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Predicted radionuclide release from marine reactors dumped in the Kara Sea

Report of the Source Term Working Group of the International Arctic Seas Assessment Project (IASAP)

INTERNATIONAL ATOMIC ENERGY AGENCY

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FOREWORD

In 1992 the news that the former Soviet Union had, for over three decades, dumped high level radioactive wastes in the shallow waters of the Kara Sea caused widespread concern, especially in countries with Arctic coastlines. The IAEA responded by launching an international study, the International Arctic Seas Assessment Project (IASAP), in order to assess the potential health and environmental implications of the dumping and to examine the feasibility of remedial actions related to the dumped wastes.

In May 1993 the Russian Federation provided information to the IAEA about the high and low level radioactive waste dumped in the Arctic Seas ("White Book-93"). According to the "White Book-93" the total amount of radioactivity dumped in the Arctic Seas was more than 90 PBq. The items dumped included six nuclear submarine reactors containing spent fuel, spent fuel from an icebreaker reactor, ten reactors without fuel, and liquid and solid low level waste. The nuclear reactors and the fuel from the icebreaker reactor were dumped in the shallow bays of Novaya Zemlya and in the Kara Sea.

Within the framework of the IASAP project the Source Term Working Group was established, with the objective of determining the information needed about the waste for use in impact assessment calculations in evaluating the usefulness of possible remedial actions. This involves having the knowledge of the inventory of radionuclides in the dumped objects, and protective barriers provided to them either by initial construction or prior to dumping, and of the likely behavior of the barriers with time in the marine environment. The efforts of the group have been focused on the dumped reactors and especially the dumped reactor fuel which contains the highest inventory of radionuclides.

The present report summarizes the work carried out by the Source Term Working Group of IASAP during 1994-1996. The report is based on the studies concerning the initial and current radionuclide inventories, operational history and construction of the reactors carried out by Y. Sivintsev of the Russian Research Center "Kurchatov Institute", Moscow and E. Yefimov of the Institute of Physics and Power Engineering, Obninsk, Russian Federation. The working group convened five times and evaluated the results of the studies and developed models for prediction of potential releases to the environment. The calculations were carried out at the Royal Naval College, Greenwich, UK, by N. Lynn, J. Warden and S.Timms and at the Lawrence Livermore National Laboratory, California, USA, by M. Mount.

The IAEA wishes to express its gratitude to all those who participated in the work of the Source Term Working Group and preparation of this report. The IAEA officer responsible for this work was K.-L. Sjoeblom of the Waste Safety Section, Division of Radiation and Waste Safety.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

In the spring of 1993, the Russian report, *Facts and Problems Related to Radioactive Waste Disposal in Seas Adjacent to the Territory of the Russian Federation* [1], was released. The findings presented in that report were the result of a scientific study commissioned in October 1992 by the Office of the President of the Russian Federation. Related to the Arctic area, the White Book, as the report was later called, reported that:

(1) between 1965 and 1988, 16 marine reactors from seven former Soviet Union submarines and the icebreaker *Lenin*, each of which suffered some form of reactor accident, were dumped at five sites in the Kara Sea;
(2) between 1960 and 1991, low level liquid radioactive waste was discharged at sites in the White, Barents, and Kara Seas; and
(3) between 1964 and 1991, low and intermediate level solid radioactive waste was dumped at sites in the Barents and Kara Seas.

Of the discarded marine reactors, six of the 16 contained their spent nuclear fuel (SNF). In addition, approximately 60% of the SNF from one of the three icebreaker reactors was disposed of in a reinforced concrete and stainless steel (SS) shell container. The vast majority of the low and intermediate level solid radioactive waste was disposed of in containers of unknown composition. The Kara Sea disposal sites for the 16 marine reactors and low and intermediate level solid radioactive waste varied in depth from 12 to 380 m. In particular, the icebreaker reactors and part of their SNF were reportedly disposed of in Tsivolka Fjord at an estimated depth of 50 m.

The White Book also reported estimates of total radioactivity at the time of disposal, these were:

(1) 8.5×10^{16} Bq of fission products in the SNF;
(2) 3.7×10^{15} Bq of activation products in the reactor components;
(3) 8.9×10^{14} Bq of unspecified origin in the low level liquid radioactive waste, over 50% of which was discharged in the Barents Sea; and
(4) 5.9×10^{14} Bq of unspecified origin in the low and intermediate level solid radioactive waste, over 95% of which was discarded in the Kara Sea [1].

With a few exceptions, information was provided on the radionuclides present in the wastes and no estimate was made of the current levels of radioactivity in the dumped wastes.

1.2. INTERNATIONAL ARCTIC SEAS ASSESSMENT PROJECT

International concern over the possible health and environmental effects both short and long-term, local, regional, and global, from disposal of these aforementioned radioactive wastes in the shallow waters of the Arctic Seas prompted the International Atomic Energy Agency (IAEA), as part of its responsibilities to the London Convention of 1972, to initiate the International Arctic Seas Assessment Project (IASAP) [2]. The IASAP formally began on February 1, 1993 in Oslo, Norway at the meeting on the Assessment of Actual and Potential Consequences of Dumping Radioactive Waste into Arctic Seas, organized by the IAEA in cooperation with the governments of Norway and the Russian Federation.

The stated objectives of the IASAP are to:

(1) assess the risks to human health and to the environment associated with the radioactive waste dumped in the Kara and Barents Seas, and

(2) examine possible remedial actions related to the dumped wastes and to advise on whether they are necessary and justified [2].

To meet these objectives, the work was organized into five working areas:

(1) source term,
(2) existing environmental concentrations,
(3) transfer mechanisms and models,
(4) impact assessment, and
(5) remedial measures.

1.3. OBJECTIVES OF THE SOURCE TERM WORKING GROUP

The Source Term Working Group was established to prepare a detailed inventory of and release rates for the radionuclides dumped at each Kara Sea disposal site. To this end, inventories were calculated for the spent nuclear fuel (SNF) and activated components at time of disposal and projected forward in time; protective barriers, if any, were evaluated for their potential effect on radionuclide release; and a number of model scenarios were developed to predict the potential release of the radionuclide inventory into the Kara Sea.

This document summarizes the efforts of the Source Term Working Group to complete the tasked objectives under the IASAP. It presents a detailed discussion of the fission product, actinide, and activation product inventories at each Kara Sea disposal site and a detailed description, with assumptions, of the models used to predict potential release of the radionuclides into the Kara Sea. Results of the release scenario models, reliability of the model input parameters, and an analysis of the sensitivity of the results to changes in the protective barrier lifetimes and SNF corrosion rates are then presented. The potential for recriticality of the SNF bearing cores and considerations for potential remedial measures are next addressed. Finally, conclusions are drawn with respect to the radionuclide releases at each Kara Sea disposal site from the SNF and activated components.

It should be noted that this document discusses only the estimates of the inventory and release of radionuclides associated with the marine reactors dumped in the Kara Sea. The low level liquid radioactive waste and low and intermediate level solid wastes discarded in the Barents and Kara Seas were not included in this study.

1.4. DISPOSAL SUMMARY

Table I presents a summary of pertinent disposal information for the marine reactors dumped in the Kara Sea [1]. Figure 1 shows a map of the Northeast coast of Russia with Novaya Zemlya and the approximate locations of the five disposal sites.

2. SOURCE TERM DEVELOPMENT

2.1. BACKGROUND

The following two sections describe the accidents that occurred aboard the nuclear submarines and icebreaker that led to their ultimate disposal in the Kara Sea.

2.1.1. Submarine pressurized water reactors [3, 4, 5, 6]

Six of the seven nuclear submarines contained two pressurized water reactors (PWRs) each. Eleven of these PWRs were dumped into the Kara Sea between 1965 and 1988: eight within and three without their reactor compartments (RCs). All these nuclear submarines suffered some form of reactor accident; however, many specifics of the reactor design, maximum thermal power, compartment layout, detailed operating histories, and accident scenarios remain classified.

TABLE I. PERTINENT DISPOSAL INFORMATION FOR THE MARINE REACTORS DUMPED IN THE KARA SEA [1]

Disposal site	**Year of disposal**	**Factory number**	**Dumped unit**	**Disposal coordinates**[1]	**Disposal depth**[2] **(m)**	**Number of reactors**		**Total activity (PBq)**	
						Without spent nuclear fuel	**With spent nuclear fuel**	**At the time of disposal**	**1994**
Abrosimov Fjord	1965	901	Reactor compartment	71° 56.03' N 55° 18.15' E	20 (10-15)	-	2	3.0	0.73
		285	Reactor compartment	71° 56.03' N 55° 18.08' E	20 (10-15)	1	1	12	0.66
		254	Reactor compartment	71° 55.22' N 55° 32.54' E	20	2	-	0.093	0.009
	1966	260	Reactor compartment	71° 56.03' N 55° 18.08' E	20	2	-	0.044	0.005
Tsivolka Fjord	1967	OK-150	Reactor compartment and special container with fuel	74° 26.10' N 58° 36.15' E	50	3	0.6[3]	20	2.2
Novaya Zemlya Depression	1972	421	Reactor	72° 40' N 58° 10' E	300	-	1	1.0	0.29
Stepovoy Fjord	1981	601	Submarine	72° 31.25' N 55° 30.25' E	50 (30)	-	2	1.7	0.84
Techeniye Fjord	1988	538	Reactors	73° 59' N 66° 18' E	35-40	2	-	0.006	0.005
Total						10	6.6	37	4.7

[1] Disposal site coordinates for all units except those from factory number OK-150 are from reference [1]. Disposal site coordinates for the OK-150 units are from reference [4].

[2] The disposal depths are from reference [1]; those in parenthesis were obtained during joint Norwegian-Russian scientific cruises in 1993 and 1994.

[3] Thermal shields, hardware, and approximately 60% of SNF discarded in special container.

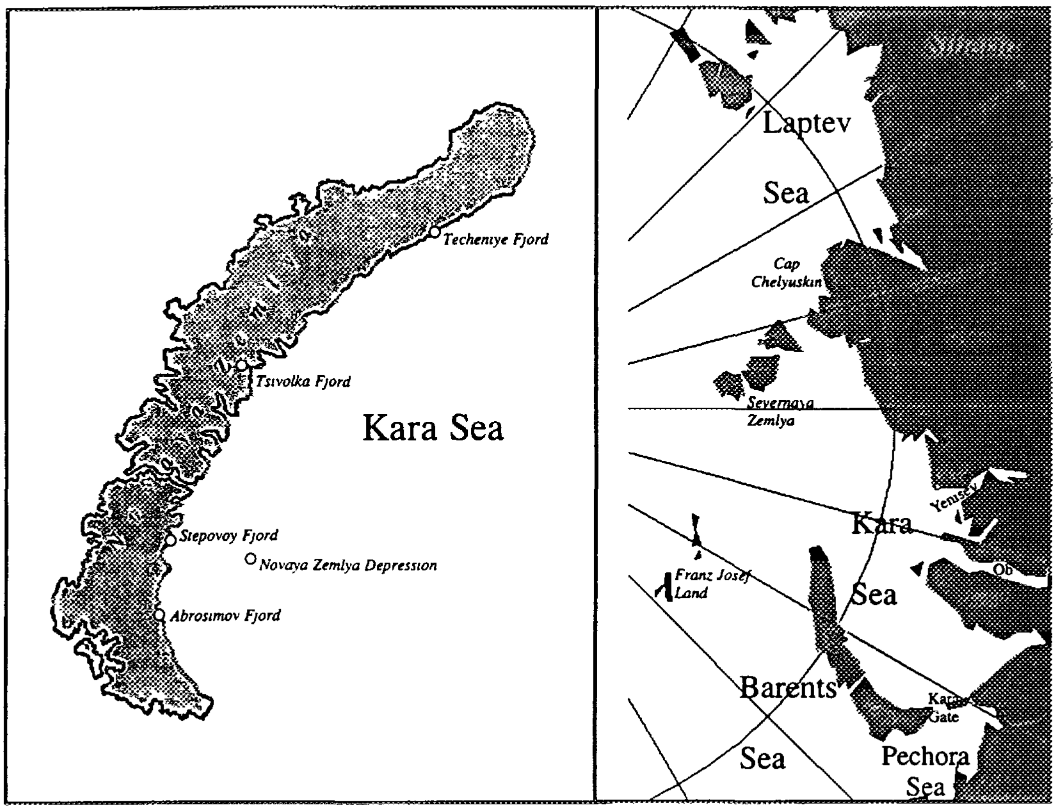


FIG. 1. Approximate locations of the marine reactor disposal sites in the Kara Sea on the northeast coast of Russia.

A criticality accident aboard the submarine identified as factory number 421 is known to have caused over pressurization of the right board reactor pressure vessel (RPV). The fuel rods were reported to not be damaged; however, a decision was made to not re-use the RPV. As such, the SNF was not removed.

2.1.2. Submarine liquid metal reactors [7, 9]

The remaining nuclear submarine, designated as factory number 601, contained two liquid metal reactors (LMRs) of 70 MW maximum thermal power each and used Pb-Bi as the coolant or heat transfer medium. The steam generating installation (SGI) began operation in December 1962 and operated successfully for the duration of the first core load. Both reactors were reloaded in September 1967 and operated at 10% of full power until May 24, 1968 when a portion of the left board reactor core channels became blocked while the submarine was at sea. As a consequence, approximately 20% of the left board reactor fuel was destroyed and deposited in the associated steam generator (SG) and volume compensator via the sealed primary circuit. The submarine subsequently returned to base on power from the right board reactor, shut down, and was sealed on or about June 6, 1968. The right board reactor was restarted in April 1972 and run for one day at approximately 20% of full power. Ultimately, it was discarded along with the left board reactor.

2.1.3. Icebreaker pressurized water reactors [6, 8]

Launched in Leningrad in 1959, the icebreaker *Lenin* was the first nuclear merchant ship in the world. During 31 years in commission, the icebreaker had two separate SGIs. The first SGI contained three PWRs of 90 MW maximum thermal power each and operated from 1959 to February 1965, when during routine repair of the SGI, an operator error allowed the core of the center line (N2) PWR to be left without water for some period of time. As a consequence, a part of the reactor core was damaged due to residual heat. It is this first SGI that forms the basis for the icebreaker source term.

The following sections detail what is currently known or estimated about the seven nuclear submarines and icebreaker with respect to the characteristics of their SGIs, their reactor operating histories, their radionuclide inventories, and their disposal operations.

2.2. CHARACTERISTICS OF THE STEAM GENERATING INSTALLATIONS

Known or estimated characteristics of the SGIs of the seven nuclear submarines and icebreaker are detailed in the following sections.

2.2.1. Submarine pressurized water reactors [3, 4, 6]

Specific information about the SGIs associated with the 11 discarded PWRs remains classified. Generally, the entire SGI, including the SGs and circulation pumps, was located aft of the submarine sail in an isolated RC. The two PWRs were aligned vertically, either in a plane perpendicular to the keel or along the keel, and were surrounded by a water-filled steel shield tank. Biological shields were located above the shield tank and around each PWR; however, the specifics of their construction materials are unavailable.

Each nuclear submarine PWR consisted of a cylindrical steel RPV, a reactor core and its associated support structure, and a series of radial and bottom thermal shields, the latter being employed to reduce heat and radiation effects on the RPV and subsequently extend its operating life. For submarine factory numbers 901, 285, 254, 260, and 538 the RPVs were made from type 15X2MΦA carbon steel, with approximate dimensions:

(1) 1.4 m diameter,
(2) 3.7 m height,
(3) 120 mm thick walls with a 6 mm thick internal cladding of type 1X18H9T SS,
(4) 310 mm thick bottom with a 5 mm thick internal cladding of type 1X18H9T SS, and
(5) 400 mm thick lid.

Figure 2 shows the cross-section of a RPV of the type used in submarine factory numbers 901, 285, 254, 260, and 538.

For submarine factory number 421, the RPVs were made from a variety of carbon steels, with approximate dimensions:

(1) 2 m diameter,
(2) 3.4 m height,
(3) 120 mm thick walls of type 15X2MΦA-A carbon steel with a 7.5 mm thick internal cladding of SS,
(4) 110 mm thick bottom of type 12X2MΦA-A carbon steel with a 7.5 mm thick cladding of SS, and
(5) 390 mm thick lid of 25X2MΦA carbon steel.

Figure 3 shows the cross-section of a RPV of the type used in submarine factory number 421.

The height and diameter of each PWR core remains classified. For the purposes of these calculations, each core was assumed to be loaded with 50 kg of ^{235}U. The ^{235}U enrichment was assumed to be 7.5% for the cores of submarine factory number 285 and 20% for all others [3]. Fuel rods were assumed to be constructed of a U-Al alloy.

For submarine factory numbers 901, 285, 254, 260, and 538, the radial thermal shields consisted of four concentric cylinders positioned around each reactor core and the bottom thermal shields consisted of four cylindrical plates positioned below the lower core plate. The four concentric

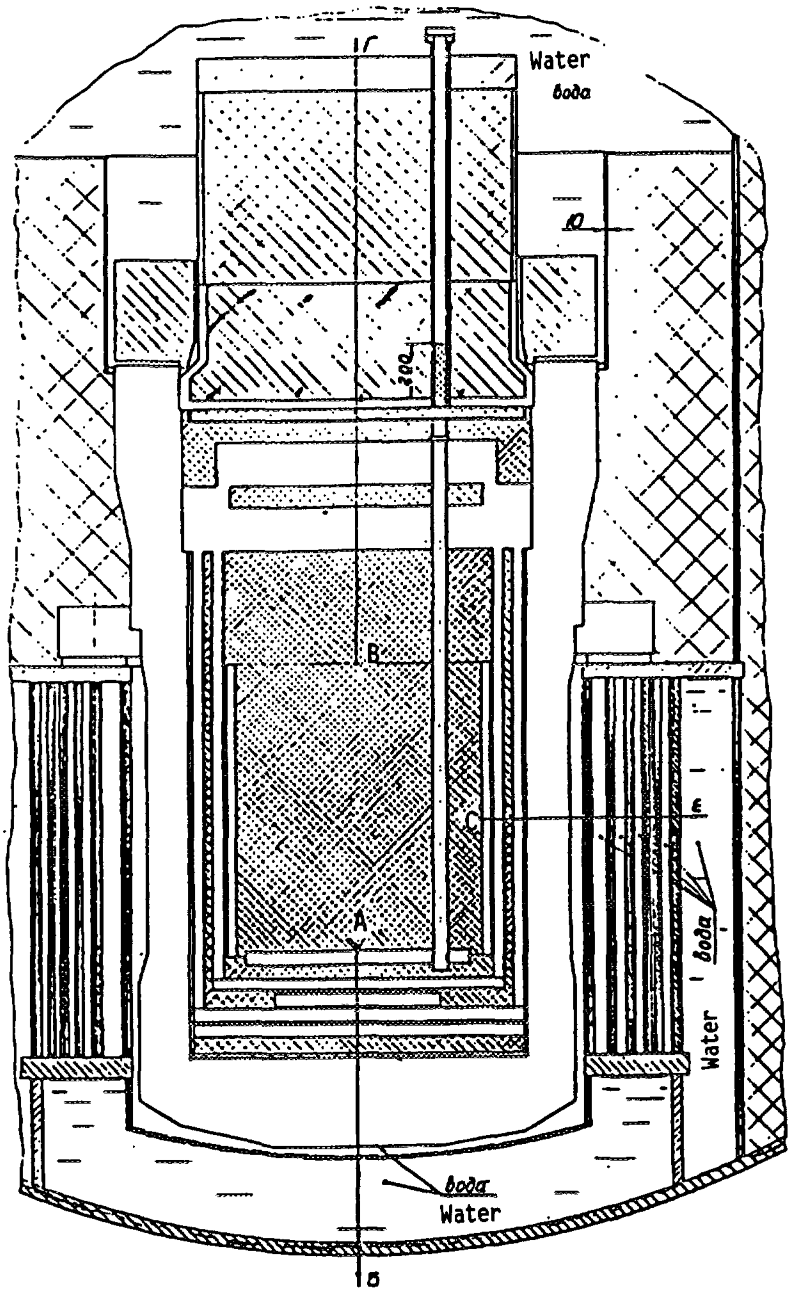
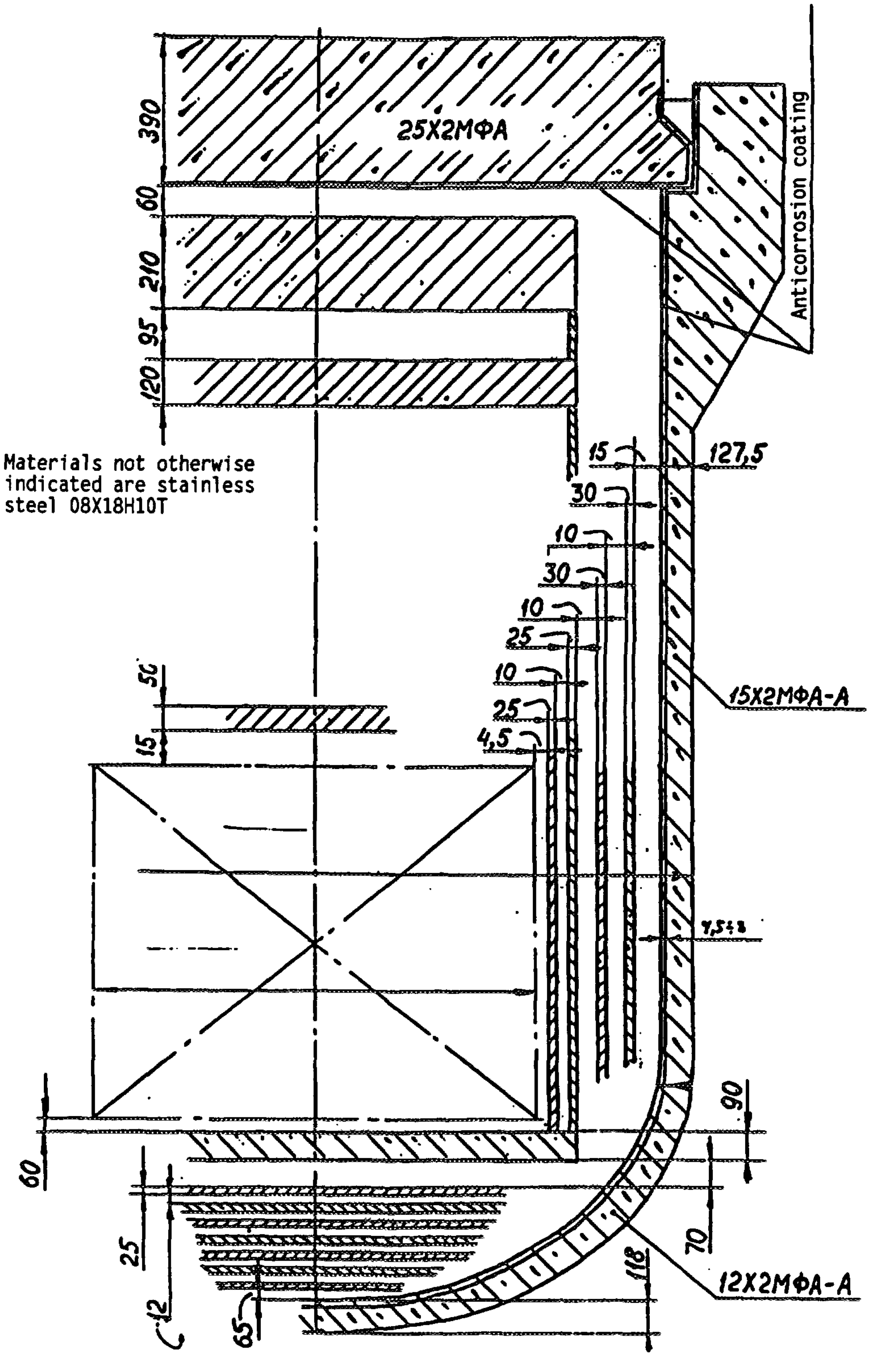


FIG. 2. Cross-section of a reactor pressure vessel of the type used in submarine factory numbers 901, 285, 254, 260, and 538 [3].



