The bed is equipped with special rollers to move and fix SFA-containing shroud. Finally, when the control system has fixed the shroud position, the car truck is returned for the following shroud.

It is in such a way that the storage facility is gradually loaded. Unloading and removal of shrouds is performed in the reverse order.

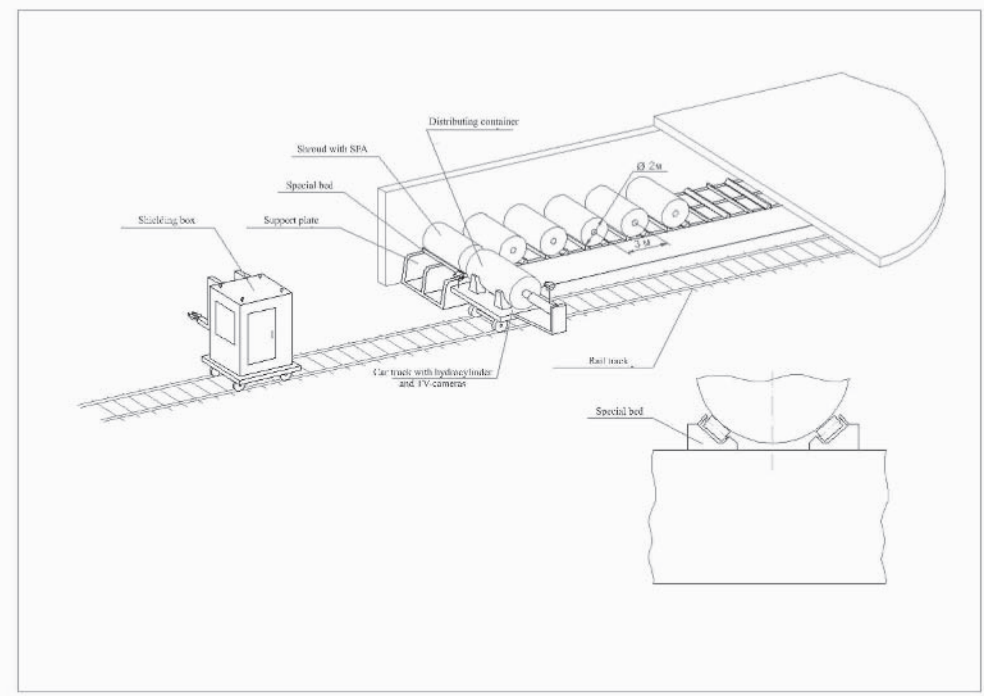

Figure 2. Scheme of SNF Loading into Storage Facility

Unloading of the remaining railcars of special train would be performed simultaneously to prevent the train delaying. Transportation casks with SFAs would be placed at a special transient-storage pad, whereas the special train would leave for the following container lot. After arrival of the following trains and unloading of transportation casks with SFAs onto the transient-storage pad, empty casks would be loaded into the train and send back.

Thus the proposed storage facility option has the following advantages:

1. SFA storage without containers
2. Natural convection
3. Saving of constructional and running expenses
4. Use of natural physical-protection barrier
5. Automated procedures of SFA loading/unloading, accounting and control.

Along with the SNF storage-relates challenges, the problems of long-term storage of high-active and medium-active radioactive waste are presently also rather acute. The above-proposed SNF-management flowsheet provides for a possibility of similar handling of RW. If necessary, high- and medium-active RW could be fragmented inside an enclosed box down to a size acceptable for loading into shrouds. Filling of shrouds could be performed at a hot chamber by special reloading machines (manipulators). RW delivery to the storage complex also in transportation casks is proposed. All subsequent operations with RW packed into shrouds could be performed in a similar way. High- and medium-active RW could be stored at the complex until a possibility would emerge on their forwarding for reprocessing. The proposed solution presents a rather useful option for RW management in Andreeva Bay and Gremikha.

The above option could be especially advantageous for long-term storage of spent-removal units with cores previously filled with lead, radiation-resistant polymer, solid-phase dispersed material or silica sand.

NUCLEAR SUBMARINES WITH DAMAGED REACTOR INSTALLATIONS: BASIC ENGINEERING SOLUTIONS AND SAFETY-RELATED PROBLEMS

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Hazardous (inadmissible) radiation situation inside Reactor Compartments (RCs) represents a distinctive characteristic of the damaged Nuclear Submarines (NSs) giving no way of performing pre-defueling operations using standard technologies. To conduct works in RCs of the damaged NSs and perform their rehabilitation, one needs developing individual technologies and special devices taking into account specific implications of the accidents, which caused withdrawal of every such NS from service. If observing special conditions, defueling and subsequent dismantlement of some damaged NSs is possible even under their present-day technical and radiation situation; at the same time defueling and dismantlement of other damaged NSs is still impossible. This means that presently one has to do with the following two main problems:

– defueling of nuclear reactors of damaged NSs; and
– RC bringing to environmentally safe condition.

To resolve these challenges, many investigations have been performed so far used as a basis for elaborating principal approaches and solutions, as applied to every damaged NS.

Of the whole amount of retired NSs four NSs were withdrawn from service due to incidents related to failures in heat removal from reactor fuel assemblies while in operation (*Echo-II*-class NSs #533 and #541; *Victor-I*-class NS #610 and *Alpha*-class NS #900) that resulted in drastic worsening of the radiation situation in their RCs. One more NS (*Echo-II*-class #175) was taken out of operation due to a nuclear accident during its repair at shipyard in Chazhma Bay.

In 1999 using special engineering procedures pre-defueling operations and defueling of the reactors of damaged *Echo-II*-class NS # 533 were performed. After unloading of Spent Fuel Assemblies (SFAs) the inner reactor space was filled with hardening F-class preserving substance according to a special procedure developed at NIKIET. Later on a three-RC unit was made up of that NS which is presently stored within the Temporary Storage Center (TSC) in Saida Bay.

Similar emergency situation with partial depressurization of SFA canisters and the primary circuit occurred at NS #541. Based on the results of radiation and engineering

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examination some additional measures were developed to ensure the NS defueling in compliance with the nuclear and radiation safety standards in force. In 2004 the NS #541 is to be transferred to FEP “Zvezda” for defueling on a solid basement and subsequent making up of a three-RC unit.

New engineering solutions and principal rehabilitation technologies have been developed for three damaged NSs which defueling is still a problem. One of such NSs – *Alpha*-class #900 – has been already transferred to environmentally safe condition:

- liquid-metal coolant was partly drained from the reactor, the remaining coolant being “frozen”;
- free space inside the reactor was filled with hardening F-class preserving substance;
- reactor was sealed using standard upper head;
- shim rods were placed in such a way as to exclude reactor coming up to power; and
- the RC area around the reactor was bituminized.

Special buoyancy tanks were attached to RC to ensure its waterborne storage (Figure 1).