Materials not otherwise indicated are stainless steel 08X18H10T

FIG. 3. Cross-section of a reactor pressure vessel of the type used in submarine factory number 421 [3].

cylinders were made of type 1X18H9T SS of, from the core surface outward, 25-, 40-, 40-, and 20-mm thickness, respectively. The four concentric cylinders and the inner surface of the RPV wall were separated by primary-circuit water of, from the core surface outward, 20-, 6-, 20-, 5-, and 5-mm thickness, respectively. The lower core plate and four cylindrical plates were made of type 1X18H9T SS of, from the core surface downward, 28-, 45-, 40-, 50-, and 40-mm thickness, respectively. The lower core plate, four cylindrical plates, and the inner surface of the RPV bottom were separated by primary-circuit water of, from the bottom surface of the core downward, 30- 47-, 5-, 5-, 5-, and 60-mm thickness, respectively. Above each core are the upper ring of the reactor core barrel and four additional cylindrical plates, the lower two comprising the balance of the upper core barrel. The four cylindrical plates were made of type 1X18H9T SS of, from the core upward, 150-, 150-, 12-, and 25-mm thickness, respectively. The four cylindrical plates and the inner surface of the lid were separated by primary-circuit water of, from the core surface upward of, 400-, 70-, 74-, and 22-mm thickness, respectively. Above each RPV lid is an additional plug consisting of, from the RPV lid upward, 650 mm thick B_4C and 120 mm thick type CT-3 steel.

For submarine factory number 421, the radial thermal shields consisted of four concentric cylinders positioned around each reactor core and the bottom thermal shields consisted of seven cylindrical plates positioned below the lower core plate. The four concentric cylinders were made of an unspecified SS of, from the core surface outward, 25-, 25-, 30-, and 30-mm thickness, respectively. The four concentric cylinders and the inner surface of the RPV wall were separated by primary-circuit water of, from the core surface outward, 4.5-, 10-, 10-, 10-, and 15-mm thickness, respectively. The lower core plate and seven cylindrical plates were made of type 8X18H10T SS of, from the core surface downward, 90-, 25-, 25-, 25-, 25-, 25-, 25-, and 25-mm thickness, respectively. The lower core plate, seven cylindrical plates, and the inner surface of the RPV bottom were separated by primary-circuit water of, from the bottom surface of the core downward, 60-, 70-, 12-, 12-, 12-, 12-, 12-, 12-, and 65-mm thickness, respectively. Above each core are the upper core plate and two additional cylindrical plates, the latter two comprising the balance of the upper core barrel. The upper core plate and two cylindrical plates were made of an unspecified SS of, from the core surface upward, 50-, 120-, and 210-mm thickness, respectively. The upper core plate, two cylindrical plates, and the inner surface of the lid were separated by primary-circuit water of, from the core surface upward, 15-, unknown-, 95-, and 60-mm thickness, respectively.

2.2.2. Submarine liquid metal reactors [7, 9]

The entire SGI, including the SGs, circulation pumps, and primary circuit volume compensators, was located aft of the submarine sail in an isolated RC. The two LMRs were aligned vertically in a plane perpendicular to the keel and were surrounded by a lead-water tank shield. The SGs, circulation pumps, and primary circuit volume compensators were enclosed in lead lined structures.

Each LMR also consisted of a cylindrical RPV, a reactor core and its associated support structure, and a series of radial and bottom thermal shields. The RPV was made from SS, with approximate dimensions:

(1) 1.8 m diameter,
(2) 3.7 m height, and
(3) 30 mm thick walls.

External to the outer surface of the RPV were two cylindrical channel regions of 30 mm each that were formed through the addition of two concentric cylindrical SS shells of 10 mm and 20 mm, respectively. Figure 4 shows the cross-section of a RPV of the type used in submarine factory number 601.

The height and diameter of each LMR core was approximately 800 mm and 780 mm, respectively. A radial reflector was present and consisted of, from the core surface outward, 10 mm

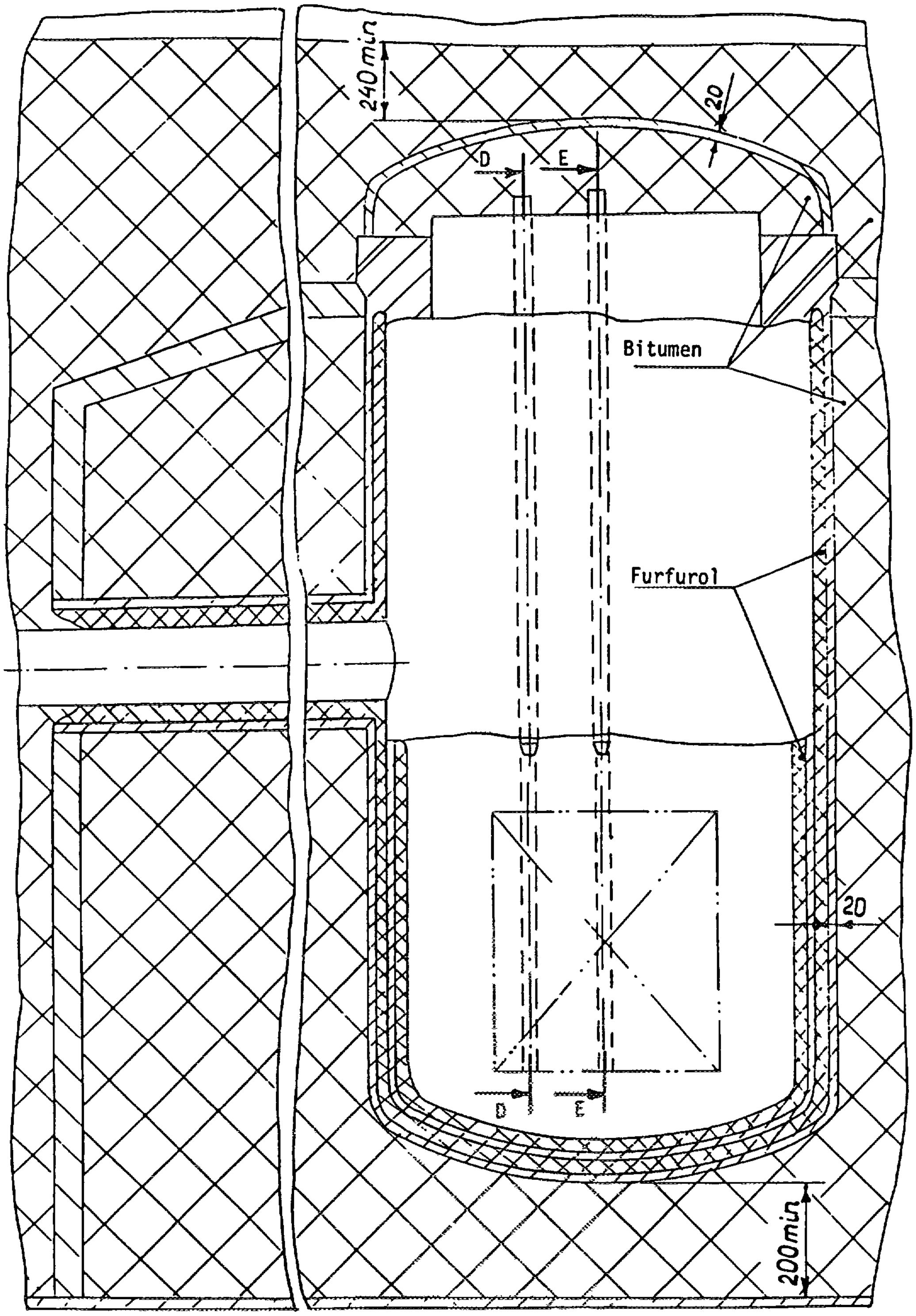


FIG. 4. Cross-section of a reactor pressure vessel of the type used in submarine factory number 601 [7].

thick SS, 65 mm thick BeO, and 8 mm thick SS. Above the core was a special shield plug through which the three emergency protection rods (EPRs), ten control or compensation rods (CCRs), and 24 emergency cooling tubes (ECTs) passed. The EPRs and CCRs extended into the core and were enclosed in special steel channels of approximately 20 mm inside diameter while the ECTs extended through portions of and about the radial thermal shield. The ten CCRs were approximately 17 mm inside diameter each and arranged in the core as follows:

(1) one at the center of the core,
(2) three evenly spaced at a radius of 97.5 mm on a 120° arc, and
(3) six evenly spaced within the annulus between radii of 97.5 and 292.5 mm.

The three EPRs were approximately 20 mm outside diameter each and evenly spaced at a radius of 195 mm on a 120° arc. Both the EPRs and CCRs were made of europium hexaboride (EuB_6). Figure 5 shows a schematic layout of the CCRs and the EPRs within the core of submarine factory number 601. The 24 ECTs were approximately 70 mm inside diameter and 80 mm outside diameter and evenly spaced on a radius of 641.5 mm. There was approximately 300 mm of the Pb-Bi coolant between the top of the reactor core and the bottom of the reactor shield plug. Each reactor core was loaded with 90 kg of ^{235}U at an enrichment of 90%. Fuel rod pellets were constructed of U-Be alloy in a BeO ceramic matrix and were of approximately 10 mm diameter. The pellets were covered with a 0.1 mm thick layer of Mg and clad in SS of 0.5 mm thickness. The resultant fuel rods were 11 mm outside diameter and arranged on a 13 mm triangular pitch. Considering the fuel rod pitch and core diameter, and accounting for the presence of the three EPRs and ten CCRs, there were approximately 3000 fuel rods in each LMR core. Typically, the distribution of LMR core materials was approximately 54% fuel, 36% Pb-Bi, and 10% SS. Figure 6 shows cross-sections for CCR and EPR channels of the type used in submarine factory number 601. Figure 7 shows a schematic layout of the ECTs and core in submarine factory number 601.

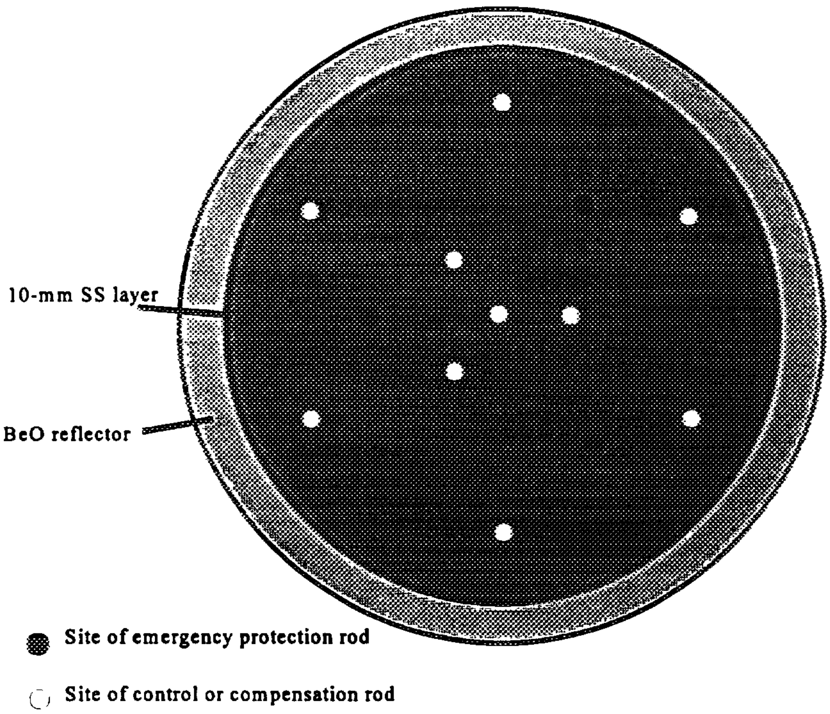
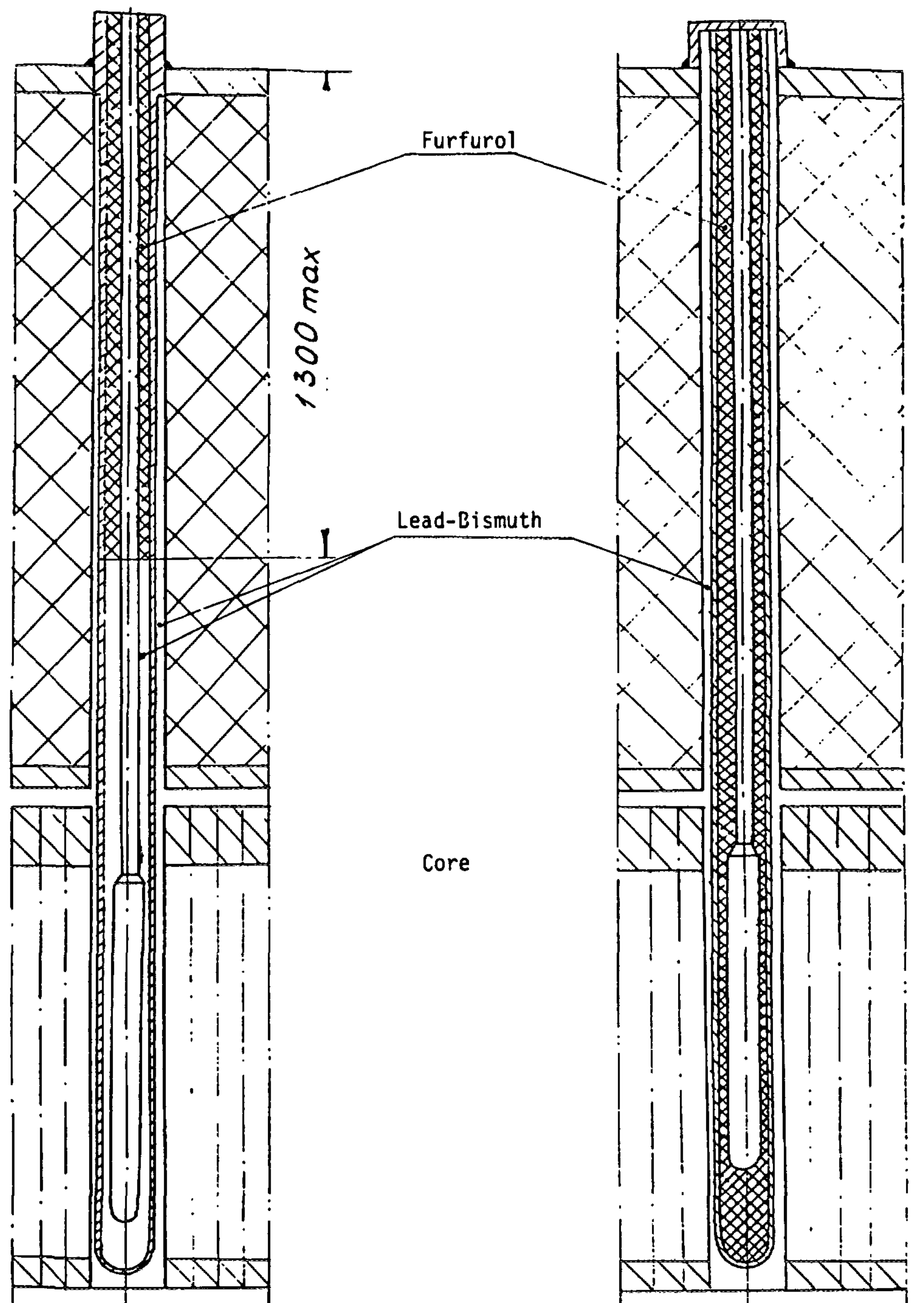


FIG. 5. Plan view of the core and reflector of a liquid metal reactor of submarine factory number 601, showing the approximate position of the emergency protection rods and the control or compensation rods.



Section DD: Channel of a control or compensation rod

Section EE: Channel of an emergency protection rod

FIG. 6. Cross-sections for control or compensation rod and emergency protection rod channels of the type used in submarine factory number 601 [7].

The radial thermal shields consisted of nine concentric cylinders positioned around each reactor core between the outer surface of the reflector region and the inner surface of the RPV wall; the bottom thermal shields consisted of two cylindrical plates positioned below the lower core plate. The nine concentric cylinders were made of an unspecified SS of, from the reflector region surface outward, 11.5-, 34-, 36-, 36-, 36-, 120-, 36-, 36-, and 25-mm thickness, respectively. Cylinders five, 36 mm thickness, and six, 120 mm thickness, were adjacent to one another with the latter containing 24 evenly spaced 80 mm inside diameter holes to accommodate the ECTs. The nine concentric cylinders and the inner surface of the RPV wall were separated by, from the reflector region surface outward, nine annular Pb-Bi coolant channels of 3 mm thickness each. Figure 8 shows the relative layout of the core, BeO reflector, SS thermal shields, Pb-Bi coolant channels, ECTs, and RPV wall in submarine factory number 601. The two cylindrical plates of the bottom thermal shield were made of an unspecified SS of, from the lower core surface downward, 200- and 100-mm thickness, respectively. The lower cylindrical plate was adjacent to the inner surface of the RPV bottom and was separated from the upper cylindrical plate by a 100 mm thick zone of the Pb-Bi coolant.

2.2.3. Icebreaker pressurized water reactors [6, 8]

The entire first SGI, including the SGs and pumps, was located in the center of the ship in an isolated RC. The three PWRs were aligned vertically in a plane perpendicular to the keel and were surrounded by a large water-filled steel shield tank. The biological shield located above the three PWRs was made of limonite concrete and a heat resistant composition of graphite and boron.

Each icebreaker PWR consisted of a cylindrical RPV, a reactor core and its associated support structure, and a series of radial thermal shields. The RPV was made from type 12X2MΦA carbon steel, with approximate dimensions:

(1) 2 m diameter,
(2) 5 m height, and
(3) 100 mm thick walls with a 5 mm thick internal cladding of type 1X18H9T SS.

The height and diameter of each core was approximately 1.6 m and 1 m, respectively, with each core containing 219 cylindrical technical fuel channels (TFCs) (fuel assemblies) on a 64 mm triangular pitch. Within 30 of the TFCs were central channels for six control rods, ganged together in two groups of three, and 24 scram rods, ganged together in six groups of four. Drives for the eight rod groups were situated on the RPV lid. Of the 219 TFCs, 189 contained 36 fuel rods and 30 contained 30 fuel rods. Overall, there were 7,704 fuel rods in each core. The fuel rods were constructed of UO_2 sintered ceramic pellets of 4.5 mm diameter, enriched to 5.0% ^{235}U. The pellets were clad in Zr-Nb alloy or SS of 0.75 mm thickness with a 0.1 mm thick gap. The resultant fuel rods were 6.1 mm outside diameter.

The operating temperatures of the water in the main inlet and outlet tubes, the UO_2 in the center of the fuel rod, the water between the TFCs, and the fuel element cladding at one-half of maximum power were 261°C, 313°C, 1400°C, 280°C, and 700°C, respectively. Water flow rates in the N2 reactor primary circuit bow and stern loops were 458 and 407 $tonne{\cdot}h^{-1}$, respectively, and the operating pressure was 180 $kg{\cdot}cm^{-2}$.

The radial thermal shields consisted of five concentric cylinders positioned around each reactor core. The five concentric cylinders were made of type 8X18H10T SS of, from the core surface outward, 19-, 15-, 21-, 25-, and 25-mm thickness, respectively. The five concentric cylinders and the inner surface of the RPV wall were separated by primary-circuit water of, from the core surface outward, 40 -, 39-, 15-, 7-, 13-, and 10-mm thickness, respectively. Two of these five concentric cylinders were welded to the upper and lower rings, which connected with the upper and lower plates used for the fixation of the TFCs to form the reactor core barrel. The upper and lower plates of the core barrel were also constructed of type 8X18H10T SS and were 250- and 185-mm thick, respectively. Overall, the approximate dimensions of the core barrel were 1.4 m diameter and 3 m height.

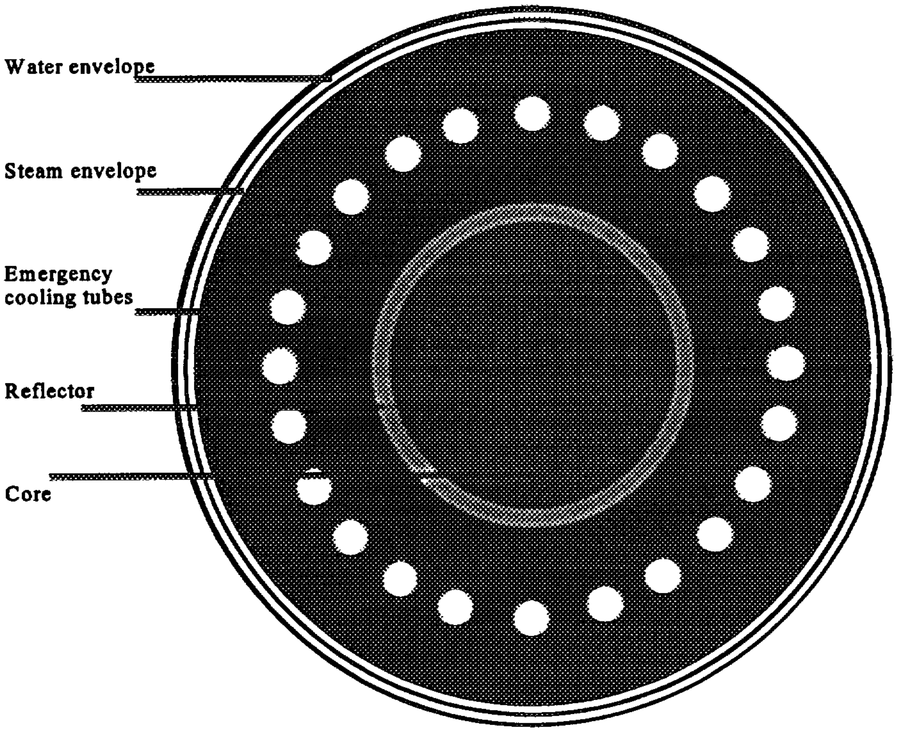


FIG. 7. Plan view of a liquid metal reactor of submarine factory number 601, showing the approximate layout of the emergency cooling tubes and core. For clarity, individual thermal shields and Pb-Bi coolant channels are not shown.

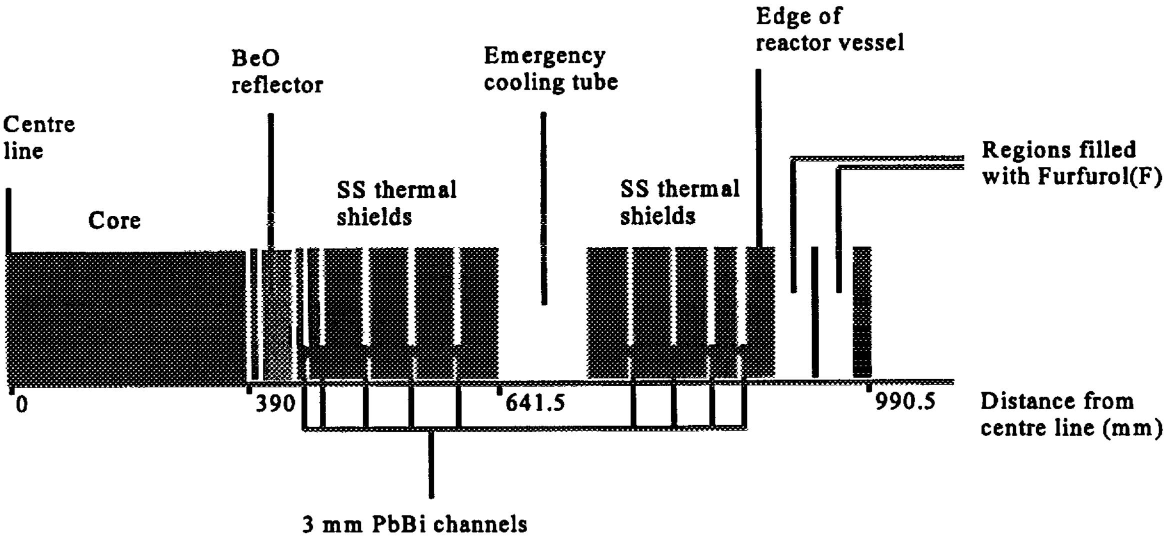


FIG. 8. Radial cross-section of a liquid metal reactor of submarine factory number 601, showing the layout and approximate dimensions of the core, BeO reflector, stainless steel (SS) thermal shields, Pb-Bi coolant channels, emergency cooling tubes, and reactor pressure vessel walls.

2.3. REACTOR OPERATING HISTORIES

Known or estimated reactor operating histories of the seven nuclear submarines and icebreaker are detailed in the following sections.

2.3.1. Submarine pressurized water reactors [3, 6]

Specific information about the operating histories of SGIs associated with the 11 discarded PWRs remains classified. The year of start-up and shutdown and the fuel burnup for each PWR is currently known, as is the fact that all contained their initial fuel load at the time of final shutdown. Start-up dates covered the ten year period from 1958 to 1968, with the earliest date that for submarine factory number 254 and the latest date that for submarine factory number 421. Shutdown dates covered the seven year period from 1961 to 1968, with the earliest date that for submarine factory number 901 and the latest date again that for submarine factory number 421. The longest period of SGI operation was four years and is that for submarine factory number 254. While the shortest period of SGI operation was one year and is that for submarine factory numbers 901 and 421. Fuel burnup for the PWRs varied from a low of 1250 MW·d for submarine factory number 421 to a high of 3880 MW·d for submarine factory number 254.

2.3.2. Submarine liquid metal reactors [7]

Subsequent to the core reload in September 1967, each reactor operated for approximately 300 effective-full-power hours (875 MW·d). The total neutron fluence of all energies on the RPVs and internal reactor constructions during SGI operation was 2.3×10^{20} $\mathrm{n{\cdot}cm^{-2}}$.

2.3.3. Icebreaker pressurized water reactors [6, 8]

The first fuel load lasted from 1959 to 1962 and consisted of 80 kg of ^{235}U in each reactor core. The integrated power productions with this first fuel load were equal to 17.8 GW·d for the N1 PWR and 18.0 GW·d for the N2 and N3 PWRs. Refuelling occurred in 1963. The second fuel load lasted from 1963 to 1965 and consisted of 129 kg ^{235}U clad in SS in the N1 PWR and 75 kg ^{235}U clad in Zr-Nb in the N2 and N3 PWRs. The N2 PWR operated from July 19, 1963 to November 17, 1963 and from June 22, 1964 to November 13, 1964 for a total period of 267 days. The mean power of the N2 PWR during the second fuel load was 53 MW. At the time of the reactor accident, the integrated power production for the N2 PWR was approximately 14.2 GW·d thermal and the burnup was equal to 9.4 $\mathrm{GW{\cdot}d{\cdot}tonne^{-1}}$ initial heavy metal. Integrated power productions for the second fuel load were equal to 22.5 GW·d for the N1 PWR and 17.5 GW·d for the N3 PWR. The total mean neutron fluence on the N2 RPV and its internal reactor constructions during operation of the first SGI with the second fuel load was 5.5×10^{20} $\mathrm{n{\cdot}cm^{-2}}$.

Table II summarizes the current available information for the steam generating installations of the marine reactors dumped in the Kara Sea [3, 4, 6, 7, 8].

TABLE II. CURRENT AVAILABLE INFORMATION FOR THE STEAM GENERATING INSTALLATIONS OF THE MARINE REACTORS DUMPED IN THE KARA SEA [3, 4, 6, 7, 8]

Factory number	Reactor		^{235}U initial conditions[1]		Steam generating installation				RPV[2] with SNF[3]
	Type	Position	Load (kg)	Enrichment (%)	Start-up date	Shutdown date	Burnup (MW·d)	Disposal date	
901	PWR[4]	Left board	50	20	1961	1961	1710	May 1965	Yes
	PWR	Right board	50	20	1961	1961	1670	May 1965	Yes
285	PWR	Left board	50	7.5	1961	1964	2780	Oct 1965	No[5]
	PWR	Right board	50	7.5	1961	1964	2730	Oct 1965	Yes
254	PWR	Left board	50	20	1958	1962	3080	1965	No
	PWR	Right board	50	20	1958	1962	3880	1965	No
260	PWR	Left board	50	20	1959	1962	1720	1966	No
	PWR	Right board	50	20	1959	1962	1940	1966	No
OK-150	PWR	Left board	129[6]	5	Aug 1959	Oct 1965	40 300	Sep 1967	No
	PWR	Center line	75[6]	5	Aug 1959	Feb 1965	32 200[7]	Sep 1967	No[8]
	PWR	Right board	75[6]	5	Aug 1959	Oct 1965	35 500	Sep 1967	No
421	PWR	Right board	50	20	1968	1968	1250	1972	Yes
601	LMR[9]	Left board	90[6]	90	Dec 1962	May 1968	1 580[10]	Sep 1981	Yes
	LMR	Right board	90[6]	90	Dec 1962	Jun 1968	1 580[10]	Sep 1981	Yes
538	PWR	Left board	50	20	1961	1963	1680	1988	No
	PWR	Right board	50	20	1961	1963	1440	1988	No

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[1] Excepting for factory numbers OK-150 and 601, ^{235}U initial conditions are assumed.

[2] Reactor pressure vessel (RPV).

[3] Spent nuclear fuel (SNF) with thermal shields, hardware, and Furfurol(F).

[4] Pressurized water reactor (PWR).

[5] Thermal shields and hardware only.

[6] For the second fuel load.

[7] Burnup for the second fuel load was 14 200 MW·d.

[8] Thermal shields, hardware, and approximately 60% of SNF discarded in special container.

[9] Liquid metal reactor (LMR).

[10] Burnup for the second fuel load was 875 MW·d.

2.4. RADIONUCLIDE INVENTORIES

Two independent estimates were prepared for the radionuclide inventories. One estimate was prepared by consulting members of the Group from the Russian Research Center "Kurchatov Institute" (RRCKI), Moscow [3, 8], and the Institute of Physics and Power Engineering (IPPE), Obninsk [7]. Another estimate was prepared by the State Institute of Applied Ecology (SIAE), Moscow [10]. In preparing the estimates, the Group core models for the PWRs and LMRs were represented by the icebreaker OK-150 and submarine factory number 601 core models, respectively. While, the SIAE estimates used a WWER-1000 core model to represent both the PWR and LMR cores. Values of the fuel burnup used in the Group inventory calculations came from RRCKI and IPPE records; those for SIAE were supplied by the Russian Navy. Computer programs used in both inventory estimates are well established and benchmarked.

Results from the Group estimate, when compared to those from SIAE, showed the following:

(1) fission products are in good agreement for the icebreaker and no worse than a factor of 0.5 for the nuclear submarines, and
(2) actinides agree within a factor of 0.5 for the icebreaker and are no worse than a factor of 0.1 for the nuclear submarines.

Upon consideration of the above, the group concluded that even though SIAE results tend to be higher, and therefore more conservative, they do not represent the best estimate for the IASAP effort. First and foremost, the core models used in the Group estimate for the icebreaker and submarine factory number 601 represent the actual configurations; the SIAE models do not. Second, even though there are differences between the core configurations of the nuclear submarine and icebreaker PWRs, the OK-150 model is more representative of the true core configurations than that of the WWER-1000. This is further substantiated by the fact that comparisons of the Group actinide results to those in other Russian reports [11] indicate differences of no more than ± 20%.

The following sections detail the methods used to estimate the radionuclide inventories of the nuclear submarines and icebreaker. The results are given in Tables III to XVI.

Table III presents a summary of the estimated total 1994 radionuclide inventories of fission products, activation products, and actinides in the marine reactors dumped in the Kara Sea [3, 7, 8, 12].

Tables IV through XVI present individual summaries of the estimated 1994 activity of long-lived radionuclides in the marine reactors dumped in the Kara Sea [3, 7, 8, 12].

2.4.1. Submarine pressurized water reactors [3, 5, 6]

2.4.1.1. Fission products and actinides

Due to classification issues, complete descriptions of the PWR nuclear SGIs are not yet available. Thus, even though the fuel for the submarine PWRs and the icebreaker were described as a U-Al alloy and UO_2, respectively, the model for the icebreaker SGI was assumed for the PWR SGIs. Fission product activities for the nuclear submarine PWRs are based on the quotient of the nuclear submarine and icebreaker burnups, and are given by the following equation:

$$A_i(S) = A_i(I)\left(\frac{B(S)}{B(I)}\right) \tag{1}$$

where

$A_i(S)$ is the activity of the i-th radionuclide in the nuclear submarine (Bq),
$A_i(I)$ is the activity of the i-th radionuclide in the icebreaker (Bq),

B(S) is the burnup in the nuclear submarine (MW·d), and
B(I) is the burnup in the icebreaker (MW·d).

Actinide activities for the nuclear submarine PWRs are based on the product of the quotients of the icebreaker and nuclear submarine enrichments and the nuclear submarine and icebreaker burnups, and are given by the following equation:

$$A_i(S) = A_i(I)\left(\frac{E(I)}{E(S)}\right)\left(\frac{B(S)}{B(I)}\right) \tag{2}$$

where

E(I) is the ^{235}U enrichment in the icebreaker (%),
E(S) is the ^{235}U enrichment in the nuclear submarine (%), and

all other terms are as previously defined.

2.4.1.2. Activation products

Again because of classification issues, activation product activities for the PWRs are based on a calculational procedure similar to that used for the icebreaker. For conservatism, it was assumed that

(1) all internal reactor constructions were made from SS;
(2) the RPVs were made from low alloy steel;
(3) the fuel elements were clad in SS;
(4) the total mass of SS in the internal reactor constructions was some 100 kg greater than that associated with the icebreaker disposal;
(5) even though some RPVs dumped without SNF also had their thermal shields and internal hardware removed, all of the activated metal remained for disposal; and
(6) neutron activation of the internal reactor constructions and RPVs was at a constant level over the operating period of the SGIs.

Suffice it to say, that almost all activation products associated with the nuclear submarine PWRs originated in the internal constructions made from SS.

Analysis of neutron activation shows that only four radionuclides are of consequence at ten or more years after reactor shutdown. They are ^{14}C, ^{60}Co, ^{59}Ni, and ^{63}Ni. Their activities were estimated from the following equation:

$$A_i = CPTN_i(\sigma_i + gI_i)\left(\frac{\lambda}{\Sigma_{235f}}\right) \tag{3}$$

where

A_i is the activity of the i-th activation product radionuclide (Bq),
C is the conversion factor between fission rate and thermal power (3.1×10^{16} fissions·s^{-1}·MW^{-1}),
P is the mean thermal power of the reactor (MW),
T is the irradiation time (d),
N_i is the number of nuclei of the i-th target nuclide, ^{13}C, ^{59}Co, ^{58}Ni, or ^{62}Ni (cm^{-3}),
σ_i is the microscopic activation cross-section for the i-th target nuclide (cm^2)
g is a relative measure of neutron spectrum hardness,
I_i is the resonance integral for the i-th target nuclide (cm^2),
λ is the radioactive decay constant of the i-th activation product radionuclide, ^{14}C, ^{60}Co, ^{59}Ni, or ^{63}Ni (d^{-1}), and
Σ_{235f} is the macroscopic fission cross-section for ^{235}U (cm^{-1}).

In the calculations, the fuel burnup, in MW·d, is substituted for the product PT, and g is assumed to equal 0.2.

2.4.2. Submarine liquid metal reactors [7, 9]

2.4.2.1. Fission products and actinides

Fission product and actinide activities for the nuclear submarine LMRs are based on their core histories and calculated neutron spectra. Calculation of the fission product and actinide inventories were accomplished with the AFPA [13] and CARE [14] codes, respectively. The latter code considers the formation and decay of 93 individual actinide chains and uses the BNAB [15] nuclear cross-section system.

In modeling the left board reactor with its damaged SNF, it is assumed that 80% of the fission products and actinides are in the RPV; the balance are located at the top of the SG.

2.4.2.2. Activation products

For the SS components of the nuclear submarine LMRs, the long-lived activation products of consequence are ^{60}Co, ^{59}Ni, and ^{63}Ni. Their activities were estimated as follows:

$$A_i = V\rho_j\sum_{j=1}^{21}\sigma_{ij}\overline{\Phi}_j(1 - e^{-\lambda T}) \qquad (4)$$

where

V	is volume of the structural material considered (cm^3),
ρ_i	is the concentration of the i-th target nuclide, ^{59}Co, ^{58}Ni, or ^{62}Ni, in the structural material considered (cm^{-3}),
$\sum_{j=1}^{21}$	is the summation over the 21 neutron energy group approximation,
σ_{ij}	is the microscopic activation cross-section of the i-th target nuclide for neutrons of the j-th energy group (cm^2),
$\overline{\Phi}_j$	is the mean thermal neutron flux density in the material for neutrons of the j-th energy group, (cm^{-2}·s^{-1}), and

all other terms are as previously defined. In these calculations, it was assumed that the mean reactor power during operation was approximately 10% of the nominal power and that the irradiation time was 10 times greater than the core burnup in effective-full-power hours. As such, the value of $\overline{\Phi}_j$ used was assumed to be 10 times less than the nominal power neutron flux density in the material for neutrons of the j-th energy group, $\Phi_{j_{Nominal}}$. Values of $\Phi_{j_{Nominal}}$ in the core and the thermal shields were calculated in a 21-group approximation and were those from the core design analyses.

The use of Pb-Bi as a coolant and Eu in the CCRs results in the production of other long-lived radionuclides such as ^{205}Pb, ^{207}Bi, ^{208}Bi, ^{210m}Bi, ^{152}Eu, and ^{154}Eu. Their activities were calculated using the approach described.

Typically, the distribution of LMR primary circuit activation products was approximately 30% in the SG, 40% in the RPV, and the balance throughout the remainder of the circuit and the volume compensator. However, in modeling the two LMRs, conservative assumptions were made. For the left board reactor where the accident occurred, 30% of the primary circuit activation products are assumed to be in the SG and the balance are in the RPV. For the right board reactor, all of the primary circuit activation products are assumed to be in the RPV.

2.4.3. Icebreaker pressurized water reactors [8]

2.4.3.1. Fission products and actinides

Calculations of core cell burn-up in the N2 PWR were performed by the spectral code GETERA [16]. The fission product and actinide activities were estimated using the RECOL [17] library data base, which was generated on the bases of the latest versions of the evaluated nuclear data files, ENDF/B-V, with corrections based on the results of critical experiments [18]. The criticality problem was solved for a realistic 3-D geometry model of a TFC by Monte-Carlo with RECOL and checked with MCNP [19] for fresh fuel load. One-group cross-sections were prepared for burn-up calculation of critical loads of both fresh and spent fuel and input to ORIGEN-2 [20] for detailed radionuclide content calculations.

2.4.3.2. Activation products

All internal reactor constructions were made from type 1X18H9T SS; the RPVs were made from type 12X2MΦA carbon steel. Almost all activation products associated with the icebreaker PWRs originated in the internal constructions made from SS.

Analysis of neutron activation shows that only four radionuclides are of consequence at ten or more years after reactor shutdown. These are ^{14}C, ^{60}Co, ^{59}Ni, and ^{63}Ni. Activities of each were estimated from the following equation:

$$A_i = CP\left[\rho_m + \rho_a + \Delta K\left(\frac{\Sigma_s}{\Sigma_{s+w}}\right)\right]T\lambda\Psi \qquad (5)$$

where

ρ_m is the fraction of the neutron flux captured in the main reactivity compensating lattices per unit fission;

ρ_a is the fraction of the neutron flux captured in the additional reactivity compensating lattices per unit fission;

ΔK is the fraction of the neutron flux captured in the shielding assemblage and lower plate per unit fission;

Σ_s is the macroscopic capture cross-section for thermal and intermediate neutrons in the steel of the shielding assemblage (cm^{-1}),

Σ_{s+w} is the macroscopic capture cross-section for thermal and intermediate neutrons in the steel and water of the shielding assemblage (cm^{-1}),

Ψ is the ratio of the activation cross-section of ^{13}C, ^{59}Co, ^{58}Ni, or ^{62}Ni to the total activation cross-section of the steel under irradiation, and

all other terms are as previously defined. Again, in the calculations, the fuel burnup, in MW·d, is substituted for the product PT.

Text cont. on p. 36.

TABLE III. ESTIMATED 1994 RADIONUCLIDE INVENTORIES OF FISSION PRODUCTS, ACTIVATION PRODUCTS, AND ACTINIDES IN THE MARINE REACTORS DUMPED IN THE KARA SEA [3, 7, 8, 12]

Factory number	Activity in 1994							
	Fission products		Activation products		Actinides		Total	
	Becquerel (Bq)	Percent (%)	Becquerel (Bq)	Percent (%)	Becquerel (Bq)	Percent (%)	Becquerel (Bq)	Percent (%)
901	7.2×10^{14}	15	6.0×10^{12}	0.13	3.4×10^{12}	0.073	7.3×10^{14}	15
285	6.3×10^{14}	13	1.3×10^{13}	0.27	8.1×10^{12}	0.17	6.5×10^{14}	14
254	-	-	9.5×10^{12}	0.20	-	-	9.5×10^{12}	0.20
260	-	-	5.1×10^{12}	0.11	-	-	5.1×10^{12}	0.11
OK-150[1]	1.8×10^{15}	39	2.3×10^{14}	5.0	8.3×10^{13}	1.8	2.2×10^{15}	46
421	2.9×10^{14}	6.1	2.9×10^{12}	0.062	2.8×10^{12}	0.061	2.9×10^{14}	6.2
601	5.3×10^{14}	11	3.0×10^{14}	6.5	3.6×10^{11}	0.008	8.4×10^{14}	18
538	-	-	4.5×10^{12}	0.096	-	-	4.5×10^{12}	0.096
Total	4.0×10^{15}	86	5.7×10^{14}	12	9.7×10^{13}	2.1	4.7×10^{15}	100

[1] The fission product, actinide, and twenty-seven percent of activation product activities were discarded in a reinforced concrete and stainless steel container.