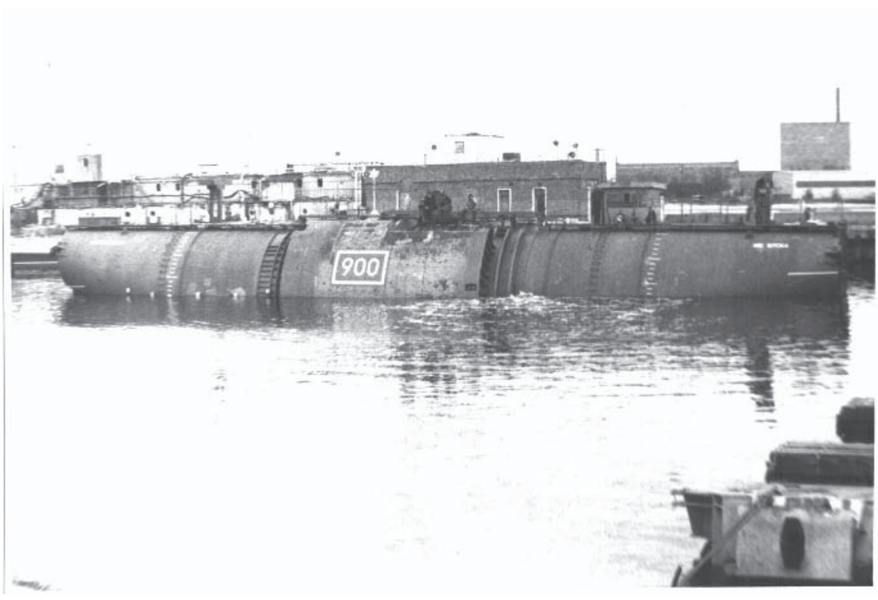

Figure 1. RC unit of damaged NS #900

Presently the RC is stored at TSC in Saida Bay (Figure 2). Further program of the RC management has not been developed yet.

Figure 2. Temporary waterborne storage of RC#900 in Saida Bay

The remaining two damaged NSs –#175 and #610 – have been stored afloat in the Pacific Russia since 1985 (Figure 3).

Figure 3. NS #175 (left) and NS #610 (right) on the berth within a waterborne storage center

One can see that the following operations have been already performed at these NSs:

– dismantlement of missile launchers; and
– dismantlement of some non-radioactive equipment.

Works on both elimination of the accident implications and RC decontamination have been also performed at these NSs. As the result, the radiation situation inside and outside RCs has slightly improved but still remains extremely hazardous and inadmissible. Some data illustrating the results of radiation examinations of the damaged NSs #175 and #610 performed in 2002-2004 are demonstrated in Table 1.

TABLE 1. Radiation situation inside and outside RCs, mSv/h

Gamma radiation in rooms	NS #175		NS #610	
	Measurements of 1996	Measurements of 2003-04	Measurements of 1999	Measurements of 2004
Reactor part of RC (on reactor upper head)	up to 250	50-180	up to 60	
Through-pass corridor	up to 100			
No-go corridor	up to 60			
Pump part of RC	up to 2.5		up to 5	
Adjacent compartment bulkheads	up to 0.3	up to 0.375	up to 2	up to 0.9
Light hull above RC		up to 1	up to 0.9	up to 0.44
Strong hull		up to 5	up to 5	up to 1.45
Main gamma radiation nuclide	^{60}Co		^{137}Cs	

From the Table 1 data it follows that despite some decrease in gamma levels, they are still extremely hazardous.

To exclude hypothetical incidents during temporary waterborne storage of damaged NS #175 and NS #610, appropriate measures were taken on ensuring their buoyancy and floodability.

In 1991 the NS #175 underwent an in-dock repair (all outboard openings of the strong hull were welded). In 1996 in-dock repair (sealing) of NS #610 was performed.

Considering that standard NS buoyancy equipment is presently in poor condition due to lack of servicing, scheduled repairs and corrosion processes, the Pacific Fleet’s personnel uses basic buoyancy devices to ensure floodability of these NSs (installation of pontoons, filling of driving ballast tanks with polystyrene).

Nuclear safety of reactors of the damaged NSs is ensured through the application of emgineering and organizational measures excluding extraction from (displacement in) the core of control rods and shim rods (no way for power supply to actuating mechanisms of the control and protection system, application of mechanical stops, impossibility of using hand drives).

However potential hazard of initiating radioecological incidents while NS storing afloat still persists, and its probability increases steadily. Thus the problem of environmental remediation of the damaged NSs #175 and #610 needs an adequate solution in the near future.

To select the safest and the most economically efficient option, Feasibility Studies (FSs) of potential engineering solutions on managing the damaged NSs under consideration were performed. For example, an alternative was considered providing for arranging both NSs within a taken-out-of-operation floating dock, its subsequent placing in shallow water and filling with concrete. Another option developed in more detail including designing of project and engineering documentation provided for placing of RC of NS #175 into a cylindrical steel case. As such case, two compartments of already dismantled *Yankee*-class or *Delta*-class NS were proposed (that project was known under the "*Sarcophagus*" conventional name).

However neither of the above options has been implemented because of doubts of some experts on safety of long-term waterborne storage of the objects under consideration.

To select the most appropriate option, the following condition was put in the forefront: any engineering solution on rehabilitation of the damaged NSs had to comply as far as possible with the environmental-safety requirements in the region during long-term storage of nuclear- and radiation-hazardous packages. With due regard for that thesis, in 2001 a joint inter-agency decision was taken on selecting an option providing for confinement of the non-defueled damaged NSs #175 and #610 within a land-based storage facility above the sea level. On the instructions of the Russian Federal Agency for Atomic Energy (Rosatom, former Minatom) project documentation is being actually developed (the first stage) on construction of a land-based facility in Razboinik Bay (Primorskiy kray) to confine the damaged NSs under consideration (Figure 4). Such storage facility is to be established at Ustrishny Cap nearby the Long-Term Storage Center (LSC) for RCs.

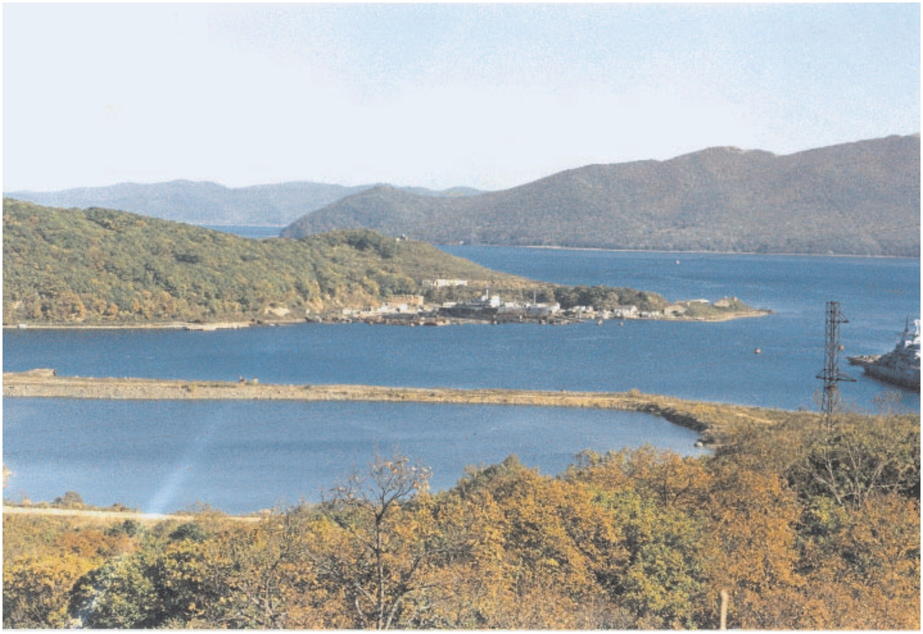

Figure 4. Razboinik Bay

Due to insufficiency of initial radiation data in other (save for RC) NS compartments and outside NS, at the first stage of the project development the general FS provided for placing of both virtually non-dismantled damaged NSs into the land-based confinement (only deckhouse barriers and stern ends were to be removed).

In June 2004 with authorization of the Main Technical Department of the Russian Navy specialists of "Expert-center" Research and Engineering Office together with NIKIET performed a detailed radiation examination inside and outside all compartments (save for RC) of NSs #175 and #610 in compliance with a specially-developed and approved program. Dose control and spectrometric measurements allowed revealing the following: maximal exposure dose rates were measured in adjacent-to-RC compartments on bulkheads from RC side (0.1-0.38 mSv/h for NS #175 and 0.83-0.9 mSv/h for NS #610).