TABLE IV. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE LEFT BOARD REACTOR PRESSURE VESSEL WITHIN THE REACTOR COMPARTMENT FROM NUCLEAR SUBMARINE FACTORY NUMBER 901 DUMPED IN THE ABROSIMOV FJORD [3, 12]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Fission products		
^{3}H	1.4×10^{11}	0.038
^{85}Kr	2.8×10^{12}	0.78
^{90}Sr	8.5×10^{13}	23
^{90}Y	8.5×10^{13}	23
^{99}Tc	2.6×10^{10}	0.0073
^{125}Sb	4.8×10^{9}	0.0013
^{129}I	2.6×10^{7}	0.0000071
^{137}Cs	9.3×10^{13}	26
^{137}Ba^{m}	9.0×10^{13}	25
^{147}Pm	9.8×10^{10}	0.027
^{151}Sm	2.1×10^{12}	0.58
Subtotal	3.6×10^{14}	99
Activation products		
^{14}C	1.4×10^{10}	0.0038
^{60}Co	5.3×10^{11}	0.15
^{59}Ni	1.4×10^{11}	0.038
^{63}Ni	2.3×10^{12}	0.63
Subtotal	3.0×10^{12}	0.82
Actinides		
^{238}Pu	1.4×10^{11}	0.039
^{239}Pu	1.2×10^{11}	0.032
^{240}Pu	5.3×10^{10}	0.014
^{241}Pu	1.2×10^{12}	0.34
^{241}Am	1.7×10^{11}	0.048
Subtotal	1.7×10^{12}	0.47
Total	3.6×10^{14}	100

TABLE V. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE RIGHT BOARD REACTOR PRESSURE VESSEL WITHIN THE REACTOR COMPARTMENT FROM NUCLEAR SUBMARINE FACTORY NUMBER 901 DUMPED IN THE ABROSIMOV FJORD [3, 12]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Fission products		
^{3}H	1.4×10^{11}	0.038
^{85}Kr	2.8×10^{12}	0.78
^{90}Sr	8.5×10^{13}	23
^{90}Y	8.5×10^{13}	23
^{99}Tc	2.6×10^{10}	0.0073
^{125}Sb	4.8×10^{9}	0.0013
^{129}I	2.6×10^{7}	0.0000071
^{137}Cs	9.3×10^{13}	26
$^{137}Ba^{m}$	9.0×10^{13}	25
^{147}Pm	9.8×10^{10}	0.027
^{151}Sm	2.1×10^{12}	0.58
Subtotal	3.6×10^{14}	99
Activation products		
^{14}C	1.4×10^{10}	0.0038
^{60}Co	5.3×10^{11}	0.15
^{59}Ni	1.4×10^{11}	0.038
^{63}Ni	2.3×10^{12}	0.63
Subtotal	3.0×10^{12}	0.82
Actinides		
^{238}Pu	1.4×10^{11}	0.039
^{239}Pu	1.2×10^{11}	0.032
^{240}Pu	5.3×10^{10}	0.014
^{241}Pu	1.2×10^{12}	0.34
^{241}Am	1.7×10^{11}	0.048
Subtotal	1.7×10^{12}	0.47
Total	3.6×10^{14}	100

TABLE VI. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE LEFT BOARD REACTOR PRESSURE VESSEL WITHIN THE REACTOR COMPARTMENT FROM NUCLEAR SUBMARINE FACTORY NUMBER 285 DUMPED IN THE ABROSIMOV FJORD [3, 12]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Fission products		
^{3}H	2.8×10^{12}	0.44
^{85}Kr	5.7×10^{12}	0.88
^{90}Sr	1.5×10^{14}	23
^{90}Y	1.5×10^{14}	23
^{99}Tc	4.4×10^{10}	0.0067
^{125}Sb	1.8×10^{10}	0.0028
^{129}I	4.2×10^{7}	0.0000065
^{137}Cs	1.6×10^{14}	25
^{137}Ba^{m}	1.6×10^{14}	24
^{147}Pm	3.7×10^{11}	0.056
^{151}Sm	3.6×10^{12}	0.55
Subtotal	6.3×10^{14}	98
Activation products		
^{14}C	1.9×10^{10}	0.030
^{60}Co	9.5×10^{11}	0.15
^{59}Ni	1.7×10^{12}	0.26
^{63}Ni	2.8×10^{12}	0.43
Subtotal	5.4×10^{12}	0.84
Actinides		
^{238}Pu	2.4×10^{11}	0.037
^{239}Pu	5.1×10^{11}	0.078
^{240}Pu	2.3×10^{11}	0.035
^{241}Pu	6.4×10^{12}	0.99
^{241}Am	7.2×10^{11}	0.11
Subtotal	8.1×10^{12}	1.3
Total	6.5×10^{14}	100

TABLE VII. ESTIMATED 1994 ACTIVITY OF LONG-LIVED ACTIVATION PRODUCTS IN THE RIGHT BOARD REACTOR PRESSURE VESSEL WITHIN THE REACTOR COMPARTMENT FROM NUCLEAR SUBMARINE FACTORY NUMBER 285 DUMPED IN THE ABROSIMOV FJORD [3]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Activation products		
^{14}C	2.6×10^{10}	0.36
^{60}Co	1.3×10^{12}	17
^{59}Ni	2.3×10^{12}	31
^{63}Ni	3.8×10^{12}	51
Subtotal	7.3×10^{12}	100

TABLE VIII. ESTIMATED 1994 ACTIVITY OF LONG-LIVED ACTIVATION PRODUCTS IN THE LEFT AND RIGHT BOARD REACTOR PRESSURE VESSELS WITHIN THE REACTOR COMPARTMENT FROM NUCLEAR SUBMARINE FACTORY NUMBER 254 DUMPED IN THE ABROSIMOV FJORD [3]

Radionuclide	**Activity in 1994**	
	Becquerel (Bq)	**Percent (%)**
Left board reactor		
^{14}C	2.2×10^{10}	0.23
^{60}Co	8.3×10^{11}	8.8
^{59}Ni	1.9×10^{11}	2.0
^{63}Ni	3.1×10^{12}	33
Subtotal	4.2×10^{12}	44
Right board reactor		
^{14}C	2.8×10^{10}	0.29
^{60}Co	1.1×10^{12}	11
^{59}Ni	2.4×10^{11}	2.5
^{63}Ni	4.0×10^{12}	42
Subtotal	5.3×10^{12}	56
Total	9.5×10^{12}	100

TABLE IX. ESTIMATED 1994 ACTIVITY OF LONG-LIVED ACTIVATION PRODUCTS IN THE LEFT AND RIGHT BOARD REACTOR PRESSURE VESSELS WITHIN THE REACTOR COMPARTMENT FROM NUCLEAR SUBMARINE FACTORY NUMBER 260 DUMPED IN THE ABROSIMOV FJORD [3]

Radionuclide	**Activity in 1994**	
	Becquerel (Bq)	**Percent (%)**
Left board reactor		
^{14}C	1.3×10^{10}	0.25
^{60}Co	4.6×10^{11}	9.2
^{59}Ni	1.1×10^{11}	2.1
^{63}Ni	1.8×10^{12}	35
Subtotal	2.4×10^{12}	47
Right board reactor		
^{14}C	1.4×10^{10}	0.27
^{60}Co	5.3×10^{11}	10
^{59}Ni	1.2×10^{11}	2.3
^{63}Ni	2.0×10^{12}	40
Subtotal	2.7×10^{12}	53
Total	5.1×10^{12}	100

TABLE X. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES FROM THE N2 REACTOR OF THE ICEBREAKER PLACED WITHIN THE CONCRETE AND STAINLESS STEEL SHELL CONTAINER DUMPED IN THE TSIVOLKA FJORD [8, 12]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Fission products		
$^{3}\mathrm{H}$	8.3×10^{11}	0.042
$^{85}\mathrm{Kr}$	1.7×10^{13}	0.83
$^{90}\mathrm{Sr}$	4.4×10^{14}	22
$^{90}\mathrm{Y}$	4.4×10^{14}	22
$^{99}\mathrm{Tc}$	1.3×10^{11}	0.0068
$^{125}\mathrm{Sb}$	5.8×10^{10}	0.0029
$^{129}\mathrm{I}$	2.1×10^{8}	0.000011
$^{137}\mathrm{Cs}$	4.8×10^{14}	24
$^{137}\mathrm{Ba}^{m}$	4.6×10^{14}	23
$^{147}\mathrm{Pm}$	1.4×10^{12}	0.069
$^{151}\mathrm{Sm}$	1.1×10^{13}	0.55
Subtotal	1.8×10^{15}	93
Activation products		
$^{14}\mathrm{C}$	4.3×10^{5}	0.000000022
$^{60}\mathrm{Co}$	1.1×10^{13}	0.57
$^{59}\mathrm{Ni}$	4.9×10^{11}	0.025
$^{63}\mathrm{Ni}$	5.0×10^{13}	2.5
Subtotal	6.1×10^{13}	3.1
Actinides		
$^{238}\mathrm{Pu}$	1.0×10^{12}	0.053
$^{239}\mathrm{Pu}$	5.0×10^{12}	0.25
$^{240}\mathrm{Pu}$	2.3×10^{12}	0.11
$^{241}\mathrm{Pu}$	6.7×10^{13}	3.4
$^{241}\mathrm{Am}$	7.1×10^{12}	0.36
Subtotal	8.3×10^{13}	4.2
Total	2.0×10^{15}	100

TABLE XI. ESTIMATED 1994 ACTIVITY OF LONG-LIVED ACTIVATION PRODUCTS IN THE THREE REACTOR PRESSURE VESSELS WITHIN THE REACTOR COMPARTMENT FROM THE ICEBREAKER DUMPED IN THE TSIVOLKA FJORD [8, 12]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Activation products		
$^{14}\mathrm{C}$	1.2×10^{6}	0.00000070
$^{60}\mathrm{Co}$	3.1×10^{13}	19
$^{59}\mathrm{Ni}$	1.3×10^{12}	0.79
$^{63}\mathrm{Ni}$	1.3×10^{14}	81
Subtotal	1.7×10^{14}	100

TABLE XII. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE METAL CONTAINER ENCASED SEPARATED RIGHT BOARD REACTOR PRESSURE VESSEL FROM NUCLEAR SUBMARINE FACTORY NUMBER 421 DUMPED IN THE NOVAYA ZEMLYA DEPRESSION [3, 12]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Fission products		
^{3}H	1.6×10^{11}	0.055
^{85}Kr	3.3×10^{12}	1.1
^{90}Sr	7.4×10^{13}	25
^{90}Y	7.4×10^{13}	25
^{99}Tc	1.9×10^{10}	0.0066
^{125}Sb	2.2×10^{10}	0.0076
^{129}I	1.9×10^{7}	0.0000065
^{137}Cs	6.8×10^{13}	23
^{137}Ba^{m}	6.5×10^{13}	22
^{147}Pm	4.7×10^{11}	0.16
^{151}Sm	1.7×10^{12}	0.57
Subtotal	2.9×10^{14}	98
Activation products		
^{14}C	1.2×10^{10}	0.0042
^{60}Co	9.8×10^{11}	0.34
^{59}Ni	1.0×10^{11}	0.035
^{63}Ni	1.8×10^{12}	0.61
Subtotal	2.9×10^{12}	0.98
Actinides		
^{238}Pu	8.7×10^{10}	0.030
^{239}Pu	1.1×10^{11}	0.039
^{240}Pu	3.9×10^{10}	0.013
^{241}Pu	2.4×10^{12}	0.83
^{241}Am	1.6×10^{11}	0.056
Subtotal	2.8×10^{12}	0.97
Total	2.9×10^{14}	100

TABLE XIII. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE LEFT BOARD REACTOR PRESSURE VESSEL WITHIN THE REACTOR COMPARTMENT OF NUCLEAR SUBMARINE FACTORY NUMBER 601 DUMPED IN THE STEPOVOY FJORD [7]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	Percent (%)
Fission products		
^{3}H	1.9×10^{13}	5.3
^{85}Kr	2.9×10^{12}	0.79
^{90}Sr	4.6×10^{13}	13
^{90}Y	4.6×10^{13}	13
^{99}Tc	1.2×10^{10}	0.0033
^{125}Sb	2.4×10^{10}	0.0064
^{129}I	2.6×10^{7}	0.0000071
^{137}Cs	5.0×10^{13}	14
^{137}Ba^{m}	4.8×10^{13}	13
^{147}Pm	3.7×10^{9}	0.0010
^{151}Sm	1.5×10^{12}	0.40
^{155}Eu	3.8×10^{10}	0.011
Subtotal	2.1×10^{14}	58
Activation products		
Control/compensation rods		
^{152}Eu	3.0×10^{13}	8.3
^{154}Eu	6.0×10^{12}	1.4
Primary circuit medium		
^{60}Co	4.2×10^{8}	0.00011
^{59}Ni	5.2×10^{6}	0.0000014
^{63}Ni	5.4×10^{8}	0.00015
^{205}Pb	6.7×10^{7}	0.000018
^{207}Bi	6.0×10^{9}	0.0016
^{208}Bi	2.2×10^{9}	0.00060
^{210}Bi^{m}	1.2×10^{9}	0.00032
Core steel components		
^{60}Co	6.3×10^{9}	0.0017
^{59}Ni	2.9×10^{7}	0.0000079
^{63}Ni	2.4×10^{9}	0.00066
Core thermal shields		
^{60}Co	4.6×10^{13}	13
^{59}Ni	7.8×10^{11}	0.21
^{63}Ni	6.9×10^{13}	19
Reactor pressure vessel		
^{60}Co	2.5×10^{11}	0.068
^{59}Ni	1.8×10^{9}	0.00051
^{63}Ni	1.1×10^{11}	0.030
Subtotal	1.5×10^{14}	42
Actinides		
^{238}Pu	3.4×10^{9}	0.00093
^{239}Pu	1.4×10^{11}	0.037
^{240}Pu	2.6×10^{9}	0.00071
^{241}Pu	1.1×10^{9}	0.00030
^{241}Am	4.6×10^{8}	0.00013
Subtotal	1.4×10^{11}	0.039
Total	3.7×10^{14}	100

TABLE XIV. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE STEAM GENERATOR OF THE LEFT BOARD REACTOR WITHIN THE REACTOR COMPARTMENT OF NUCLEAR SUBMARINE FACTORY NUMBER 601 DUMPED IN THE STEPOVOY FJORD [7]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Fission products		
^{3}H	4.8×10^{12}	9.0
^{85}Kr	7.2×10^{11}	1.4
^{90}Sr	1.2×10^{13}	22
^{90}Y	1.2×10^{13}	22
^{99}Tc	3.0×10^{9}	0.0056
^{125}Sb	5.9×10^{9}	0.011
^{129}I	6.5×10^{6}	0.000012
^{137}Cs	1.3×10^{13}	24
^{137}Ba^{m}	1.2×10^{13}	22
^{147}Pm	9.2×10^{8}	0.0017
^{151}Sm	3.7×10^{11}	0.69
^{155}Eu	9.6×10^{9}	0.018
Subtotal	5.3×10^{13}	100
Activation products		
Primary circuit medium		
^{60}Co	1.8×10^{8}	0.00034
^{59}Ni	2.2×10^{6}	0.0000042
^{63}Ni	2.3×10^{8}	0.00043
^{205}Pb	2.9×10^{7}	0.000054
^{207}Bi	2.6×10^{9}	0.0048
^{208}Bi	9.4×10^{8}	0.0018
^{210}Bi^{m}	5.0×10^{8}	0.00094
Subtotal	4.4×10^{9}	0.0083
Actinides		
^{238}Pu	8.5×10^{8}	0.0016
^{239}Pu	3.4×10^{10}	0.063
^{240}Pu	6.5×10^{8}	0.0012
^{241}Pu	2.8×10^{8}	0.00052
^{241}Am	1.1×10^{8}	0.00021
Subtotal	3.6×10^{10}	0.067
Total	5.3×10^{13}	100

TABLE XV. ESTIMATED 1994 ACTIVITY OF LONG-LIVED RADIONUCLIDES IN THE RIGHT BOARD REACTOR PRESSURE VESSEL WITHIN THE REACTOR COMPARTMENT OF NUCLEAR SUBMARINE FACTORY NUMBER 601 DUMPED IN THE STEPOVOY FJORD [7]

Radionuclide	**Activity in 1994**	
	Becquerel (Bq)	**Percent (%)**
Fission products		
^{3}H	2.4×10^{13}	5.7
^{85}Kr	3.6×10^{12}	0.86
^{90}Sr	5.8×10^{13}	14
^{90}Y	5.8×10^{13}	14
^{99}Tc	1.5×10^{10}	0.0036
^{125}Sb	2.9×10^{10}	0.0070
^{129}I	3.2×10^{7}	0.0000077
^{137}Cs	6.3×10^{13}	15
^{137}Ba^{m}	6.0×10^{13}	14
^{147}Pm	4.6×10^{9}	0.0011
^{151}Sm	1.8×10^{12}	0.44
^{155}Eu	4.8×10^{10}	0.011
Subtotal	2.7×10^{14}	64
Activation products		
Control/compensation rods		
^{152}Eu	3.0×10^{13}	7.2
^{154}Eu	5.0×10^{12}	1.2
Primary circuit medium		
^{60}Co	6.0×10^{8}	0.00014
^{59}Ni	7.5×10^{6}	0.0000018
^{63}Ni	7.7×10^{8}	0.00018
^{205}Pb	9.6×10^{7}	0.000023
^{207}Bi	8.5×10^{9}	0.0020
^{208}Bi	3.1×10^{9}	0.00075
^{210}Bi^{m}	1.7×10^{9}	0.00040
Core steel components		
^{60}Co	6.3×10^{9}	0.0015
^{59}Ni	2.9×10^{7}	0.0000069
^{63}Ni	2.4×10^{9}	0.00057
Core thermal shields		
^{60}Co	4.6×10^{13}	11
^{59}Ni	7.8×10^{11}	0.18
^{63}Ni	6.9×10^{13}	17
Reactor pressure vessel		
^{60}Co	2.5×10^{11}	0.060
^{59}Ni	1.8×10^{9}	0.00044
^{63}Ni	1.1×10^{11}	0.026
Subtotal	1.5×10^{14}	36
Actinides		
^{238}Pu	4.3×10^{9}	0.0010
^{239}Pu	1.7×10^{11}	0.040
^{240}Pu	3.3×10^{9}	0.00078
^{241}Pu	1.4×10^{9}	0.00033
^{241}Am	5.7×10^{8}	0.00014
Subtotal	1.8×10^{11}	0.043
Total	4.2×10^{14}	100

TABLE XVI. ESTIMATED 1994 ACTIVITY OF LONG-LIVED ACTIVATION PRODUCTS IN THE METAL CONTAINER ENCASED SEPARATED LEFT AND RIGHT BOARD REACTOR PRESSURE VESSELS FROM NUCLEAR SUBMARINE FACTORY NUMBER 538 DUMPED IN THE TECHENIYE FJORD [3]

Radionuclide	Activity in 1994	
	Becquerel (Bq)	**Percent (%)**
Left board reactor		
^{14}C	1.3×10^{9}	0.028
^{60}Co	5.3×10^{11}	12
^{59}Ni	1.0×10^{11}	2.3
^{63}Ni	1.8×10^{12}	39
Subtotal	2.4×10^{12}	54
Right board reactor		
^{14}C	1.1×10^{9}	0.024
^{60}Co	4.8×10^{11}	11
^{59}Ni	8.9×10^{10}	2.0
^{63}Ni	1.5×10^{12}	34
Subtotal	2.1×10^{12}	46
Total	4.5×10^{12}	100

2.5. DISPOSAL OPERATIONS

The following sections detail what is currently known about the disposal operations related to the seven nuclear submarines and the icebreaker.

2.5.1. Submarine pressurized water reactors [3, 4, 5, 6]

With the exception of the right board RPV from submarine factory number 421 and the two PWRs from submarine factory number 538, all PWRs were dumped in their separated RCs. The SNF was removed from the left board RPV of submarine factory number 285 and both RPVs of submarine factory numbers 254, 260, and 538. The SNF remained in the right board RPV of submarine factory number 285, the right board RPV of submarine factory number 421, and both RPVs of submarine factory number 901.

Before disposal, the primary circuit loops and equipment of all PWRs were washed, dried, and sealed. However, there is no indication that the seals were hermetic. Those RPVs containing SNF were filled with Furfurol(F), a hardening compound based on furfural, prior to disposal. Before filling each RPV with Furfurol(F), the 30 CCR guide tubes were sealed and a 10 mm diameter breather hole was drilled through the upper wall of two. During filling, the RPV was heated, one breather hole was used as the inlet and one breather hole was used as the outlet. Once the process was completed, each 10 mm diameter breather hole was capped with 2.5 mm thick weld.

The shallow waters of Abrosimov Fjord were used for four separate disposal operations. Separated RCs from submarine factory numbers 901, 285, and 254 were dumped in 1965 at estimated depths of 50 m [4], 20 m [1, 4], and 20 m [1], respectively. In 1966, the separated RC from submarine factory number 260 was also dumped at an estimated depth of 20 m [1].

At the time of disposal, the RCs were allowed to flood thereby exposing a significant portion of the external surface of each RPV and the cavities and internal constructions of those RPVs without SNF to sea water. As such, sea water is assumed to have been within the left board RPV of submarine factory number 285, and both RPVs of submarine factory numbers 254 and 260 for 30, 30, and 29 years, respectively.

The right board RPV, with its SNF, was removed from the RC of submarine factory number 421, placed into a steel collar-like support structure within the hull of a barge, and covered with concrete. The concrete layer above the RPV lid was about 200 mm thick. The concrete between the outer surface of the RPV wall and the inner surface of the barge hull was no less than 800 m thick.

In 1972, the barge containing the right board RPV of submarine factory number 421 was dumped in the Novaya Zemlya Depression at an estimated depth of 300 m [1, 4].

Both RPVs, their associated SGs, and their associated primary circuit pumps, were removed from the RC of submarine factory number 538 and placed into a steel collar-like support structure within the hull of a barge. The RPV lids and all penetrations into the lids were sealed by welding. No other protective barriers were provided.

The barge containing both RPVs, their associated SGs, and their associated primary circuit pumps of submarine factory number 538 was sunk in the shallow waters of Techeniye Fjord in 1988 at an estimated depth of 35-40 m [1]. The external surfaces, cavities, and internal constructions of each RPV are assumed to have been exposed to sea water since the time of disposal, a period of about 17 years.

2.5.2. Submarine liquid metal reactors [6, 7, 9]

The SNF remained in the two LMRs of submarine factory number 601. Before disposal, a number of actions were taken. The following is a summary of those actions:

(1) solidified Pb-Bi was removed from all sections of the undamaged right board reactor primary circuit except the RPV;
(2) solidified Pb-Bi remained in all sections of the damaged left board reactor primary circuit;
(3) control rods were permanently fixed in the cores, their drive mechanisms were removed, and the upper ends above the reactor lids were cut away;
(4) channels containing the CCRs were filled with Furfurol(F) in the regions above the Pb-Bi;
(5) channels containing the EPRs were filled with Furfurol(F) in their entirety and sealed by welding steel covers to the upper surface of the reactor lids;
(6) cylindrical channel regions external to the RPV that were formed through the addition of two concentric cylindrical steel shells were filled with Furfurol(F);
(7) SGs of the damaged left board reactor were filled with Furfurol(F) via the secondary circuit;
(8) 20 mm thick elliptic covers were welded to the top of each reactor lid and the volumes between the covers and lids, a maximum of 490 mm thick, filled with bitumen; and
(9) the volumes of the structures containing the SGs, circulation pumps, and volume compensators, the tank of the lead-water shield, and the RC to a level of 240 mm above the elliptic covers were filled with bitumen.

Overall, some 2 m^3 of Furfurol(F) and 250 m^3 of bitumen were used to prepare the RC for disposal.

In September 1981, over 13 years after the reactor accident, submarine factory number 601 was sunk in the shallow waters of Stepovoy Fjord at an estimated depth of 50 m [1]. At the time of her sinking, the hatches of the RC were open. As such, sea water has been in the compartment above the bitumen filler for over 14 years.

2.5.3. Icebreaker pressurized water reactors [4, 5, 8]

2.5.3.1. Reactor compartment

All SNF and the core barrel from the N2 reactor were removed from the three RPVs. Before disposal, the primary circuit loops and equipment were washed, dried, and sealed; and the ceiling of the RC was equipped with special pressure relief valves.

The icebreaker, with the RC aboard, was towed from Murmansk to Tsivolka Fjord for the disposal operations. On September 19, 1967, the RC with three RPVs was dumped in the shallow water of Tsivolka Fjord at an estimated depth of 60 m directly from the icebreaker through the bottom of the hull. The disposal site was approximately one kilometer from the site that was used for the damaged SNF and core barrel from the N2 reactor.