At 2-m distance from the bulkheads the dose rate decreases by 20-30 times. In other compartments gamma radiation dose rate is considerably less: 0.01 mSv/h in case of NS #175 and 0.002 mSv/h in case of NS #610.

On the outer NS surfaces maximal gamma radiation dose rates were measured above the reactor compartments (0.5-1.5 mSv/h on the light hull and 1.5-5.0 mSv/h on the strong hull).

At NS #175 local radiation sources above other compartments were also revealed mainly due to specific effects of RC. The results of performed measurements are generalized in Table 2.

TABLE 2. Generalized results of gamma dose rate measurements, June 2004

NS #175					NS #610				
Compartment number	Mean dose rate, µSv/h	Mean β-contamination, 1/min ·cm^2	Mean dose rate on light hull, µSv/h	Mean dose rate on strong hull, µSv/h	Compartment number	Mean dose rate, µSv/h	Mean β-contamination, 1/min ·cm^2	Mean dose rate on light hull, µSv/h	Mean dose rate on strong hull, µSv/h
I	2	≤250	15.0	300.0					
II	<9	≤250	15.0	30.0					
III	1,5	≤20	2,5	10.0					
IV	1,2	≤100	15.0	15.0	I	0.14	≤20	0,8	0.7
V	9-10	≤250	40.0	20.0	II	23.0	≤300.0	6.2	5.0
VI	RC				III	RC			
VII	1,5	≤20	2.0-40.0	5.0-10.0	IV	30.0	≤450	1.6	2.0
VIII	0,1-0,4	≤20	1.0	4.0	V	1,4	≤180	1.0	1.8
IX	0,5	≤20	0,8	0,5	VI	0,15	≤20	0.3	0,3
X	0,5	≤20	0,5	0,5	VII	0,14	≤20	0,2	---

Based on the results of 2004-year measurements, the following main conclusions may be yielded:

- the adjacent-to-RC compartments of both NSs have local contamination sources (mainly on passageways and under input hatches) which levels exceed the admissible values. Maximal values were measured at bulkheads on the RC side: up to 0.4 mSv/h at NS #175 and up to 0.9 mSv/h at NS #610. The average exposure dose rate values within these compartments reach 0.01 mSv/h at NS #175 and 0.03 mSv/h at NS #610;
- there are also local contamination sources in other compartments but with considerably lower radiation levels: ≤0.01 mSv/h at NS #175 and ≤0.002 mSv/h at NS #610. The mean dose rates in compartments vary within 0.0005-0.002 mSv/h in case of NS #175 and make up 0.0015 mSv/h at NS #610;
- outside the NSs maximal gamma dose rates were recorded above the reactor compartments: up to 5 mSv/h on the strong hull (up to 1 mSv/h on the light hull) at NS #175 and up to 1.5 mSv/h on the strong hull (up to 0.45 mSv/h on the light hull) at NS #610. Increased radiation levels were also recorded above the adjacent-to-RC compartments: 0.015-0.058 mSv/h above the light hull and 0.02-0.54 mSv/h above the strong hull of NS #175 and 0.0002-0.006 mSv/h above the light hull and 0.0002-0.001 mSv/h above the strong hull of NS #610;
- some compartments are contaminated by beta-emitting nuclides. The density of beta-particle flux does not exceed 450 particle/min·cm^2;
- to perform rehabilitation of NSs #175 and #610, removal (confinement) of local radiation sources from (in) NS compartments, save for RC, and the hull is necessary. Restrictions should be imposed for personnel staying above the reactor compartments;
- if the radiation safety standards and requirements in force are observed, the actual radiation situation in end compartments allows performing works on making up three-RC units out of NSs #175 and #610 followed by their placing into the land-based confinement.

Reducing of the mass and dimensions of the objects to be confined would allow decreasing considerably the size and cost of the land-based confinement and diminish the load on equipment involved into transportation and installation of the fragments of damaged NSs #175 and NS #619 for long-term storage.

According to special proposal of “Rubin” Central Design Bureau for Marine Engineering, three-RC units of both damaged NSs are to be made within a floating dock on a special pontoon-basement on which they are to be transported to the land-based confinement. RC transfer to the slipway is to be also performed using the above pontoon-basement. After the pontoon-basement installation on the slipway, it is to be slightly submerged, fixed on the sea bottom and filled with concrete in order to fulfill the function of a special “basement” for RC units; the pontoon touchdown on the sea bottom is to be performed in such a way as the RC units be above the sea level. A schematic diagram of RU arrangement within the land-based confinement is demonstrated in Figure 5.

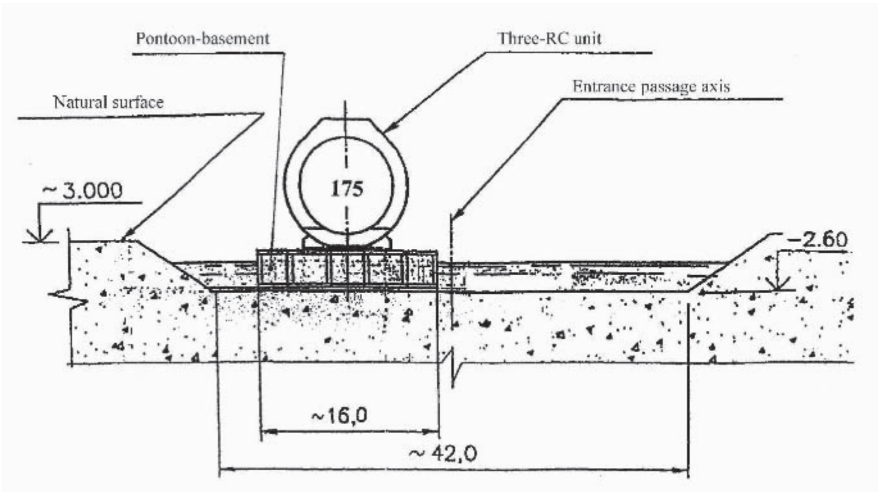

Figure 5. View of reactor compartment unit on pontoon-basement

While making up the three-RC units, one needs to prepare both the reactor installations and RCs for long-term storage (coolant removal, installation of additional shielding outside the compartments, etc.). To do this, one needs developing an appropriate project and procedure of RC preparing for storage including selection and justification of the modes of their mothballing and protection against corrosion.

An example of possible land-based confinement for three-RC units of the damaged NS #175 and #610 is demonstrated in Fig. 6.

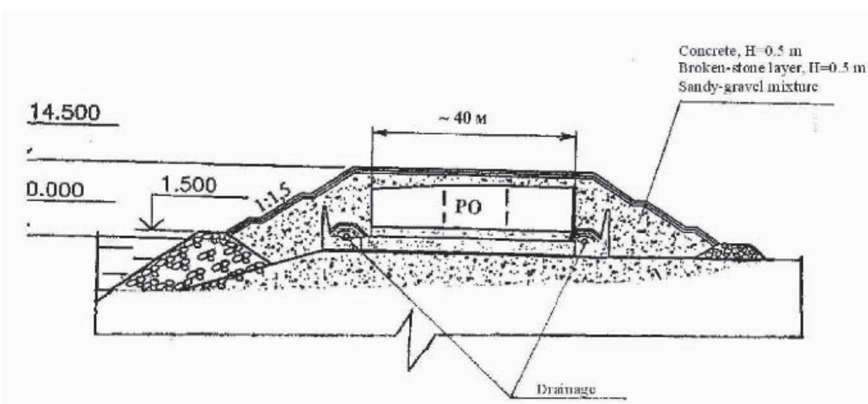

Figure 6. Land-based confinement

Thus to implement the remediation plan for the damaged NSs #175 and #610, one needs:

- developing appropriate designs for making up three-RC units of the damaged NSs #175 and #610 including the radiation-safety measures;
- developing project documentation for construction of the pontoon-basement;
- developing a design for installation of pontoon-basements with RC units into land-based confinement;
- determining the procedure and develop appropriate corrosion protection technology for outer surfaces of the RC units;
- manufacturing pontoons-basements; and
- constructing the land-based facility to confine the three-RC units.