2.5.3.2. Core barrel and spent nuclear fuel

As a consequence of the accident, only 94 of the 219 TFCs from the N2 RPV could be disposed of in a normal manner. The remaining 125 TFCs and the core barrel from the N2 RPV, hereafter known as Configuration A, were placed within a reinforced concrete and SS shell container, hereafter known as Container B. Figure 9 shows a schematic cross-section of Configuration A. Container B consisted of the following constructions:

(1) a primary inner steel liner of 1.5 m inside diameter, 50 mm thick walls, and 3.55 m height;
(2) a secondary inner steel liner in the region of the active core of 1.6 m inside diameter, 30 mm thick walls, and 2.4 m height;
(3) 100 mm thick lid;
(4) an outer steel shell of 2.7 m outside diameter, 5 mm thick walls, and 3.55 m height;
(5) an inner concrete annulus of 545- to 515 mm thickness; and
(6) a 100 mm thick bottom.

The voids within the cavity of Container B were filled with Furfurol(F), and the lid was secured by welding. Figure 10 shows a schematic cross-section of Configuration A within Container B. Once sealed, Container B was then moved to a temporary land storage facility constructed of concrete blocks. After about one and one half years, Container B was removed from the temporary storage facility and placed in a specially prepared caisson, hereafter known as Container C, aboard a 6.5 m diameter by 12.5 m long steel pontoon. The walls and lid of Container C were constructed of type 8X18H10T SS of 12- and 18-mm thickness, respectively. The voids between Container B and the interior confines of Container C were filled with Furfurol(F), and the lid was secured by welding.

Like the icebreaker, the pontoon was towed from Murmansk to Tsivolka Fjord for disposal. During transit, a storm occurred in the region of the Kara Gate and the pontoon was temporarily lost due to rupture of the towing cable. The pontoon was subsequently found, secured to the towing vessel *Lepse*, and towed to Tsivolka Fjord. On September 18, 1967, the pontoon was dumped within one kilometer of the site that would be used for the RC.

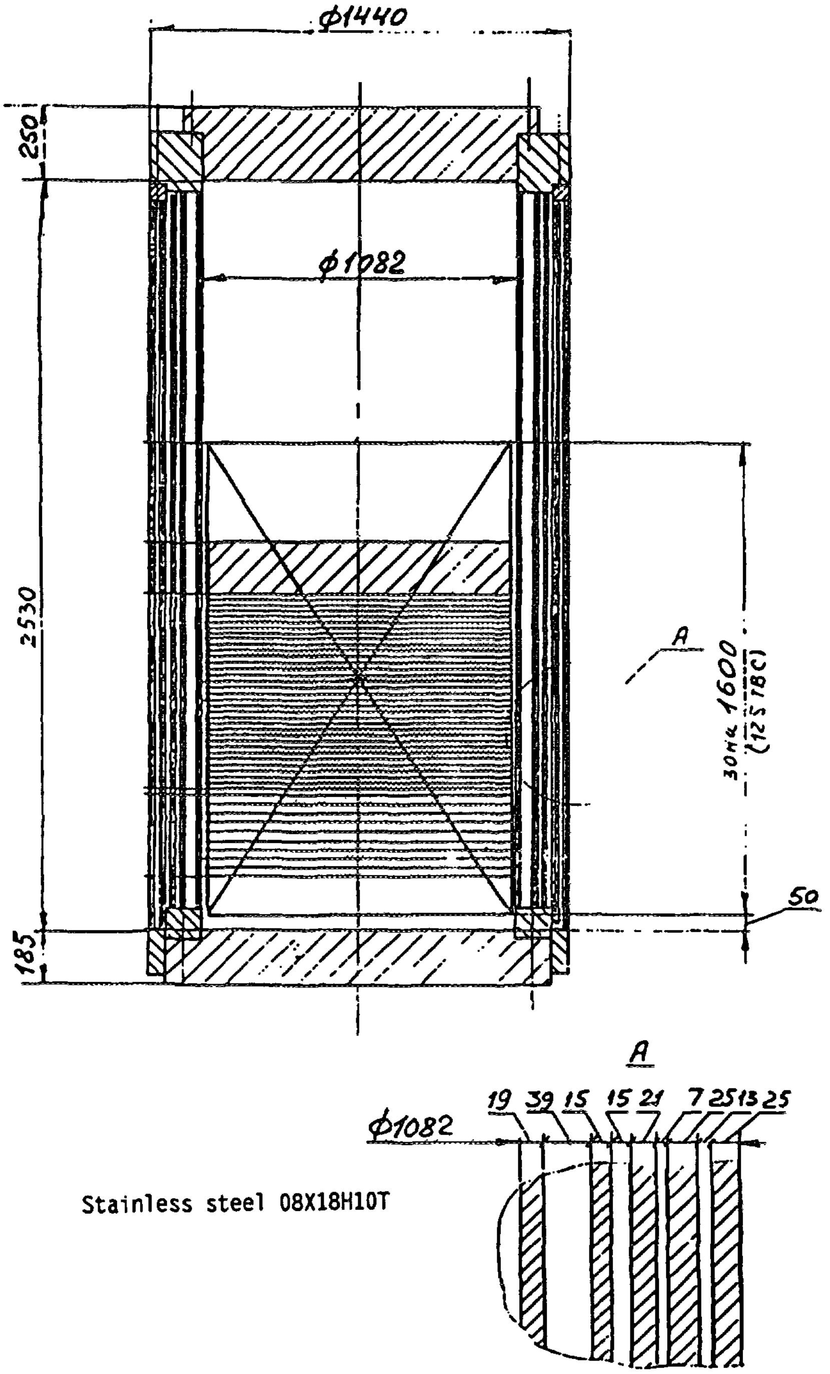

FIG. 9. Schematic cross-section of the technical fuel channels and core barrel from the icebreaker N2 pressurized water reactor [4].

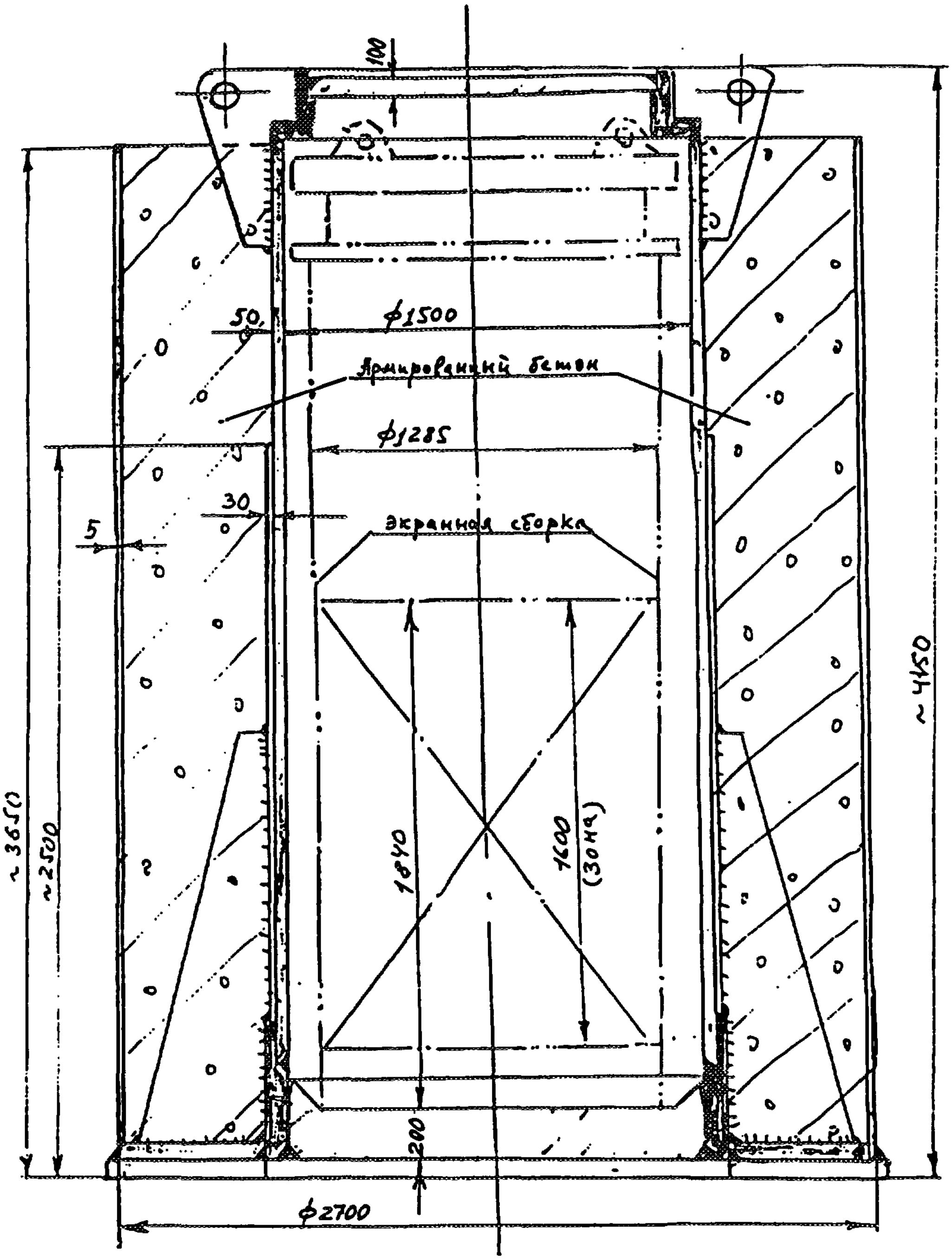

FIG. 10. Schematic cross-section of the technical fuel channels and core barrel from the icebreaker N2 pressurized water reactor within Container B [4].

3. MODELLING STRATEGY

In order for the IASAP to provide an assessment of the radiological impact of the dumped marine reactors, source terms were required for the following release scenarios:

(1) Scenario A: the "best estimate" discharge scenario.
(2) Scenario B: the "plausible worst case" scenario is a situation where a disruption, e.g., collision or munitions explosion, causes a complete breach of the containment including the SNF from the icebreaker. This will be assumed to occur at the year 2050.
(3) Scenario C: the "climate change" scenario refers to a major environmental disruption where global cooling followed by glaciation scours out the fjords. Subsequent warming would then release activity directly into the Kara Sea from the disrupted/crushed reactor cores. This release will be assumed to occur at 3000 AD, one thousand years from now.

The information presented in Sections 3.1 (Methods and Assumptions), 3.2 (Model Construction), and 3.3 (Corrosion Rates) are particularly germane to the development of the release Scenario A.

3.1. METHODS AND ASSUMPTIONS

3.1.1. Initial assumptions

At the outset it was realized by the Group that a detailed and accurate prediction of radionuclide release rates was impossible, taking into account the small amount of information available and its accuracy and reliability. There were many factors that would be difficult, if not impossible, to model with any degree of confidence. It was assumed that the release of radionuclides from the dumped objects would be driven by sea water corrosion of the active material in the objects, but any further analysis quickly raised problems:

How does the water enter the containment barriers?
At what rate does corrosion occur?
How does biofouling affect the corrosion rates?
Is there enough oxygen inside the containment to support corrosion?
Is there any flow through the containment barriers?
How is corroded material removed from the containment?
What fraction of material escapes from the containment and at what rate?
How do the filler materials (Furfurol(F), concrete, bitumen) behave in sea water? How does their effectiveness degrade with time, following immersion in the sea?
Does the build up of corroded material hamper further corrosion and removal of material?
To what extent is the core damaged and how does it affect the integrity of the containment?

In view of the aims of the IASAP (to "assess the risks to human health..." and "examine possible remedial actions....") it was decided to apply stated assumptions in order to obtain data that could be considered to be the worst case, i.e., fastest rates of release.

The assumptions used were:

(1) All material of a particular type (e.g., mild steel, SS, Pb-Bi) corroded at a fixed base rate, modified by a correction factor. The choice of corrosion rate values is discussed in Section 3.3 (Corrosion Rates).
(2) The correction factors used to modify the best corrosion rates (BCRs), known as k factors, were dependent on the degree to which the containment barriers had been breached. They crudely model the slowing of the corrosion rate as the oxygen is used inside a volume with little contact to the open sea and the restriction in water flow through the reactor.

(3) All material was assumed to be released to the environment as soon as it was corroded. This avoids modelling the removal processes from corrosion to the open sea, and immediately provides the worst case.

(4) The filler materials were assigned a lifetime instead of a corrosion rate. At the time of dumping, the filler was assumed to be a perfect barrier to sea water; the filler material then degrades at a constant rate until, at the end of the assumed lifetime, the filler ceased to provide any kind of barrier to water ingress or radionuclide release.

(5) The presence of fuel pin cladding was ignored as the extent of fuel pin damage was unknown.

The validity and reliability of these assumptions is discussed in more detail in Section 4.1 (Reliability)

3.1.2. Routes for water ingress

The next stage in the modelling process was to ascertain the most probable routes for water ingress to the active material in the dumped objects. This involved detailed discussion with the Russian members of the Group and examination of the structural information provided by the Russian Federation. Because of the high activities involved, initial studies were concentrated on those objects containing SNF. In all cases, particular points of weakness such as control rod channels were identified which would lead to water ingress to the SNF. The detailed strategies used for the submarine PWRs, submarine LMRs, and the icebreaker PWRs are covered in the individual Sections 3.2.1 (Submarine pressurized water reactors), 3.2.2 (Submarine liquid metal reactors), and 3.2.3 (Icebreaker pressurized water reactors), respectively.

3.1.3. Modelling methods

Once the route and timing of water ingress were identified, the process of modelling corrosion and release of active material could begin. The time at which water first made contact with each material was taken to be the start time for release. The BCR, modified by the relevant k factor, was then used to determine the rate at which material corroded and hence released to the environment. The model hence depended on calculation of the k factors and the rate at which material was released by corrosion. The following three sections illustrate how the effective corrosion rates were obtained to allow release rate calculations.

3.1.3.1. Calculation of k factors

In most cases, the active material contained in the SGI was enclosed by an outer containment barrier such as the submarine hull or RPV. Although breached and allowing ingress of sea water, these containment barriers were assumed to provide a restriction in the oxygen transport into the reactor interior, slowing the true corrosion rate. This effect was modelled by introducing a factor k_c, where:

$$k_c = \frac{\text{area of breach in containment}}{\text{total surface area of containment}} \qquad (6)$$

In practice, the containment factor k_c consists of the k factors derived for various RPV penetrations, e.g., breather holes, CCR tubes, primary coolant tubes, etc. The value of k_c varies through the corrosion process; for example, most of the PWRs were dumped in a separated RC, which was sealed with steel end plates. When these corrode and allow ingress, a step increase in the value of k_c models the increase in size of the breach. The factor k_c was scaled to give a value of unity once both end plates have fully corroded. Where the corroding material is surrounded by more than one barrier, the k factors are multiplied together to give a cumulative factor.

Furfurol(F) has the effect of inhibiting the activity released from the SNF encased in it by the factor k_f. The multiplying factor k_f is zero until the material is dumped, and then ramps to unity over the ensuing 100 years:

$$k_f = \begin{cases} 0.01t & (t \leq 100 \text{ years}) \\ 1 & (t > 100 \text{ years}) \end{cases} \tag{7}$$

where

t is the time from commencement of corrosion (a).

Thus the effective corrosion rate for a material inside a Furfurol(F) filled RPV is:

$$X_{eff} = k_c k_f X \tag{8}$$

where

X_{eff} is the effective corrosion rate ($mm \cdot a^{-1}$),
k_c is the RPV containment k factor,
k_f is the Furfurol(F) k factor,
X is the BCR ($mm \cdot a^{-1}$).

Concrete was used to encase the RPV from submarine factory number 421. Again, with little information on the type of concrete or its behavior in sea water, a factor $k_{concrete}$, with values identical to that of k_f, was introduced to represent the effect of the concrete barrier. The RC of the dumped submarine factory number 601 was partly filled with bitumen prior to dumping, and the model for release from this unit used a factor k_b, again with a value identical to k_f.

3.1.3.2. Analysis of fuel pin geometry

Once sea water has entered the core region of RPVs with SNF, corrosion of the cylindrical fuel pins will occur, allowing release of fission products and actinides. The release rate will depend on the size of the fuel pin: the volume of fuel alloy released per year will be given by

$$v(t) = 2\pi h X_{eff}\left(R - X_{eff}t\right) \tag{9}$$

where

$v(t)$ is the volume of SNF released per year ($m^3 \cdot a^{-1}$),
h is the height of the fuel pin (m),
X_{eff} is the effective corrosion rate, as defined in Eq. (8),
R is the initial fuel pin radius (m)
t is the time from commencement of corrosion (a).

The activity of fission products and actinides released per year can be obtained from:

$$a(t) = \frac{A(t)v(t)}{V} \tag{10}$$

where

$a(t)$ is the activity released per year ($Bq \cdot a^{-1}$)
$A(t)$ is the total activity of all the material, corroded and uncorroded, at time t (Bq)
V is the total initial volume of the material (m^3)

Expressing V in terms of R and h, and using Eq. (9), the activity release of fission products and actinides can be shown to be:

$$a(t) = A(t)\frac{2X_{eff}}{R^2}\left(R - X_{eff}t\right) \qquad (11)$$

where all terms are previously defined.

3.1.3.3. Analysis of activation product release

The release rates of activated material from SGIs were calculated by applying effective corrosion rates to simplified geometries representing the structure in question. For the submarine PWRs and icebreaker, the majority of the activation products came from the thermal shields and RPVs; these were modelled using plane geometry for simplicity.

In the case of the LMRs, the bulk and complex geometry of the thermal shields and ingress routes required a more detailed analysis, and corrosion rates were modelled using a variety of circular corrosion geometries applied to the core, thermal shields and SGs.

3.1.4. Calculation of activity release rates

Activity release rates were calculated using a FORTRAN code developed at the Royal Naval College Greenwich, running on a 486DX PC. For each component of the SGI, the actual corrosion rate (and hence activity release rate) was calculated using derived effective corrosion rates.

The decay from the original inventory for each of the isotopes used was calculated for timesteps into the future. Only the Pu-Am decay scheme was considered for daughter product accumulation.

The remaining thickness of each component at the timesteps was calculated using the effective corrosion rates, allowing for the fact that these rates will change as levels of containment disappear. For example, the protection provided by the RPV greatly decreases the corrosion rates of the contents. The SS thermal shields for instance will generally outlast the RPVs. When a component has corroded through, it was removed from the calculation and adjustments made in effective corrosion rates from that time onwards.

The effective activity release rates in each year were then calculated. The thickness of material which would corrode away in that year, as a fraction of the original thickness, multiplied by the activity still remaining, gives the release rate to the ocean.

It was assumed that the distribution of activity in activated components was linear. For the thicker steel components, the RPV wall for instance, the distribution will not be linear but information was not available and again, the assumption tends to the pessimistic in the early years.

Since model timescales were of the order of hundreds or thousands of years, any radionuclide in the core inventories with a half-life of a year or less was ignored. Any release rate less than 1 $Bq \cdot a^{-1}$ was also eliminated from the output.

The data was written to ASCII files for graphing using Microsoft EXCEL, and for distribution to the IASAP Modelling Group. Files for the yearly release rates of fission products, activation products and the actinides, and totals for each group were prepared.

3.2. MODEL CONSTRUCTION

For all SGIs, corrosion of the outer surface of the RPV and reactor components begins immediately after dumping, as the RCs are free to flood, albeit initially at a slow rate owing to the limited access to the open sea water via hull penetrations. The first release is therefore almost immediate, with activation products released at the rate of corrosion of the carbon steel of the RPV; this release is dominated by ^{60}Co and so rapidly decays over the first twenty five years after dumping.

After that, the activity release pattern is primarily dependent on the way in which water enters the RPV containment and corrodes the SNF and active material inside.

3.2.1. Submarine pressurized water reactors

For the submarine PWRs, the fastest ingress routes into the core are:

(1) via the two breather holes, as described in Section 2.5.1, into the walls of certain CCR guide tubes. For the typical RPV, there were 10 mm diameter holes used to fill the RPV with Furfurol(F), capped with a 2.5 mm thick weld. Pitting corrosion attacking these joints will offer the first access to the interior of the reactor and the SNF. These tubes will open at about 50 years from time of dumping.
(2) via the primary circuit inlet and outlet tubes, which had been cropped and welded shut with a 10 mm thick steel plate cap; and
(3) ingress into the control rod channels, via the 10 mm thick steel caps. Pitting corrosion would enable ingress through both these routes 60 years after time of dumping. Much later, the RPV itself will corrode away, leaving behind the SNF, the SS cladding and the thermal shields. By then however, there is little left of the SNF and the major component of the release comes from the activated shields.

For the U-Al alloy fuel assemblages, a faster rate of release was given to some of the more soluble and mobile radionuclides of the fission product inventory, typically 20% of the total activity.

3.2.2. Submarine liquid metal reactors

The first barrier to sea water ingress is the bitumen which covers the whole SGI. This bitumen is assumed to degrade linearly in 100 years from full to zero effectiveness, modelled by the bitumen factor $k_b = 0.01t$ ($t \leq 100$ years). The fuel, and hence the fission products and actinides, is surrounded above and below by at least 300 mm of solidified Pb-Bi coolant, and multiple layers of SS. Using the most pessimistic estimates of corrosion rates and thicknesses, the minimum time for water ingress to the fuel via corrosion directly from above or below is over 40 000 years. This timescale is much longer than that of ingress through the EPR tubes and ECTs; hence, this method will be assumed to be the primary means of release and is studied in more detail below.

Barrier breakdown of the LMRs was considered in three main areas.

(1) *The steam generators*: the port SG contains the remaining 20% of the fuel from the left board reactor (i.e., 10% of the total inventory of the submarine SGI) in the form of damaged fuel pins, and both SGs contain activation products in the Pb-Bi coolant.
(2) *The thermal shields*: the majority of the activation products are contained in the thermal shields around each core. These will corrode via the ECTs and from the outside.
(3) *The reactor cores*: ninety per cent of the total fission product and actinide inventory is contained in the two cores, which will corrode initially from the EPR channels, and later from the outside of the cores as the thermal shields corrode away. Activation products will also be released at the same rate from the Pb-Bi coolant and structural material in each core.

3.2.2.1. Corrosion of coolant and damaged fuel in the left board steam generator

The left board SG contains 10% of the total fission product and actinide inventory. Once the bitumen and Furfurol(F) have degraded, sea water can attack the exterior of the SG and then reach the damaged fuel pins inside, allowing release of fission products and actinides. The sea water will also attack the Pb-Bi coolant in the SG, releasing activation products. A diagram of the SG, showing the weld sites and SNF, is shown in Figure 11.

The SG is covered by bitumen and the secondary side filled with Furfurol(F). The primary side of the SG is filled with Pb-Bi coolant. The Furfurol(F) and bitumen both degrade in 100 years, allowing pitting corrosion of the welds at the secondary/primary header connection to commence. The fastest ingress route into the region of the primary header with SNF was assumed to be through weld A, then the Furfurol(F) in the secondary header, and through the primary header tube walls below the header plate. The total time for water to reach the SNF by this method is 185 years. If the maximum IASAP corrosion rates (i.e., BCR + 50%) are used, the time drops to about 115 years.

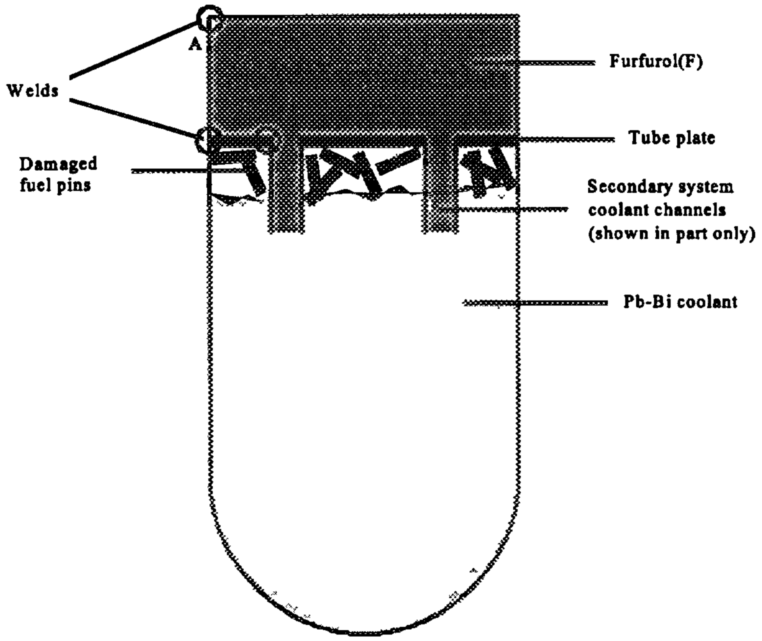

FIG. 11. Cross-section through the left board steam generator of the liquid metal reactors of submarine factory number 601, showing the damaged fuel pins, Pb-Bi coolant in the primary circuit, and Furfurol(F) in the secondary header and channels.

The fuel pins are assumed to be lying on top of the denser coolant, and their whole surface area is exposed and available for sea water corrosion. Thus, the fuel pin corrosion rate can be applied to calculate the activity release rate from the SNF in the SG, which will fall to zero 500 years after first contact with water, i.e., at D + 685 years.

The activation products in the coolant will also begin release as the water breaches the tubes. In this case, the release rate will rise linearly as the surface area of corroding coolant increases with distance from the tube. All coolant will corrode by D + 1185 years.

3.2.2.2. Corrosion of thermal shields

The majority of the activation products are contained in the thermal shields around the core. These shields are penetrated by the 24 ECTs which are filled with Furfurol(F) and will become open to sea water as the Furfurol(F) degrades. The ECTs provide the primary site for corrosion of the shields. Corrosion of the shields also occurs radially inward once the bitumen and Furfurol(F) barriers have degraded and allowed sea water on to the RPV exterior and into the "steam envelope." This 30 mm space between two external SS layers, filled with Furfurol (F) at time of dumping, is assumed to be prone to water ingress as the Furfurol(F) must have been injected via a suitable opening; once the Furfurol(F) degrades, this would allow water to penetrate.

Water ingress to the ECTs occurs once the elliptical cap and bitumen fail to be effective barriers, which can be assumed to be at D + 100 years. Corrosion of the SS and Pb-Bi layers can then commence; the corrosion rate of the layers can be treated as that of a homogeneous mix of SS:Pb-Bi in the ratio 36:3, since most of the SS layers are 36 mm wide and all the Pb-Bi channels are 3 mm wide (see Figure 8). Assuming BCRs for SS and Pb-Bi of 0.02 mm$\cdot$a^{-1} and 0.001 mm$\cdot$a^{-1}, respectively, the mean rate for the homogeneous mix can thus be taken as 0.019 mm$\cdot$a^{-1}. Corrosion will then expand from each of the 24 ECTs as shown in Figure 12. Corrosion will continue to release activation products as the circles expand, until they meet after 3100 years (D + 3200 years).

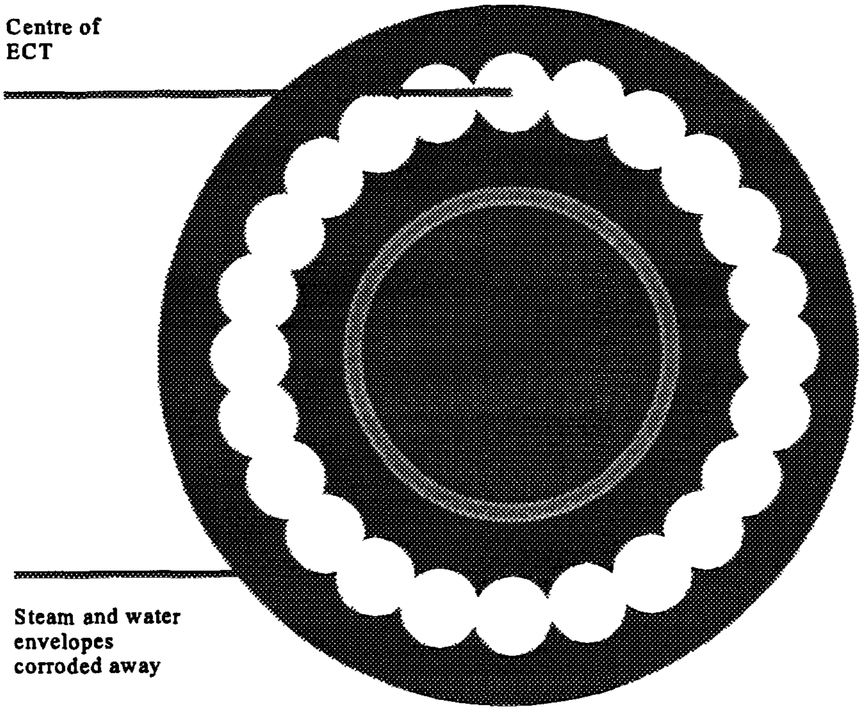

FIG. 12. Formation of an annulus from the emergency cooling tubes in the liquid metal reactors of submarine factory number 601 at about 3200 years after dumping.

The release rate will now become complex as the circles merge; an approximation can be made by simplifying the geometry to form an annulus as shown in Figure 13. Corrosion and activation product release can then continue inward and outward at 0.019 mm$\cdot$a^{-1} until the BeO reflector or the corrosion attacking inward from the edge of the structure are reached.

During the formation and expansion of the annulus, corrosion will also be occurring at the outside edge of the thermal shields. The outer edge of the RPV will be corroded by sea water penetrating into the "steam envelope." This space has an inner edge at 900.5 mm from the core center, and corrosion will begin at that edge at approximately D + 100 years, once the bitumen and Furfurol(F) have degraded.

The corrosion from the steam envelope will meet the corrosion from the ECTs after 1600 years, so the outer thermal shields will have corroded away by D + 4800 years, as shown in Figure 14.

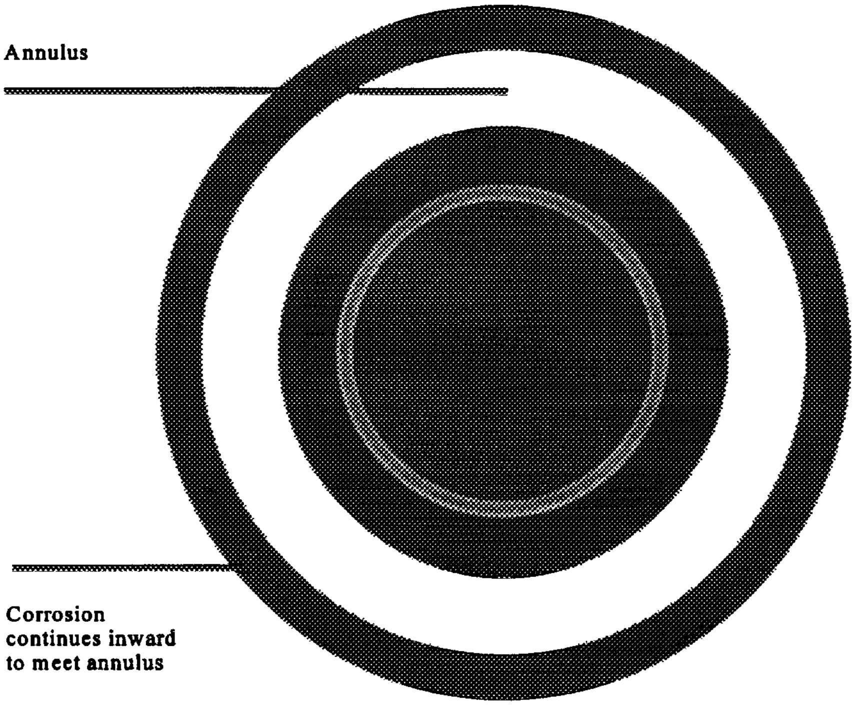

FIG. 13. Formation of an annulus from the emergency cooling tubes and corrosion from the outside of the thermal shields in the liquid metal reactors of submarine factory number 601.

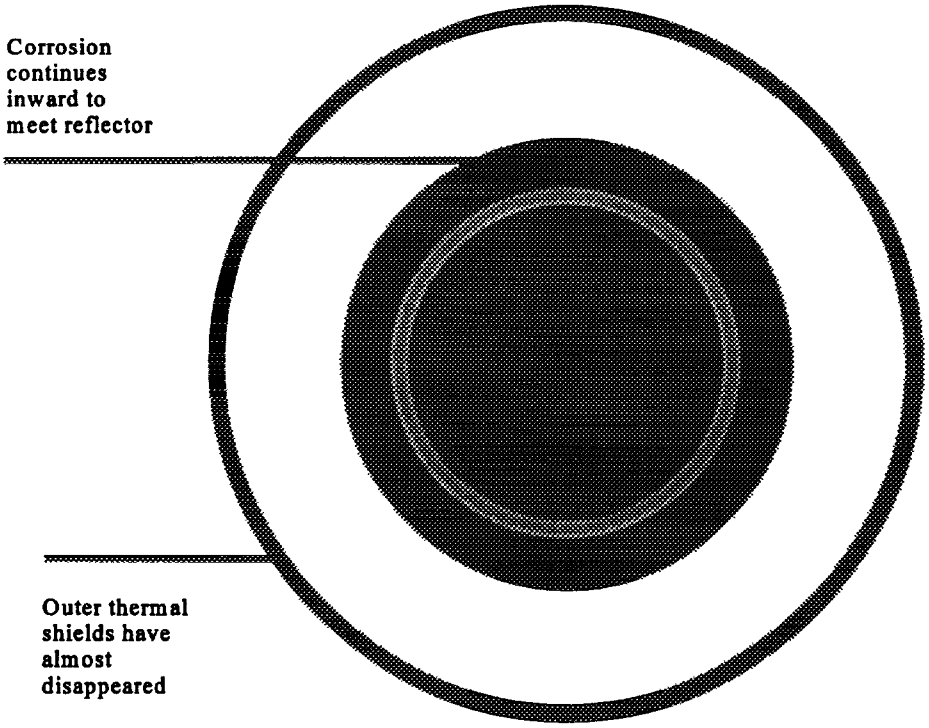

FIG. 14. Loss of the outer thermal shields and continuing corrosion of the inner shields of the liquid metal reactors of submarine factory number 601 at about 4800 years after dumping.

The inner thermal shields will continue to corrode until the reflector is reached at D + 9380 years.

The BeO reflector is assumed to corrode at 0.01 mm·a^{-1} and so will corrode away after 6500 years (D + 15 880 years). Inside the reflector is a 10 mm thick SS layer, which will corrode away in 500 years (D + 16 380 years). The water will then reach the outside of the core.

The fractional release rate of material from the thermal shields is shown in Figure 15.

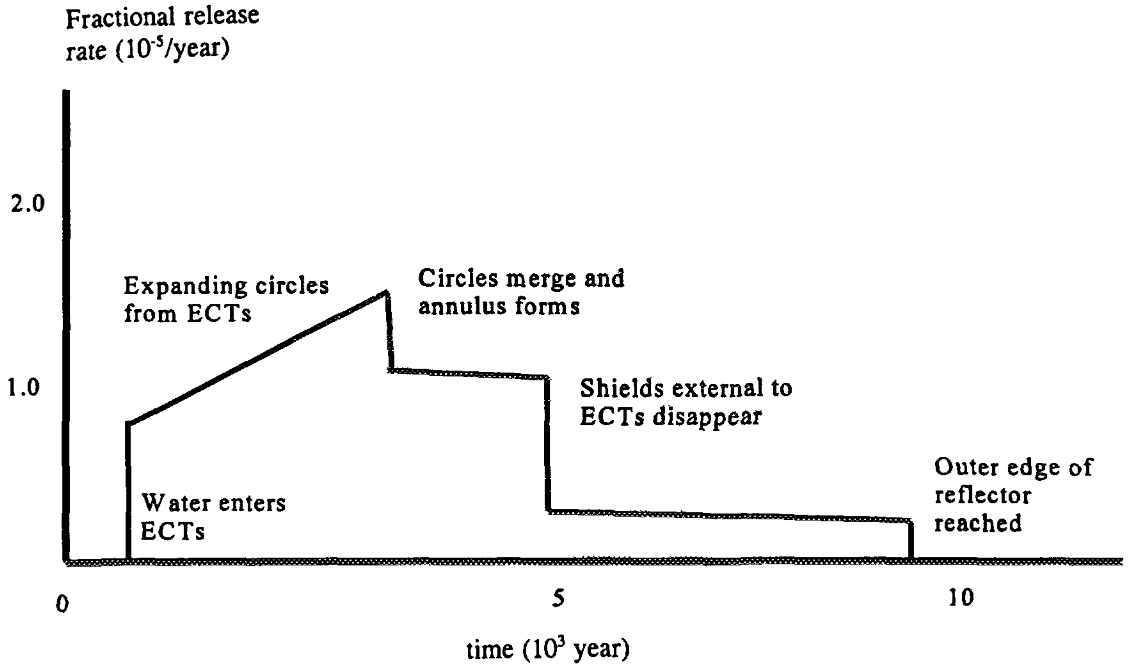

FIG. 15. Fractional release rate versus time for activation products from the corrosion of thermal shields via the emergency cooling tubes (ECTs) and steam envelope of the liquid metal reactors of submarine factory number 601.

3.2.2.3. Corrosion of core

The elliptical shield that covers the RPV lid is 20 mm thick; pitting corrosion of the SS at 0.4 mm·a^{-1} will be modified by the bitumen factor $k_b = 0.01t$ for the first 100 years, giving a mean corrosion rate of 0.2 mm·a^{-1} and so penetration of the welds of the elliptical shield will occur after 100 years. The Furfurol(F) in the CCR and EPR tubes will have ceased to be a useful barrier by this stage, so water will be able to enter the tubes and attack the tube walls. The CCR tubes are filled with solidified Pb-Bi up to approximately 500 mm above the top of the core (see Figure 6), with the remainder of the tube filled with Furfurol(F). The EPR tubes are completely filled with Furfurol(F), hence, once the Furfurol(F) has degraded, water will penetrate to the bottom of the EPR tubes and can attack the tube walls next to the core. The walls are 0.5 mm thick, allowing penetration to occur after 25 years. Hence, 125 years after dumping (D + 125 years), the water has reached the core via the EPR tubes.

Once water has reached the core via the EPR tubes, the corrosion will spread out concentrically from each of the three ingress sites into the Pb-Bi, as shown in Figure 16. The nearest fuel pins are about 6 mm distance from the rod channels; corrosion will progress at the Pb-Bi corrosion rate until the fuel is reached, so activity release from the core for the first 600 years (D + 725 years) is solely activation products from the Pb-Bi. After 600 years, the expanding circles of corrosion will reach the first fuel pins and fission products and actinides will begin to be released. The solidified Pb-Bi is assumed to have a corrosion rate of 0.01 $\mathrm{mm\cdot a^{-1}}$; the U/Be/BeO fuel is assumed to have a corrosion rate of 0.001 $\mathrm{mm\cdot a^{-1}}$; and the steel structural material a rate of 0.02 $\mathrm{mm\cdot a^{-1}}$. Treating the fueled area as a homogeneous mixture of 54% fuel, 10% SS and 36% coolant, the mean corrosion rate is

$$(0.01 \times 0.36 + 0.001 \times 0.54 + 0.02 \times 0.1) = 0.006\ \mathrm{mm\cdot a^{-1}}$$

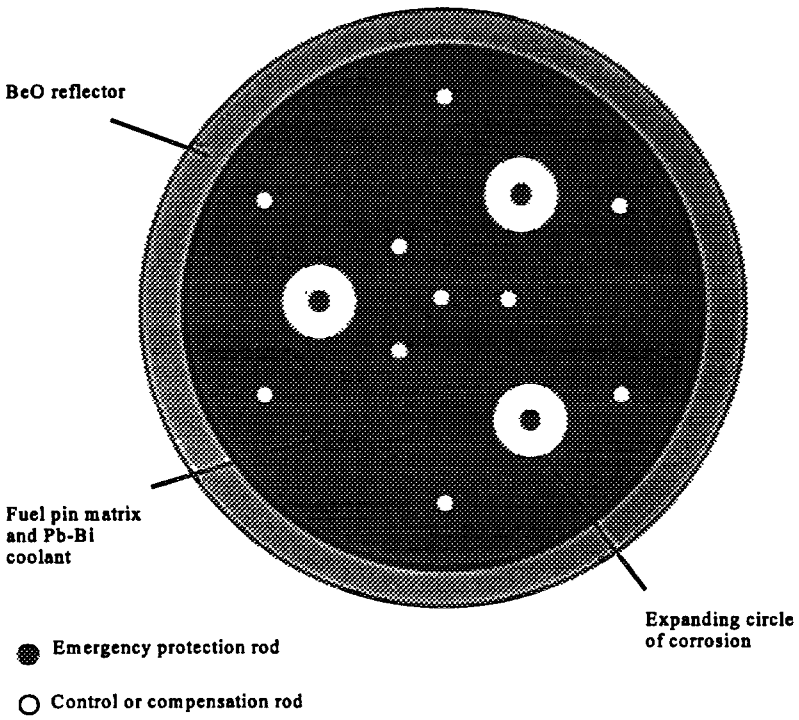

FIG. 16. Circles of corrosion expanding from the emergency protection rod channels of the liquid metal reactors of submarine factory number 601.

The corrosion through the thermal shields will reach the edge of the core by D + 16 380 years, by which time the expanding circles from the EPR channels will have corroded out to 98 mm from the EPRs; at this stage the situation will be as shown in Figure 17. At this point, the corrosion from the outside begins to release fission products and actinides. The corrosion from the outside will meet up with the EPR corrosion after 7670 years, i.e., at D + 24 000 years. The geometry then becomes complex as the expanding circles and the outside corrosion merge: the arrangement is shown in Figure 18. For simplicity, it is assumed that the material release rate then remains constant until all core material has corroded away, so the core will corrode away in 10 600 years, i.e., at D + 34 600 years. The release rate of material from the core is shown in Figure 19.

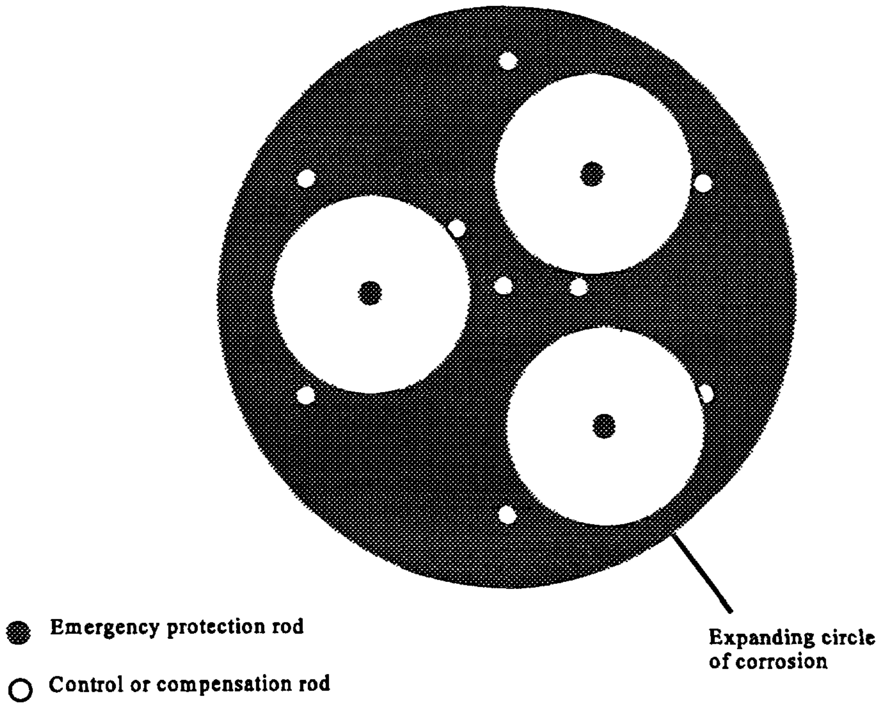

FIG. 17. Circles of corrosion from the emergency protection rod channels of the liquid metal reactors of submarine factory number 601 at about 16 400 years after dumping. The stainless steel thermal shields and the BeO reflector outside the core have corroded away, allowing water to attack the outside of the core.

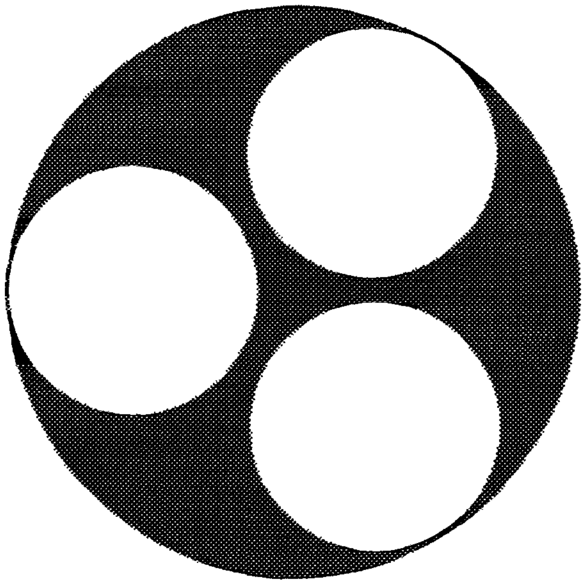

FIG. 18. Circles of corrosion centered on the emergency protection rod channels merging with corrosion from the core exterior of the liquid metal reactors of submarine factory number 601 at about 24 000 years after dumping.