Because implementation of the whole project will take several years, all the related activities should be included into a list of top-priority works since the hazard of further afloat storage of the damaged NSs increases every year. It is also obvious that under the present-day level of the project financing from the Russian Budget funds, its implementation would last for many years. Thus the project support by the whole international community and by individual countries, especially in the Pacific region, would considerably contribute to acceleration of the resolution of this topical problem.

APPLICATION OF LASER BEAM TECHNOLOGIES TO DECONTAMINATE EQUIPMENT WHEN DISMANTLING NUCLEAR SUBMARINES

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Laser beam technologies can be applied with major efficiency during Nuclear Submarine (NS) dismantling operations for: -cutting of metal constructions; -welding of boron-containing-steel containers to store nuclear fuel; and -decontaminating NS units and assemblies.

Since 2001 "Prometey" Central Research Institute for Constructional Materials, "Safety" Close Corporation and Saint-Petersburg Technology Institute (University), Saint-Petersburg, Russia, have been developing laser decontamination technologies.

The following requirements were formulated for the decontamination process:

- High efficiency
- Removal of solid-phase surface radioactive contamination with no liquid radioactive waste generation
- No environmental contamination
- Ability of decontaminating sophisticated-geometry details.
- Remotely-operated process to minimize occupational radiation effects
- Equipment mobility and
- Possibility of decontaminating inner surfaces of tubes.

The work was performed at three phases:

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 385–390.

1. Equipment selection and debugging of an oxide-layer-removal technology (2001).
2. Manufacturing of a "hot chamber" to operations with contaminated objects (2001-2002) and
3. Development and manufacturing of a mobile laser prototype (2003).

At present CO2- lasers of 10.6-µm wavelength and solid-state lasers of 1.06-µm wavelengths are used in industry most often. To perform works, solid-state lasers were selected owing to their compactness and the possibility of adjusting time parameters of laser emission within a wide range.

It was expected that, because the contamination products were concentrated within the surface oxide layer, removal of the latter would eliminate radioactive contamination. Steel samples with oxide layers up to 100 µm in width were processed [1].

An installation with vertical laser beam issue was used in work (Fig.1). The studied samples were displaced using a positioning table.

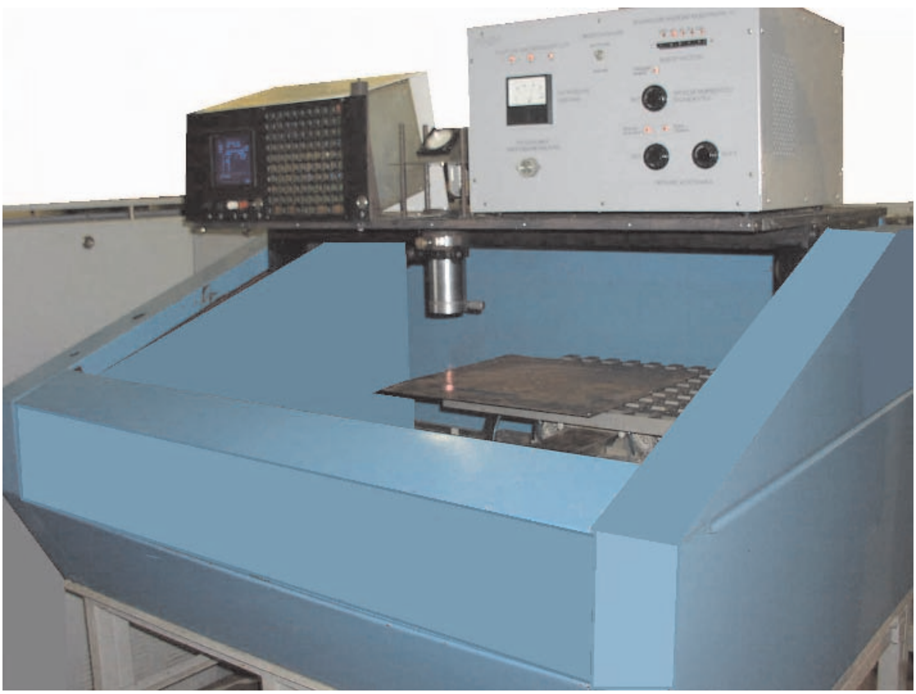

Figure 1. Fig. 1 Installation with vertical issue of laser beam

The studied time parameters of laser emission varied from continuous mode to nanosecond pulse-length range. According to theoretical estimates [2] confirmed by experimental investigations, thin oxide layers can be removed from surfaces most efficiently under the impacts of nanosecond-length pulses. It is the mechanism of thermal-impact oxide-layer removal that is realized in such a case due to rapid heating/cooling of the processed oxide layer. “Fireworks” of luminescent decontamination products are observed during the process. The length of jet above the surface of processed detail reaches several centimeters being directed toward laser emitter. Particles from several microns to millimeters in diameter have asymmetric shape.

As the result of performed investigation the following peculiarities of the oxide-layer-removal process were discovered:

1. Oxide layers are removed under the impact of laser beam 5- 7 mm in diameter; as distinct from cutting and welding procedures wherein concentrated beams 0.1-0.3 mm in diameter are applied; oxide layers are removed under one-pulse impact allowing attaining acceptable decontamination rates (e.g., at 50 Hz pulse-frequency the decontamination capacity can reach 3-5 m^2 per hour).
2. The process is performed at several-meter distance between the emitter and the surface to be decontaminated depending only slightly on its relief.
3. Decontamination of surfaces is also possible at glancing angles of laser beam incidence to surfaces of processed details (Fig. 2).

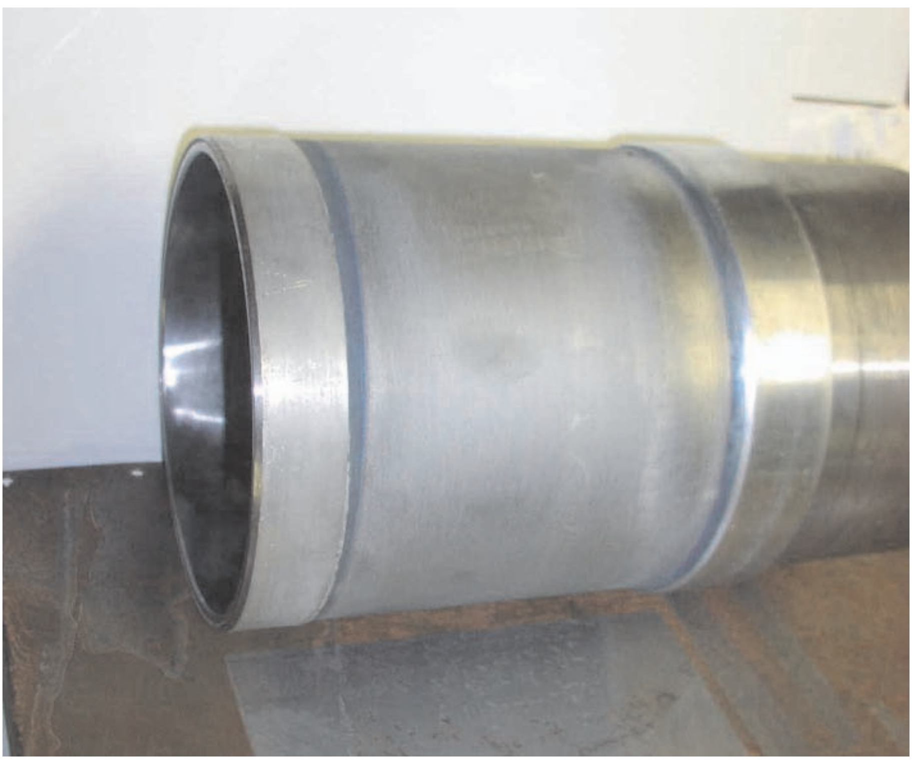

Figure 2. Decontamination of surfaces at glancing angles of laser beam incidence

When using laser emission of higher-length pulses (within microsecond and millisecond ranges), as well as under continuous laser operation mode both metal surface melting and oxide layer "backing" into the melt are observed.