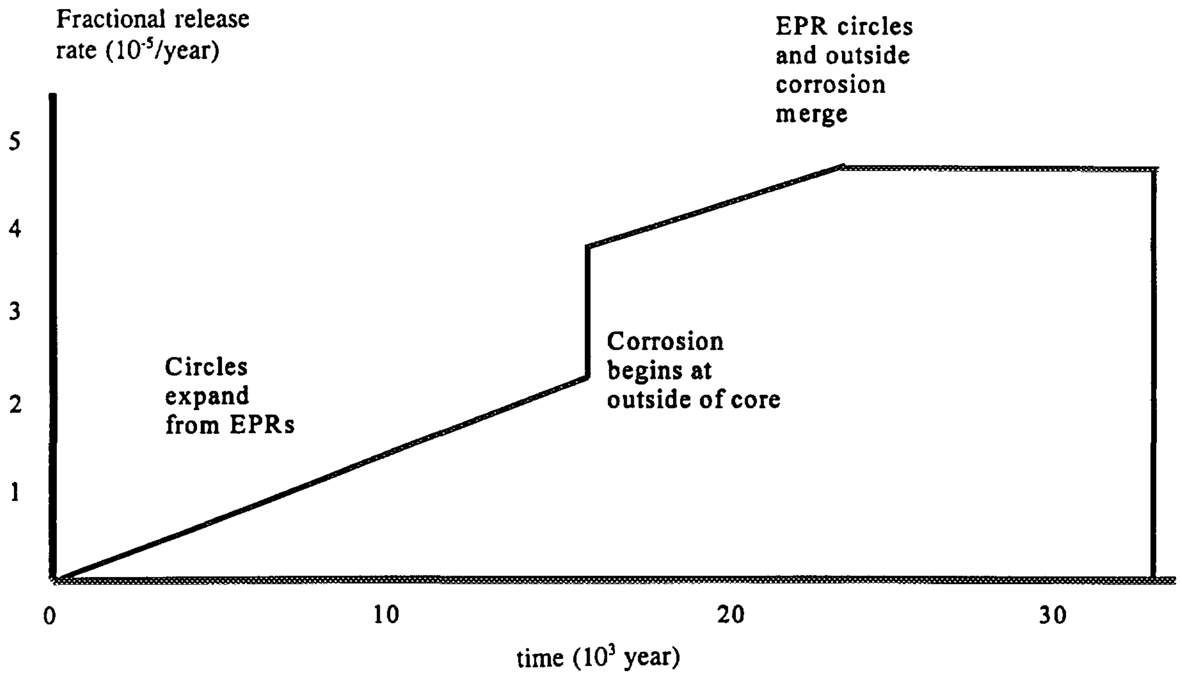

FIG. 19. Fractional release rate of fission products and actinides from corrosion via the emergency protection rod (EPR) channels of the cores of the liquid metal reactors of submarine factory number 601.

3.2.2.4. Corrosion of control rods

As indicated earlier, the EPR and CCR are made of EuB_6, a material with ceramic properties. Little information is available about corrosion rates of EuB_6 in water. It is assumed here that the rate is similar to that of other ceramics such as BeO, and a rate of 0.01 $mm \cdot a^{-1}$ is assumed. Water enters the channels at D + 125 years; hence, corrosion of the rods will commence at that date. The EPR and CCR have radii of 10 mm and 8.5 mm, respectively, and the CCRs are surrounded by approximately 1 mm of Pb-Bi coolant within the channel. Both types of rod channel were filled with Furfurol(F) prior to dumping, but the CCRs are also surrounded by solidified Pb-Bi filling the channel to a height of approximately 500 mm above the core, which will prevent water reaching the rods from above for at least 50 000 years. Hence, only the EPRs will be attacked by sea water within the timescales of interest, beginning at D + 125 years. The EPRs have a 10 mm radius and so will corrode away after 1000 years (D + 1125 years). The resulting release rates of activated Eu will be modified by the change in size (i.e., surface area) of the rod; this mechanism will be identical to the modification to the fuel pin corrosion rate.

3.2.3. Icebreaker pressurized water reactors

Although the geometry of the structure containing the SNF and the core barrel from the N2 reactor is complicated, modelling the release rates is relatively simple [21]. It has been assumed that no activity is released from the whole unit, until the lid of Container C has corroded off, followed by the lid of Container B, using pitting corrosion rates applicable to the material of each container, acting on the lid welds. Once seawater penetrates Container B, the five SS thermal shields within the N2 PWR core barrel are assumed to corrode away on both sides. The results are discussed in more detail in Section 4.2 (Sensitivity), but here one can note that the containers open in the year 2305, approximately 340 years after dumping.

The damaged fuel from the N2 PWR is assumed to comprise UO_2 pellets, 5% enriched, 4.5 mm diameter, 10 $g \cdot cm^{-3}$ density, carried in Zr-Nb alloy cladding. As soon as the fuel is in contact with sea water, an immediate release of 20% of the fission product and actinide inventory is assumed from the fuel grain boundaries. The remaining fission products and the actinides are released through dissolution of fuel grains at a pessimistic rate of 30×10^{-7} $g \cdot cm^{-2} \cdot d^{-1}$ [22]. For fuel pins with a 4.5 mm diameter and 10 $g \cdot cm^{-3}$ density, this equates to a constant dissolution rate of 0.0011 $mm \cdot a^{-1}$. Once exposed to seawater, the fuel will take some 2250 years to corrode away.

For the RPVs dumped in the RC, the same modelling philosophy which was applied to the submarine PWRs was used (Section 3.2.1). The same breather holes, control rods and primary pipework caps were welded onto the entry points to the RPV.

The differences were:

(1) The RC was sealed with a steel plate over the upper hatch. The welds were attacked by pitting corrosion and yielded in 60 years, allowing access for RPV corrosion to begin.
(2) As these early reactors had bolted, instead of welded flanges, a gap of 1 mm was assumed between the joints of the RPV head and all primary pipework couplings. This allowed immediate, if limited entry for water into the RPV, to attack the internal components.
(3) No Furfurol(F) was put into the RPVs.

The result was that first release of activation products would be 60 years after dumping, with rates increasing 5 years later, when the RPV pipework caps fall off.

3.3. CORROSION RATES

The modelling strategy relies heavily upon corrosion rates of containment and barrier materials in the dumped SGIs. Such materials include the metals used in their original fabrication, and fillers applied prior to dumping to inhibit metallic corrosion and provide additional barriers to activity release into the sea. Strictly speaking, therefore, in this context the term *corrosion* encompasses all forms of material degradation, and it is this mechanism that liberates radionuclides through the breakdown of radioactive materials and of the barriers encasing them. This process is considered under the heading of each material type later in this section.

Factors affecting the lifetime of materials exposed to sea water include: dissolved oxygen, temperature, pH, salinity, sulphates, pressure, and marine life (biofouling). Heiser and Soo [23] provide a detailed review of the influence of these factors on a range of materials, with data derived from analysis of dump sites in the Atlantic and Pacific. However, little data is available on the behavior of materials in relatively shallow Arctic waters. To derive corrosion rates for the IASAP, comprehensive reviews of available source references were made in support of this work, with notable contributions from Carter [22] and Heiser and Soo [23].

In all cases, realistic but conservative values of material corrosion rates were determined; their values are summarized in Table XVII. The rates lie towards the high end of recorded data in order to err towards the pessimistic side of the spectrum. For modelling purposes, corrosion rates established in this way are designated BCRs and filler lifetimes. Where there is a paucity of data, e.g., in the case of the Pb-Bi eutectic, nuclear fuels, and certain filler materials, best use has been made of available information and judgement. It should be noted, however, that containment k factors, as described in Section 3.2 (Model Construction), effectively inhibit BCRs by restricting pathways through barrier materials.

To account for the uncertainties in corrosion rates employed in the models, values generally 50% either side of the BCRs are applied in the sensitivity analyses described in Section 4 (Results and Analyses), which also has an assessment of the reliability of the data used. The ranges in the parameters studied in the sensitivity analyses are also included in Table XVII.

3.3.1. Metals

3.3.1.1. Steels

Not surprisingly, much greater data is available on the performance of steels in sea water than any other material. The range of values, however, is wide, and related to the factors cited earlier. Applying the BCR principle detailed above, realistic but conservative corrosion rates were determined for long term immersion in sea water under two distinct mechanisms: bulk corrosion and pitting corrosion, with the latter applied under conditions of high stress in the heat affected zones of welded areas. As can be seen from Table XVII, for simplicity two types of steel were assumed: SS for specific SGI components and the icebreaker SNF containment (Containers B and C), and mild steel (low alloy and or carbon steels) for RPVs and submarine structure. To account for the possible accelerating effect of biofouling, corrosion rates were doubled for steels with outside surfaces fully exposed to the sea.

3.3.1.2. Europium hexaboride

This material is used in the CCRs and EPRs of the LMRs installed in submarine factory number 601. The Group could find no published data on the corrosion behavior of this material, and assumptions had to be made for modelling purposes. Although pure Eu is a reactive metal, in ceramic B_6 form it is expected to be stable and relatively impervious to corrosive attack. The BCR of 0.01 $mm{\cdot}a^{-1}$ was selected on the assumption that it was unlikely to be any higher than this. The importance of this parameter in the LMR model relates to the criticality issue; and its value is shown in Section 5.2 (Reactor Criticality) to be insensitive in the criticality analysis, and therefore not of significance with regard to accuracy.

3.3.1.3. Lead-bismuth eutectic

The Pb-Bi eutectic is used as a coolant in the LMRs installed in submarine factory number 601. In the absence of information on the performance of this material in sea water, the model assumes the same corrosion rate as Pb. Heiser and Soo [23] report uniform corrosion rates of 0.008 to 0.013 $mm{\cdot}a^{-1}$, and a BCR for Pb of 0.01 $mm{\cdot}a^{-1}$ is therefore not unreasonable. There is clearly some uncertainty over the performance in sea water of the Pb-Bi eutectic compared with pure Pb. However, until further data becomes available on the former, it is considered a representative rate to use in the model at this stage.

3.3.2. Reactor fuels

3.3.2.1. Uranium-aluminum pressurized water reactor fuel

The sole source of information on the corrosion of $U\text{-}Al_3$ alloy fuel used in PWR SGIs is that of Yefimov, et al. [24], which reports the results obtained from immersion in laboratory sea water of spent fuel from the icebreaker *Siberia.* In view of the high corrosion rate of 0.3 $mm{\cdot}a^{-1}$ established for the sample fuel, this value is used in the model for the soluble fission products fraction only. A rate an order or magnitude lower, 0.03 $mm{\cdot}a^{-1}$, is applied to the insoluble fraction. Due to the unknown extent of damage to fuel cladding, its effect as a barrier is ignored in the model.

3.3.2.2. Uranium-dioxide icebreaker pressurized water reactor fuel

Not much is known of the behavior of UO_2 fuel pellets in sea water. A review of available data by Carter [22] enabled an estimated grain dissolution rate of 30×10^{-7} $g{\cdot}cm^{-2}{\cdot}d^{-1}$ to be determined for UO_2 fuel pellets. For fuel pins of 4.5 mm diameter, of regular geometry and 10 $g{\cdot}cm^{-3}$ density, this equates to a dissolution rate of 0.0011 $mm{\cdot}a^{-1}$. The same reference summarizes work by Shoesmith and Sunder [25] in which the release of activity from oxide fuel in a waste repository vault is considered. This suggests:

(1) rapid release of volatiles from the fuel and cladding gap,
(2) leaching of fission products from fuel grain boundaries, and
(3) slow release of radionuclides from the fuel matrix as a result of fuel grain dissolution.

However, because of the difficulties and lack of information in estimating the rates of radionuclide release from grain boundaries, the model assumes an instantaneous release from the fuel surface and grain boundaries once sea water comes into contact with the fuel. As a conservative estimate, this is assumed to account for 20% of the total fission product and actinide inventory in the fuel. Similarly, since the extent of damage to the Zr-Nb alloy cladding around the fuel is unknown, the model assumes that cladding is ineffective as a barrier to sea water and radionuclide release.

3.3.2.3. Uranium-beryllium liquid metal reactor fuel

This fuel is assumed to be in a ceramic form, manufactured from U-Be alloy sintered into BeO, and therefore relatively impervious to corrosive attack. Again, in the absence of data for the behavior of this material in sea water, the Group consensus was to assume a similar order of dissolution rate to that used for UO_2 fuel grains. Accordingly, a rate of 0.001 $mm \cdot a^{-1}$ was modelled. As for other reactor fuels, the barrier afforded by fuel cladding is ignored because of the unknown extent of damage prior to disposal.

3.3.3. Fillers

3.3.3.1. Organic materials

Little information is available about the long term behavior of the filler substances when immersed in sea water and under irradiation. Bitumen is known to become brittle and crack below room temperature. As Furfurol(F) is a patented material, information about its composition and behavior was not available to the Group, and long term behavior is, in any case, difficult to predict. It is known to be a mixture of the following constituents: epoxy resin, amine type solidifier, mineral filler, shale distillate and furfuryl alcohol [26]. An effective lifetime of 500 years is quoted in the White Book [1] for this material.

In the absence of reliable data on the performance of Furfurol(F) in such environmental conditions, a conservative lifetime of 100 years in the radiation environment is assumed in the model and this was supported by a preliminary evaluation [27]. Hence, for the calculation of the source term, it was assumed that at the time of dumping, the fillers were fully effective as barriers to water ingress and radionuclide release at the time of disposal, but quickly began to degrade through shrinkage, embrittlement and cracking and become ineffective after the 100 year lifetime.

3.3.3.2. Concrete

Concrete was used to encase the RPV from submarine factory number 421. It is known that concrete is, in almost all conditions, porous to water. However, with little information on the type of concrete or its behavior in sea water, a similar effective lifetime to that of the other fillers is assumed in the models.

TABLE XVII. CORROSION RATES AND FILLER LIFETIMES USED FOR IASAP STEAM GENERATION INSTALLATION MODELS.

Material		**Best corrosion rates (mm a^{-1})**	**Sensitivity ranges (mm a^{-1})[6]**
Stainless steel	Bulk	0.020[1]	0.01-0.1
	Pitting	0.50	0.25-0.75
Mild steel	Bulk	0.075[1]	0.038-0.11
	Pitting	0.166	0.08-0.33
U-Al alloy		0.03[2]	0.015-0.045
UO_2		0.0011	0.0001-0.01
U-Be ceramic[3]		0.001	0.0001-0.01
Pb-Bi coolant		0.01	0.001-0.1
Boronated graphite		0.01	0.005-0.05
Europium hexaboride		0.01	0.001-0.1
Bitumen[4]		100 year lifetime[5]	50-500
Furfurol(F)[4]		100 year lifetime	25-500
Concrete[4]		100 year lifetime	50-500
Biofouling factor		2[1]	1-3

[1] For steels, bulk corrosion rates on outer surfaces were increased by a factor of 2 to account for biofouling.

[2] For fission products with low solubility and actinides; rate used for soluble fraction of fission products is 10 times the slow rate.

[3] Uranium beryllium alloy in beryllium oxide ceramic matrix.

[4] *Filler materials were given a lifetime in preference to a corrosion rate.*

[5] Lifetime is the period after which the filler no longer provides a physical barrier.

[6] The ranges are not necessarily those given in references but chosen for sensitivity analyses purposes.

3.4. RELEASE SCENARIOS

Throughout this section, calculated events are rounded to the five year date.

3.4.1. Submarine pressurized water reactor releases

The overall Scenario A and C release rates for all dumped submarine PWR SGIs are shown in Figures 20 to 25. For example, Figure 20 shows the release of fission products, actinides and activation products, together with the sum total release rate, from unit 421 from the time of dumping in 1972 until the year 3710 when the last of the steel, with its associated activation products, has corroded away.

Unit 421 was encased in concrete and as discussed earlier, a slowly degrading lifetime of 100 years was assumed for this containment barrier. A similar lifetime was assumed for the Furfurol(F) encapsulating the SNF. As the concrete barrier becomes more and more porous, activation products are released from the outside of the RPV. Then the breather hole into the interior of the RPV is corroded open in the year 2005, allowing fuel and interior SS corrosion to begin. The other RPV penetrations and barriers begin to open up in the year 2035, shown by the peak in release rates for the fission products, 370 GBq·a^{-1} and actinides, 0.2 GBq·a^{-1}. Coupled with the continuing steel corrosion, the total peak release rate is 370 GBq·a^{-1}.

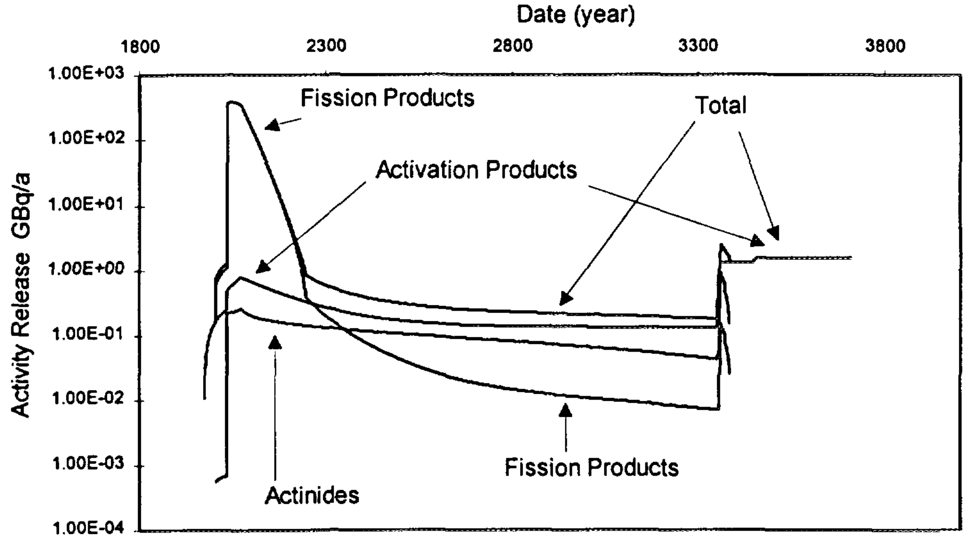

FIG. 20. Novaya Zemlya Depression, unit 421, total activity release for Scenario A.

After this, the fission product release rate falls away quickly as the more mobile atoms are released, until the year 2500. After that, the rate is follows the less soluble atoms in the fission product inventory. The actinide release follows this slow degradation rate; as time goes on, it is seen however, that the overall rate of activity release is dominated by the activation products. Between the years 2500 and 3360, the release rate is of the order of 0.2 GBq·a^{-1}.

In the year 3360, the RPV has finally disintegrated, replacing the slow effective rate of corrosion for the interior of the structure by the faster BCR, and a short peak in fission and activation product release is seen, totalling 2.5 GBq·a^{-1}. After that, the fuel rapidly corrodes and is corroded away by the year 3385.

The internal RPV cladding and the thermal shields are finally corroded away by the year 3710; in this last period, the release rates are dominated by the long-lived activation products at 1.6 GBq·a^{-1}.

Figure 21 shows the same situation for unit 421, but for the overall Scenario C release. All original processes detailed above occur until disintegration of the unit in the year 3000. Then, a total of 630 GBq is released into the Kara Sea in that year, from what remains of the unit.

Figures 22 and 23 show the same scenarios for the units in Abrosimov Fjord. These diagrams are a summation of the releases from all the units, with and without SNF, and take into account their different dumping dates. The units were dumped in their RCs, which represents an outer containment barrier, and again, the units with SNF have an interior protection of Furfurol(F).

In Figure 22, the overall release for Scenario A, the peak release comes in the year 2040, with a total of 2700 GBq·a^{-1}. After that, the fission product contribution to the release decreases and the activation products begin to dominate the total release. By the year 2350, the rate is on the order of 3.3 GBq·a^{-1} until the year 2690, when the RPVs have corroded away, allowing the fuel release peak to be seen again. The fuel has corroded away by the year 2740, and the remaining steels by the year 3075.

Figure 23 shows the overall release for Scenario C for Abrosimov Fjord. This shows that the fuel has corroded away and only the activation products, at 880 GBq, remain to be released into the Kara Sea in the year 3000. Comparing Figures 21 and 23 for Abrosimov Fjord shows how the concrete barrier of unit 421 has extended the life of the fuel material into the glacier scenario.

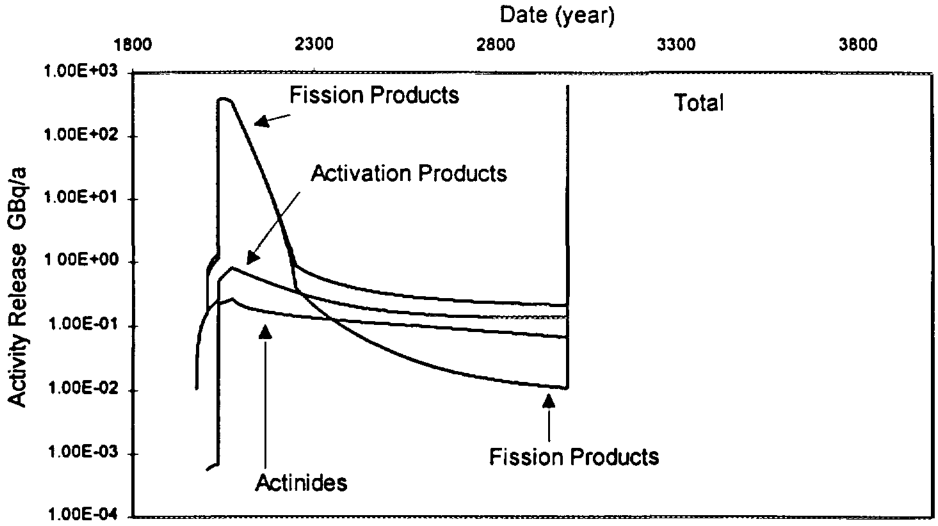

FIG. 21. Novaya Zemlya Depression, unit 421, total activity release for Scenario C.

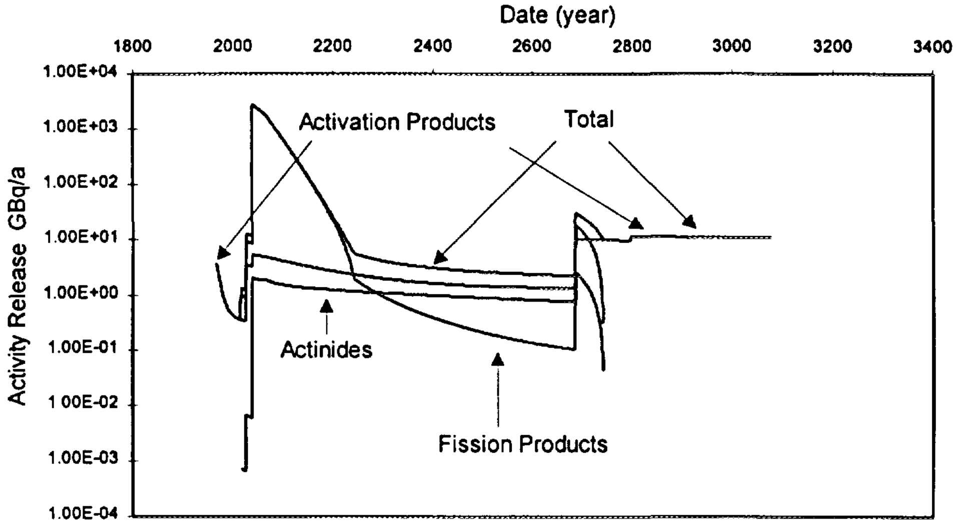

FIG. 22. Abrosimov Fjord, units 254, 260, 285 and 901, total activity release for Scenario A.