Non-admission of environment contamination is one of principal requirements to the decontamination process. A procedure of 100% catching of decontamination products by special sorbing films was developed providing for laser emission passing through sorbing films. In such a case oxides "thrown" from surface of processed detail are sorbed at adhesive layer applied to one side of the film. Thanks to sorbing film application "contaminated" and "decontaminated" zones can be separated while processing contaminated details. After decontamination, sorbing film and particles deposited on it can be recovered together (Fig. 3).

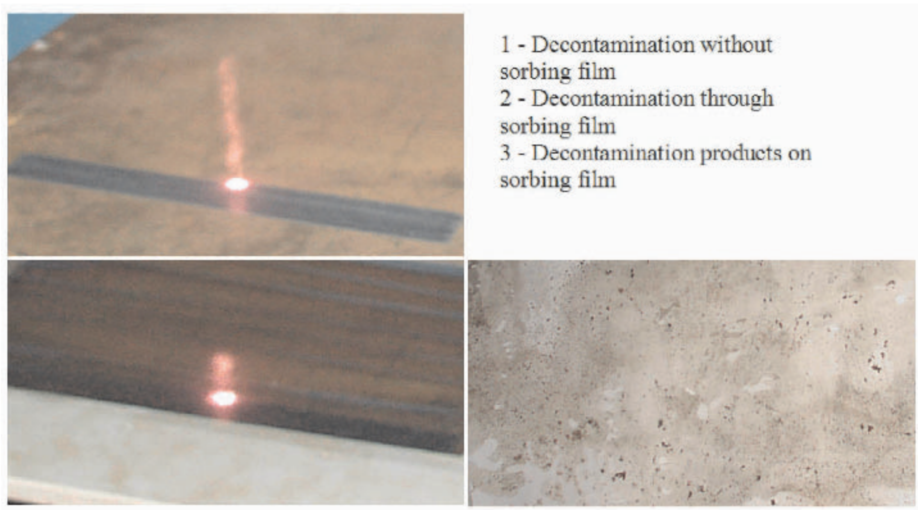

Figure 3. Decontamination using sorbing films

At the second work phase a "Laser Decontamination" laboratory was established at Nuclear Physics Institute (Gatchina, Leningrad region), and a "hot chamber" was made for works with contaminated details. The "hot chamber" is equipped with two laser installations to perform studies. Laser light enters the chamber via inlet window and has a possibility of two-coordinate scanning with the help of mirrors guided from processor. As established in experiments, when processing contaminated surfaces, the contamination level by individual radionuclides decreases by 70% and more (Fig. 4).

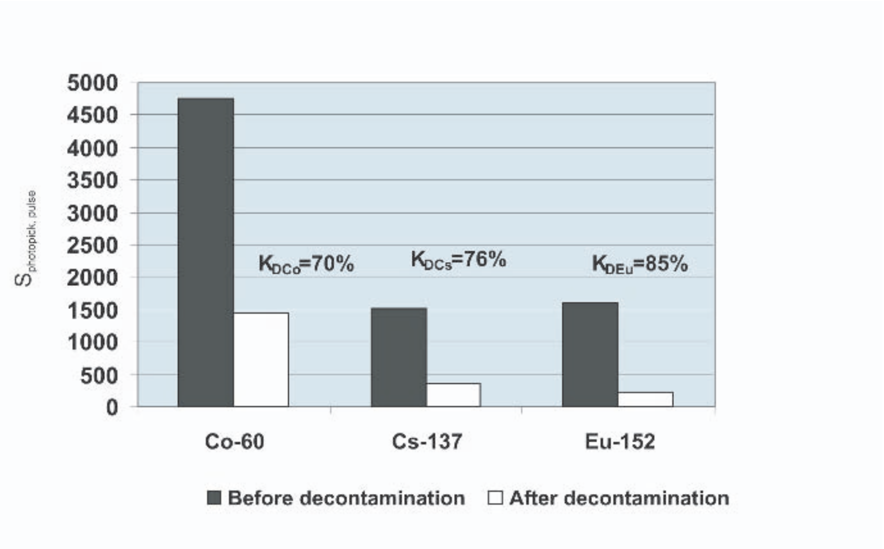

Figure 4. Efficiency of sample surface decontamination

At the 3rd work stage (2003) a mobile laser installation was developed and manufactured (Fig.5).

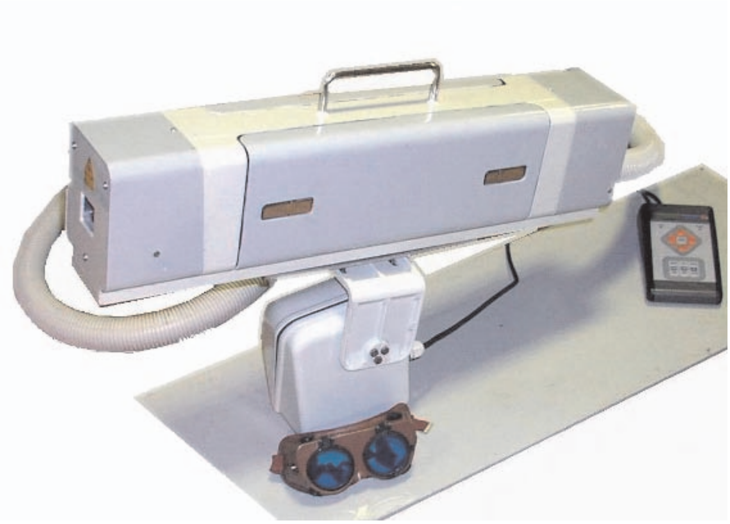

Figure 5. Mobile laser installation

The installation comprises: laser emitter, laser power-supply unit, water-air cooling system and guiding computer. Total weight of the installation is 40 kg; consumed power is 3 kW from power network 220 V. The installation capacity reaches to 2 m^2 per hour; the distance from emitter to surface to be decontaminated can attain 1.5 m. Laser emitter is installed at a remotely operated rotator.

The facility has been put into trial operation at "Laser Decontamination" laboratory.

To work the decontamination procedure through, real samples of equipment and systems of nuclear submarines under dismantlement were used along with samples of oil-gas pipes with depositions containing natural radionuclides (^{238}U, ^{232}Th and ^{226}Ra), Fig. 6.

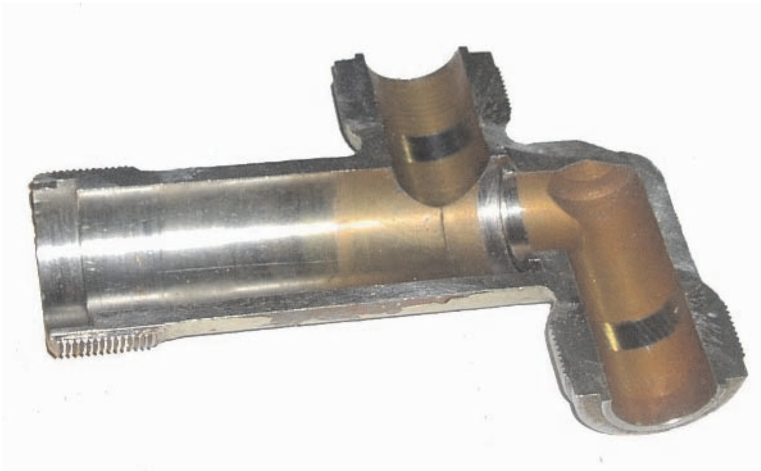

Figure 6. Use of real samples in laser decontamination studies

Conclusions

- A "dry decontamination" procedure has been developed.
- A prototype model of mobile laser installation has been manufactured.
- A procedure of preventing environment contamination by radioactive products during the decontamination process has been worked through.
- Decrease in the decontamination level by 70% and more after laser processing of contaminated details has been experimentally proved.
- A set of laser decontamination equipment has been manufactured.
- A process technology for decontamination of inside surfaces of pipes without their mechanical destruction has been worked through.

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ENVIRONMENTALLY APPROPRIATE AND ECONOMICALLY EFFICIENT COMPLEX DECOMMISSIONING OF NUCLEAR SUBMARINES AS AN IMPORTANT FACTOR FOR SUSTAINABLE DEVELOPMENT OF LOCAL-SCALE NUCLEAR INDUSTRY IN RUSSIA

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After about 15-year period of nuclear power industry stagnation throughout the world a trend for its step-by-step development has recently become apparent. According to the IAEA's data, as of December 2002, 441 power units of 358 661 MW integral electric capacity were in operation, and 32 new power units of 26 910 MW integral electric capacity were under construction (Table 1). The table data show that 70.3% of the world's nuclear power are produced by only five countries: the USA (30.3%), France (16.4%), Japan (12.2%), Germany (6.3%) and Russia (5.1%), the remainder amount being generated by 24 countries of four continents of the Globe. The structure of fuel-energy balance and power production by type of energy resources is illustrated in Table 2. In 1998 the proportion of nuclear power in the global fuel-energy balance and electric power generation made up 6% and 17%, respectively [1].

TABLE 1. Nuclear power generation throughout the world *(IAEA's data, as of 31.12.2002)*

Regions and countries	Power units in operation		Power units under construction		Power generation by nuclear power plants in 2002	
	Power unit number	Total electric power, MW	Power unit numbe r	Total electric power, MW	Total, TWh	Contribution to world's nuclear power generation,
North America						
Canada	14	10 018	-	-	71.0	2.8
USA	104	98 230			780.1	30.3
Latin America						
Argentina	2	935	1	692	5.4	0.2
Brazil	2	1 901	-	-	13.8	0.5
Mexico	2	1 360	-	-	9.4	0.4
West Europe						
Belgium	7	5 760	-	-	44.7	1.7

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 391–400.

UK	31	12 252	-	-	81.1	3.1
Germany	19	21 283	-	-	162.3	6.3
Spain	9	7 574	-	-	60.3	2.3
Netherlands	1	450	-	-	3.7	0.1
Finland	4	2 656	-	-	21.4	0.8
France	59	63 073	-	-	415.5	16.4
Switzerland	5	3 200	-	-	25.7	1.0
Sweden	11	9 432	-	-	65.6	2.6
East Europe and CIS countries						
Armenia	1	376	-	-	2.1	0.1
Bulgaria	4	2 722	-	-	20.2	0.8
Hungary	4	1 755	-	-	12.8	0.5
Lithuania	2	2 370	-	-	12.9	0.5
Russian Federation	30	20 793	3	2 825	130.0	5.1
Rumania	1	655	1	655	5.1	0.2
Slovak Republic	6	2 408	2	776	18.0	0.7
Slovenia	1	676	-	-	5.3	0.2
Ukraine	13	11 207	4	3 800	73.4	2.9
Check Republic	6	3 468	-	-	18.7	0.7

Africa						
South African Rep.	2	1 800	-		12.0	0,5
Middle East and South Asia						
India	14	2 503	7	3 420	17.8	0.7
Iran	-	-	2	2 111	-	-
Pakistan	2	425	-	-	1.8	0.1
Far East						
North Korea	-	-	1	1 040	-	-
China	7	5 318	4	3 275	23.4	0,9
Republic Korea	18	14 890	2	1 920	113.1	4,4
Taiwan	6	4 884	2	2 700	33.9	1.3
Japan	54	44 287	3	3 696	313.8	12.2
World total:	441	358 661	32	26 910	2 574.2	100%

TABLE 2. Structure of world's fuel-energy balance and power generation by energy carrier, %

Energy carrier	Fuel and energy balance	Electric power output
Oil	40	9
Coal	25	37
Gas	22	16
Hydropower*)	7	20
Nuclear energy	6	17
Total	100	100

*) including other renewable power sources

In 2003 the integral power generation in Russia made up 888.2-billion kilowatt-hour, the proportion of nuclear power being 16.7% that was above that of 2002 by 6.3%. The capacity factor of all operating Russian Nuclear Power Plants (NPP) equaled 76.3% that was also above that of 2002 (by 4.6%). In 2002 the capacity factor at Volgodonsk NPP reached an unprecedently high index of 83.3% [2].

In Finland a decision has been recently made on construction of a new NPP with a French-German power unit. The USA being the largest world's nuclear power producer (about 30% of the world's nuclear power generation) are preparing for the next step in nuclear power industry development.

Changes for the better in the world's nuclear power industry can be explained by the following factors:

1. Steady growth of the Earth's population, which, according to OCDE's forecasts, would increase by a factor of 1.5 over the coming 50 years;

2. Doubling of the power consumption level by 2050 and fivefold increase in power demand during the following 100 years. Today about 2 billion people have no chance of using electric power;

3. Intensification of the greenhouse effect due to releases of gases accompanying power generation technologies based on combustion of organic fuel. According to the World Energy Council forecasts, the integral global emissions of CO_2 would increase from 24 billion tones in 2000 to 36 billion tones in 2020 and would double by the end of the century (Table 3) [3].

TABLE 3. Global emissions of CO_2

Years	Emissions of CO_2, billion t			
	World	OECD	Countries of transition economies	Developing countries
1990	20.878	10.640	4.066	6.171
1997	22.561	11.467	2.566	8.528
2010	29.575	13.289	3.091	13.195
2020	36.102	14.298	3.814	17.990

From the above it follows that nuclear power is presently the only real and well-organized industry that is capable of generating electric power and heat under simultaneous decrease of greenhouse gas emissions due to nonuse of organic fuel and thus able to ensure energy and environmental safety of fuel-energy complex in Russia and its regions.

However no development of nuclear power industry is possible without appropriate solution of a variety of complex and expensive problems related to transportation, storage, processing and disposal of radioactive waste generated during both normal operation and decommissioning of nuclear facilities including Nuclear Submarines (NS).

The problem of NS complex decommissioning in Russia became of special concern after 1992–1994 when a number of Governmental Decrees determining the principal decommissioning stages, the technological policy and the key enterprises entrusted with work execution was issued by the Russian Federation (RF) Government.

In pursuance of the above decrees by a special decision of RF Minatom, Ministry of Defense and Ministry of Economic Development (May 20, 1998) a work group had been established, which developed a time-schedule of topical works to be executed in support of the program of complex decommissioning of NS and environmental remediation of radiation-hazardous naval facilities in Kola Peninsula and the Arkhangelsk region. The time-schedule comprised the following activities:

– Waterborne storage of non-defueled NS withdrawn from service;
– SNF unloading from NS reactors and further management;
– Selection and justification of sites to construct temporary facilities for storage of

one-compartment reactor units at shipyards performing NS dismantlement;

– Development and implementation of projects on reducing the environmental risk due to operation of coastal naval bases and environmental remediation of their sites;
– Radioactive Waste (RW) management;
– Development of engineering, managerial and administrative documentation.

Unfortunately, the due dates of priority activities determined by the time schedule have not been kept at all times slowing down the decommissioning process.

Complex decommissioning of NS in Russia is based on the following principles:

– reducing in all possible way both the time of waterborne storage of NS taken out of service and logistical expenditures for their safe afloat storage;
– priority defueling of NS reactors with damaged Spent Fuel Assemblies (SFA);
– minimizing the amount of non-reprocessable waste generated during NS decommissioning operations;
– reducing environmental risks caused by both operation of naval radiation-hazardous facilities and subsequent remediation of their sites in the North-west and the Far East Russia.

To perform complex decommissioning of NS, which principal stages are illustrated in Fig. 1, a variety of documents are necessary. E.g., at present only at the pre-defueling stage over 30 managerial, administrative and engineering documents are required [4]. This complicates the work of personnel in a case of off-normal situation and increases the risk of negative effects of the human factor. Thus to date a thorough analysis of the list and contents of the totality of developed NS decommissioning documents is necessary.

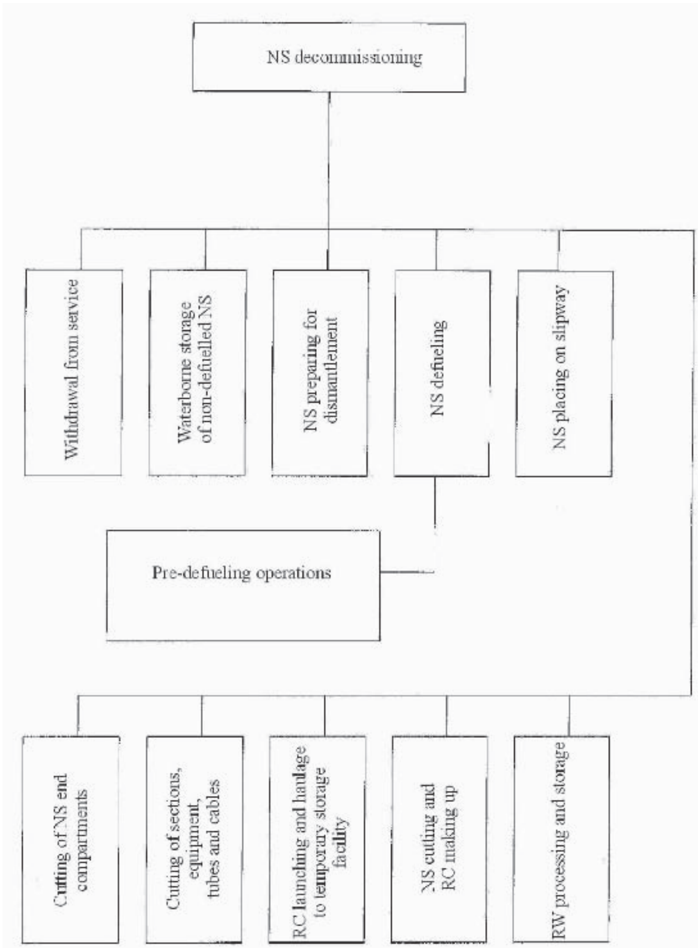

Figure 1. Principal NS complex decommissioning stages

Among the key NS decommissioning problems those of SNF storage and shipping should be considered first. SNF management-related challenges are equally important for naval Power Reactor Installations (PRI) and commercial NPP power units. To perform SNF unloading operation at the special on-shore defueling complex of "Zvezdochka" shipyard commissioned in 2002 the following equipment is used: "TK-18" and "TUK 108/1" transportation and storage casks and transport railcars "TK-VG-18" serving for SNF shipment to PA "Mayak". The above-mentioned time schedule of priority NS decommissioning activities provided for fabrication of a pilot lot (15 units) of two-purpose metal-concrete containers (MBK) and transportation & storage casks (TUK) in 1999; their serial production was to start in January 2000.

So far a variety of investigations of different SNF storage options have been performed in different countries with highly developed nuclear power industry. Dr. O.K. Earle, an expert of the US Argonne National Laboratory presented at the NATO-Russia ARW in Moscow (April 2002) the results of recent investigations in this area [5].

In Table 4 different SNF storage options and their relative costs are compared. From the table data it follows that in terms of relative costs the "drywell storage" is the most advantageous option [5].

TABLE 4. Relative cost of SNF storage options [5]

Storage System	Relative Costs
Cask – dual purpose	0,1
Storage Cask	0.61 - 0.86
Vault	0.62 - 0.75
Caisson (silo)	0.5 – 0.6
Drywell	0.29 - 0.6

When defueling the taken-out-of-service NS, storing and transporting their SNF, the problems of ensuring nuclear and radiation safety of personnel, population and environment are especially pressing. The highest risk is due to a hypothetical possibility of initiating spontaneous chain reaction. An option of SNF storage directly within PRI reactor core after the primary circuit unwatering, filling with heavy metal/polymeric materials followed by their curing developed by Russian scientists and engineers would eliminate nuclear hazard of non-defueled PRI reactors including those housing damaged SFAs. A comparative analysis between the mode of SNF storage in two-purpose metal-concrete transport casks (MBK TUK) and the "in situ" SNF storage revealed the advantages of the latter option for all safety indices [6].

The whole problem of SNF unloading, storage and transportation – especially in the case of damaged SFA - represents an extremely complex, expensive and hazardous challenge. Note that NS reactors contain a relatively minor SNF amount, and thus the expediency of its reprocessing raises doubts. Based on this consideration and taking into account both buoyancy problems of many taken-out-of-service NS and a high probability of primary circuit depressurization due to corrosion of pipelines, a complex of fundamental investigation and application studies under a special federal-level program seems expedient. Such program should be focused on justification of feasibility, economic efficiency and environmental safety of underground disposal of non-defueled NS reactors filled with liquid-metal melt of lead. Disposal of one-compartment non-defueled reactor units filled with lead-containing materials in dry underground mines (similar to those proposed for underground NPPs, Fig. 2) or in deep stable geological structures represents a logic development of the above option.

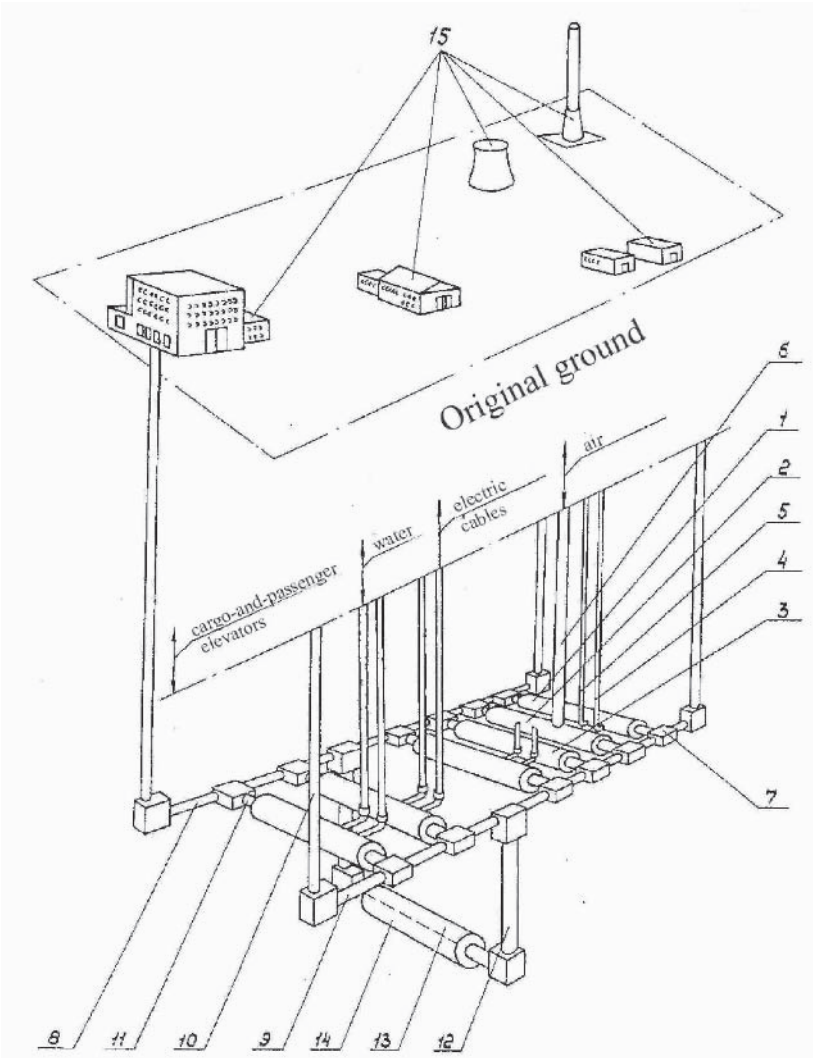

Figure 2. Underground nuclear power plant

1. nuclear power unit; 2. input ventilation system; 3. RW collection and processing system; 4. cable routing shaft; 5. water cooling lead shaft; 6. ventilation shaft; 7. tunnel for transport and communication lines: non-contaminated (8) and contaminated (9); 10. shaft for cargo-and-passenger elevators; 11. end buffer areas of process units; 12. shaft between upper-level and bottom-level process units; 13. SNF storage facility; 14. RW repository; 15. support buildings on original ground.

It is not improbable that, if using a complex approach to analyze the today's modes of SNF unloading, storage, transportation, reprocessing and disposal, the proposed reactor compartment disposal option would become the most safe and inexpensive. Anyway, to dispose non-defueled reactor compartments of all-type decommissioned NS in dry mines, a parcel of land of about 415 м^2 (i.e. comparable to that of the first stage of Leningrad NPP) would be necessary.

During NS decommissioning activities large quantities of solid and liquid radioactive waste and of toxic waste are produced, a part of which is released to the environment. Even "normal" (accident-free) NS decommissioning leads to pollution and contamination of atmospheric air, soils and waters. The flame cutoff procedure accompanied by release of dust and metal (nickel, manganese, iron and chromium) oxides to the atmosphere is the main pollution source. Another product of NS complex decommissioning - solid industrial waste – is a source of soil pollution and contamination. Liquid radioactive waste and oil-containing wastewater contaminate and pollute aquatic systems. To improve the environmental safety when decommissioning NS, processing of toxic waste at enterprises and replacement of the thermal cutting technology by more efficient and environmentally appropriate procedures is necessary.

It is noteworthy that so far most of projects of foreign companies and important international financial flows have been mainly concentrated on the development of infrastructure to support decommissioning of strategic-missile NS withdrawn from military service ahead of schedule. The most of funding into re-equipment of "Zvezdochka" shipyard and Far East Plant "Zvezda" was granted by the U.S. Government and the U.S. Department of Defense to establish an on-shore defueling complex for retired NSs, perform reconstruction of slip and the areas of metal structure and cable cutting. A variety of French, Norwegian and U.S. companies together with Russian enterprises also addressed the problems of processing SRW and LRW produced during NS complex decommissioning.

The problems under consideration have been the objective of many publications. Unfortunately, virtually none of them contains information on cost parameters and technico-economic indices of the decommissioning process complicating the choice between domestic producers and foreign supplies of the decommissioning equipment and techniques. In this context, large-scale investigations focused on analysis, ranking and generalization of factual costs of different stages and the whole NS decommissioning process seem appropriate. Objective appraisal of the scope and efficiency of the international financial and technical assistance to Russia and its effects on further economic development of the country is especially important.

From the above analysis the following conclusions and recommendations can be drawn up:

1. Conduct an analysis of the integrity of documentation on NS complex decommissioning in order to eliminate parallelism and optimize the amount of documents.

2. Perform a complex of fundamental and applied investigations to justify feasibility, economic expediency and environmental safety of underground disposal of non-defueled one-compartment reactor units filled with molten lead-containing material in dry mines or stable geological structures at an environmentally safe depth.
3. Based on real industrial costs, perform economic analysis of the whole range of NS complex decommissioning activities.
4. Concentrate the efforts of scientists and engineers on improvement of the available NS decommissioning technologies and development of new ones focused on both minimization of duration and cost of the whole process and enhancement of nuclear and radiation safety for personnel, population and environment.

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UNLOADING AND STORAGE OF SPENT NUCLEAR FUEL DURING DISMANTLING OPERATIONS OF FRENCH NUCLEAR SUBMARINES

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The first generation of French SSBNs "LE REDOUTABLE class" has been dismantled.

The fuel which was used in these SSBNs is based on a UZr metallic alloy with highly enriched uranium for which no reprocessing process has been developed.

The management of this spent fuel consists in the following operations:

- unloading of the spent fuel elements
- transfer under a lead cask
- interim storage in a desactivation pool on the maintenance shipyard
- transfer in type B containers according to IAEA regulations from the shipyard to the TECHNICATOME facility in the nuclear center of CADARACHE
- in pool interim storage in the TECHNICATOME facility
- conditioning for dry interim storage in the TECHNICATOME hot cell
- in pool interim storage of conditioned elements
- transfer of the conditioned elements to the dry storage facility CASCAD on the site of CADARACHE in type B containers.

Unloading and transfer of the spent fuel elements

A heightening of the vessel is installed to enable the unloading operations under water. The fuel elements are extracted with an unloading machine and loaded through the top of the lead cask in which they are transferred to the desactivation pool.

The main principles which are applied are the following:

- maintenance in subcritical configuration
- residual power evacuation
- protection against irradiation and contamination
- containment
- protection of the vessel against shocks and migrant objects
- use of secured handling tools

A. Sarkisov and A. Tournyol du Clos (eds.), Scientific and Technical Issues in the Management of Spent Fuel of Decommissioned Nuclear Submarines, 401–402.

– assurance of return to safe conditions in case of breakdown.

Interim storage in a desactivation pool on the maintenance shipyard

Main function:

– storage under water of irradiated fuel elements

Other functions:

– reception of the containers
– dimensional control of the fuel elements
– tightness control
– operations and conditioning of fuel elements

The transfer zone of the pool is equipped with a load accompanying device of the transfer lead cask.

In normal conditions, the piston goes down at the rolling speed of the cable

In case of accident on the lifting line, the piston alone supports the load and masters the descent speed.

The fuel elements taken out of the transfer lead cask are stored in racks for desactivation and then transferred in type IU15 containers for transport to CADARACHE.

In pool interim storage in the Technicatome facility and conditioning for dry interim storage in the Technicatome hot cell

At their arrival in CADARACHE the IU15 containers are put down in the canal, fuel elements are placed in racks waiting for conditioning, put in canisters which will be used for storage in the dry storage facility CASCAD and transferred to the conditioning hot cell for final preparation.

In the cell, the canisters are drained after having been tiped up, the inside of the canisters is dried under vacuum, the canisters are closed by a cap, the absence of external contamination is controlled. Afterwards, the canisters return to the pool, and after last controls are ready for transport to the dry storage facility CASCAD.