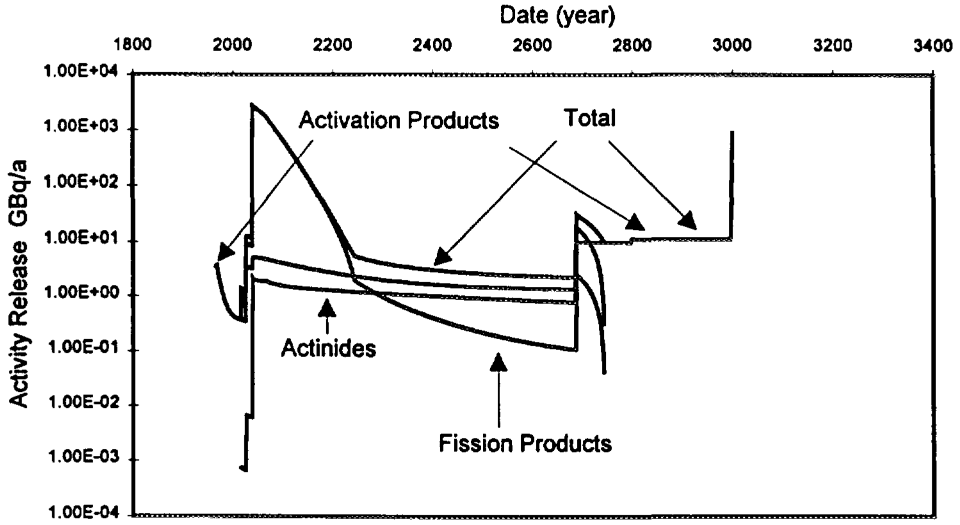

FIG. 23. Abrosimov Fjord, units 254, 260, 285 and 901, total activity release for Scenario C.

Figures 24 and 25 show the situation for unit 538 in Techeniye Fjord. Here, the unit was dumped without SNF in 1988 and only the activation products in the steels corrode away, releasing their activity. Starting at 0.8 $GBq\cdot a^{-1}$, the release rate falls to 0.06 $GBq\cdot a^{-1}$ at the year 2655, when the RPV finally corrodes away. Afterwards, only the cladding and the thermal shields are left, and by the year 3075, they too have corroded away. The final release rate is 0.4 $GBq\cdot a^{-1}$. Figure 25 shows the peak in the year 3000 due to the glacier disintegration of the remaining material.

3.4.2. Submarine liquid metal reactor releases

Overall Scenario A release rates are shown in Figure 26. From the dumping in 1981, initially, no active material is expected to appear due to the hull and bitumen barriers. When it does start, the initial fission product and actinide release is less than 0.0001 $GBq\cdot a^{-1}$ at the year 2105 when the fission product and actinide inventory in the corroded left board SG starts to appear. Corrosion of the outer surfaces of the activated RPVs by that time contributes about 8 $GBq\cdot a^{-1}$.

By the year 2180, fission products and actinides from the damaged and undamaged reactor cores join the release stream and the total release rate is of the order of 5 $GBq\cdot a^{-1}$. By the year 3000, the rise in release rate of the SNF and thermal shields of the reactors, caused by the expanding circles of corrosion exceeds the fall due to decay, and the release rate rises until the year 5200 when the ECTs merge to form an annulus. The release rate then varies as the shields external to the ECTs corrode away by the year 6800 and the left board SG loses all its SNF by the year 7500. By this stage, the release rate has fallen to 0.07 $GBq\cdot a^{-1}$.

The release rate remains steady until the year 11400 when the thermal shields corrode away. Release rates then continue at 0.004 $GBq\cdot a^{-1}$ from the SNF and the Pb-Bi, as the reflector is attacked by corrosion. A small rise occurs in the year 18400 as the reflector corrodes away and the water is able to attack the outside of the core. Release then continues at a rate of 0.007 $GBq\cdot a^{-1}$ until all the SNF has corroded away in the year 36580.

The overall release for Scenario C for unit 601 is shown in Figure 27. The peak in the year 3000 is of the order of 2100 GBq. Note the change in time-scales between Figures 26 and 27.

Despite the prediction from the model that the first release will not occur until early in the 21st century, a sample of seafloor sediment was taken by the Joint Norwegian-Russian Expert Group in 1993 showing Eu isotopes ^{152}Eu and ^{154}Eu and higher than expected levels of ^{137}Cs [28]. As the sample site was close to submarine factory number 601, it was suggested in the report that this activity might have originated from the LMRs.

Although this is a possibility, the first predicted pathway to the damaged SNF is via the welded body of the SG. To open this path requires the corrosion of two SS welds of 20 mm thickness and the degradation of the Furfurol(F) inside the SG and the bitumen surrounding the whole SGI. However, the SNF will have a mixture of fission products and actinides, including ^{155}Eu and ^{137}Cs.

A larger source of Eu, but with no Cs, is in the EPR and CCR channels of the RPV structure itself. However, this pathway is not predicted to leak until the end of the 21st century.

The model results and the observations conflict; this might be because:

(1) The RC was contaminated by ^{152}Eu, ^{154}Eu, and ^{137}Cs before disposal.
(2) The Eu and Cs in the sediment comes from another source.
(3) The bitumen and Furfurol(F) barriers have been breached and corrosion of the SG, reactor cap, and CCR cap welds has been much faster then anticipated; the SGI is in fact leaking.

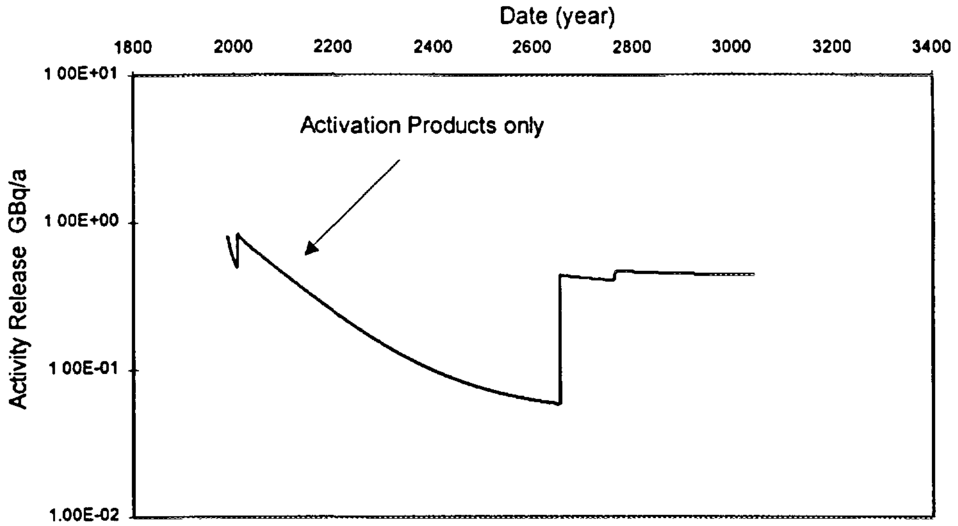

FIG. 24. Techeniye Fjord, unit 538, total activity release for Scenario A, activation products only.

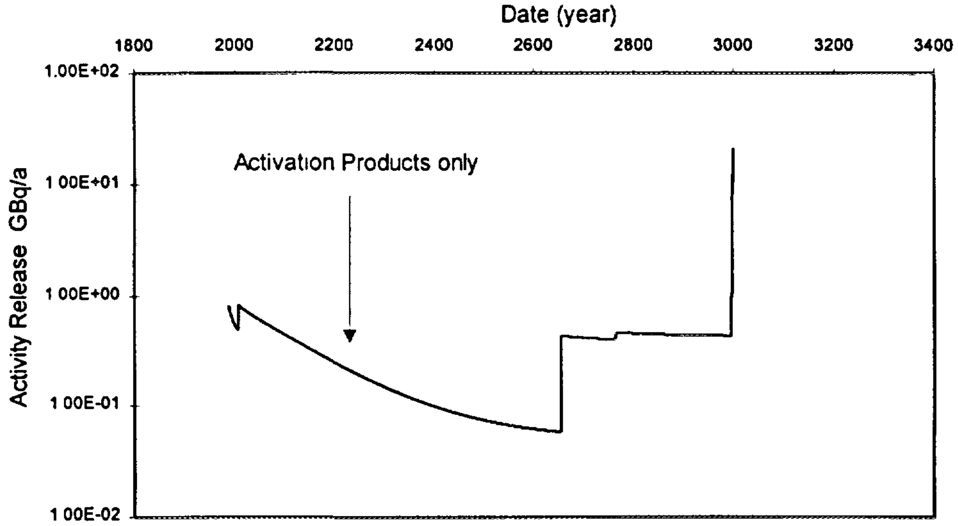

FIG. 25. Techeniye Fjord, unit 538, total activity release for Scenario C, activation products only.

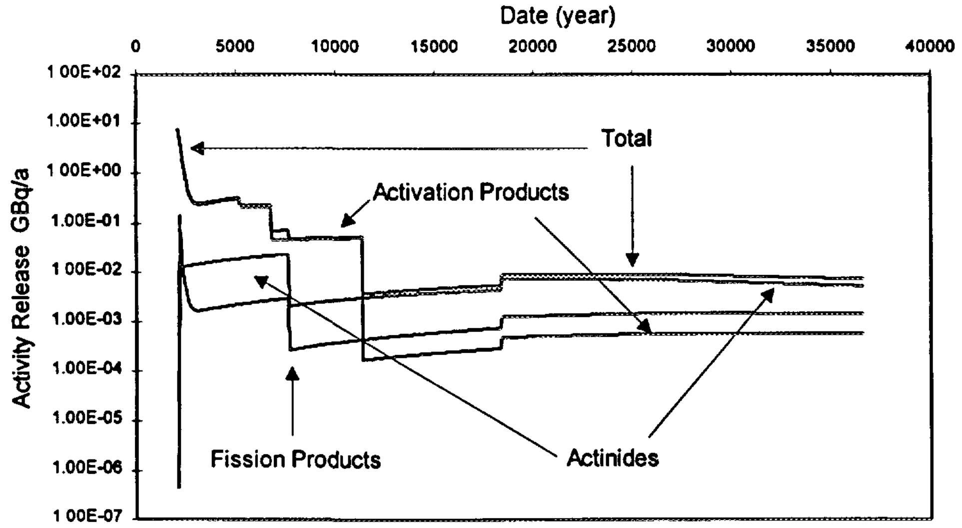

FIG. 26. Stepovoy Fjord, unit 601, total activity release for Scenario A.

If (1) and (3) were the case, it is thought that the other isotopes from the LMR would have been observed in the sediment sample. It is suggested that the observations be explained by contamination coming from another source in Stepovoy Fjord.

3.4.3. Icebreaker pressurized water reactor releases

Total release rates for fission products, actinides, and activation products into Tsivolka Fjord are shown in Figures 28 to 33. As the two dumped components are resting close to each other in the Fjord, the release rates shown are from both units combined.

3.4.3.1. Fission product and actinide release

Figure 28 shows the overall release for Scenario A. Figures 29 and 30 break this down to illustrate the fission product and the actinide releases separately.

Figures 29 and 30 show the peak generated as the containers on the pontoon are finally breached in the year 2305. Five hundred GBq of fission products and 1600 GBq of actinides are immediately released to the Kara Sea from the cracks and porosity of the damaged fuel. In the following year, the rate of release reverts to the calculated corrosion rate of the oxide fuel; fission product and actinide release rates are 1.7 $GBq{\cdot}a^{-1}$ and 5.7 $GBq{\cdot}a^{-1}$·, respectively. The fuel slowly corrodes away and the activity of the fuel itself decreases; in the year 3300, the release rates for fission products and actinides are 0.05 $GBq{\cdot}a^{-1}$ and 2.7 $GBq{\cdot}a^{-1}$, respectively. The fuel is finally corroded away by the year 4570.

3.4.3.2. Activation product release

The activation product release rate shown in Figure 31 is more complicated as various k factors come into play on the two dumped units. Again using Scenario A, the hull surrounding the RPV compartment is first breached in the year 2030, commencing outer wall RPV corrosion release, with a small contribution from material inside the RPVs through gaps with are assumed to exist in the bolted flanges. The release rate starts at 2.3 $GBq{\cdot}a^{-1}$, but five years later the hull fails completely and the release rate jumps to 23 $GBq{\cdot}a^{-1}$. Caps on the main coolant pipes, control rod tubes, and other penetrations fail in the year 2055, increasing the release rate to 41 $GBq{\cdot}a^{-1}$. Until the year 2300, the release rate falls, largely due to decay of the ^{60}Co component of the total activation activity.

At the year 2305, the containment of the Containers A and B on the pontoon is breached and the thermal shields of reactor N2 are exposed to corrosion at the full BCR. This shows as an increase in the overall release rate to 22 $GBq{\cdot}a^{-1}$. These shields are corroded away by the year 2665 and the rate drops to 3.6 $GBq{\cdot}a^{-1}$, as the only material left is in the RPV assemblies.

The RPVs within the RC corrode away at the year 2700 and their cladding by the year 2795. This exposes the remaining material left in the thermal shields to corrosion at the full BCR and the release rate jumps to 6.0 $GBq{\cdot}a^{-1}$. The rate falls away gradually for the final 250 years until all the activated material is corroded away by the year 3050. From Section 3.4.3.1 (Fission product and actinide release), there is still an appreciable amount of fuel material left to corrode away after the activated steels have corroded away.

3.4.3.3. Icebreaker total release rates

As indicated earlier, Figure 28 shows the overall release for Scenario A. The initial release is dominated by the activation product material from the RC, with the peak between the years 2035 and 2055 on the order of 22 $GBq{\cdot}a^{-1}$. Release rates drop until the year 2305 when Containers A and B holding the damaged SNF open and the fission product and actinide peak of 2100 $GBq{\cdot}a^{-1}$ dominates the release in that year.

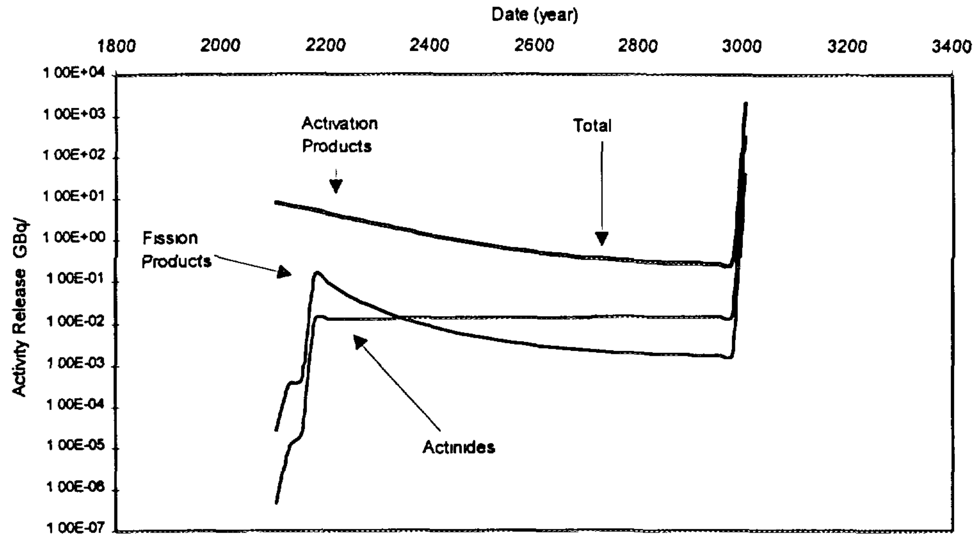

FIG. 27. Stepovoy Fjord, unit 601, total activity release for Scenario C.

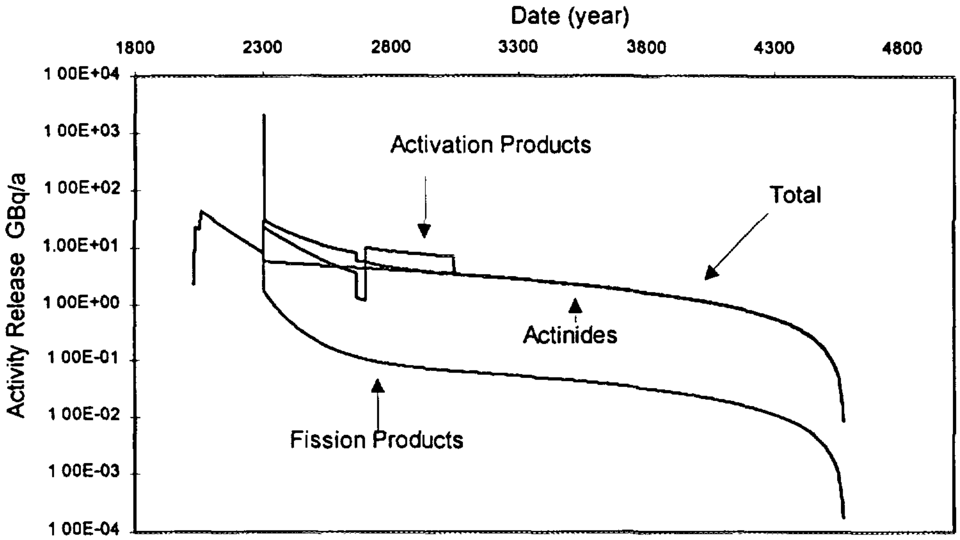

FIG. 28. Tsivolka Fjord, icebreaker reactor compartment and fuel container, total activity release for Scenario A.

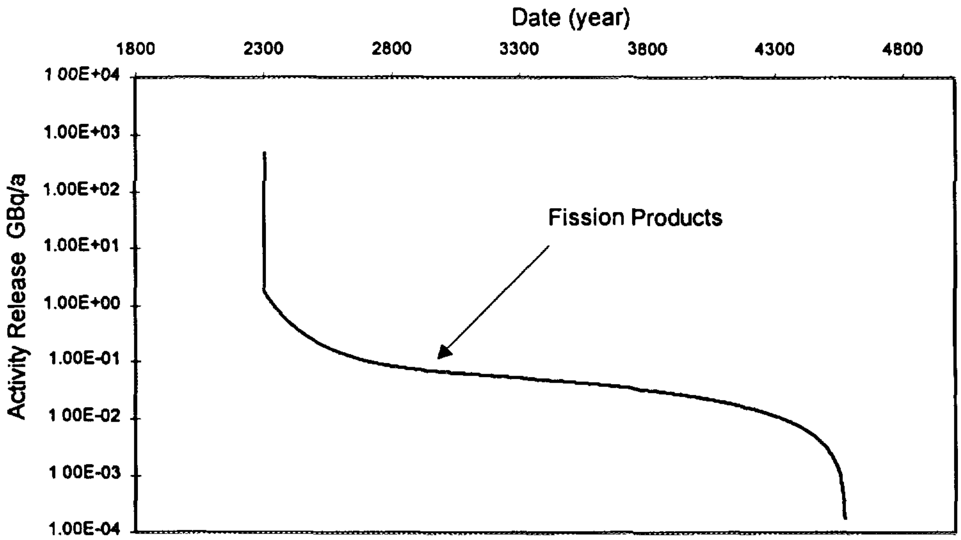

FIG. 29. Tsivolka Fjord, icebreaker reactor compartment and fuel box, fission product release for Scenario A.

The decrease in the total release rate at the year 3050, when the last of the thermal shields corrodes away, is marked as the rate drops to 3.4 GBq·a^{-1}. This leaves only the fuel to continue to corrode on to the year 4570.

Figure 32 shows the overall release for Scenario B. A munitions accident or other mishap is postulated in Tsivolka Fjord, near the dump site in the year 2050. This accident is assumed to breach Containers B and C on the pontoon, and also break off the RPV lids in the RC. The contents of all the units are then exposed to BCRs, with no containment barriers.

The initial corrosion rate is observed, then the accident produces a release peak of 110 000 GBq·a^{-1}. Due to the increased corrosion rates, the activated components are now corroded away by the year 2565 (as opposed to the year 3050 for Scenario A) and the fuel corrodes away by the year 4320 (as opposed to the year 4570 for Scenario A).

Figure 33 shows the overall release for Scenario C. The situation up to the time of glacial disintegration of the units in the year 3000 is assumed to be that for Scenario A, then the remaining active material (a total of 3000 GBq) is released to the environment in that year.

3.4.4. Kara Sea total release rates

Figure 34 shows the Scenario A total radionuclide release rates from all the dumped PWR SGIs dumped into the Kara Sea. Unit 601 is excluded from this diagram because of the much extended time frame for its corrosion. The 3000 GBq·a^{-1} peak from the release of PWR fuel between the years 2030 and 2050, the 2100 GBq·a^{-1} peak from the fuel container in the year 2305, and the sharp fall when thermal shields corrode away and cease to contribute to the total release rate are noticeable.

Figure 35 shows the same Kara Sea total for the release Scenario C. Here, the release in the year 3000 totals 4500 GBq·a^{-1}.

4. RESULTS AND ANALYSES

4.1. RELIABILITY

The main sources of errors in the predicted release rates are as follows:

(1) information on the steam generating installation structures and materials;
(2) radionuclide inventory;
(3) values of BCRs; and
(4) degree of pessimism used in the models.

4.1.1. Information on the steam generating installation structures and materials

The reliability of the information on the submarine and icebreaker SGI structures and materials centers on the reactor cores, the thermal shields, the RPVs, and their associated support structures. Core details for the submarine LMRs is the most comprehensive, with information on the fuel rod materials, dimensions, and pitch, CCR and EPR materials and locations, and overall materials distribution. For the icebreaker PWRs, the information is essentially limited to the fuel rod materials, dimensions, and pitch and the configuration of the control rods (CCRs) and scram rods (EPRs) within the cores. In both cases, the core details are substantially more reliable than the core details for the submarine PWRs, where assumptions were made. Information for the thermal shields and the RPVs is reasonably complete with regard to the materials, dimensions, and locations of one with respect to another. The most detail is associated with the submarine LMRs. Information on the support structures for the reactor cores, thermal shields, and RPVs has not been provided to any significant degree. As such, it is the area where the largest number of assumptions have been made and where the data is the least reliable. With respect to future concerns of potential reactor criticality and remedial actions, the lack of details on the support structures may impact future decisions.

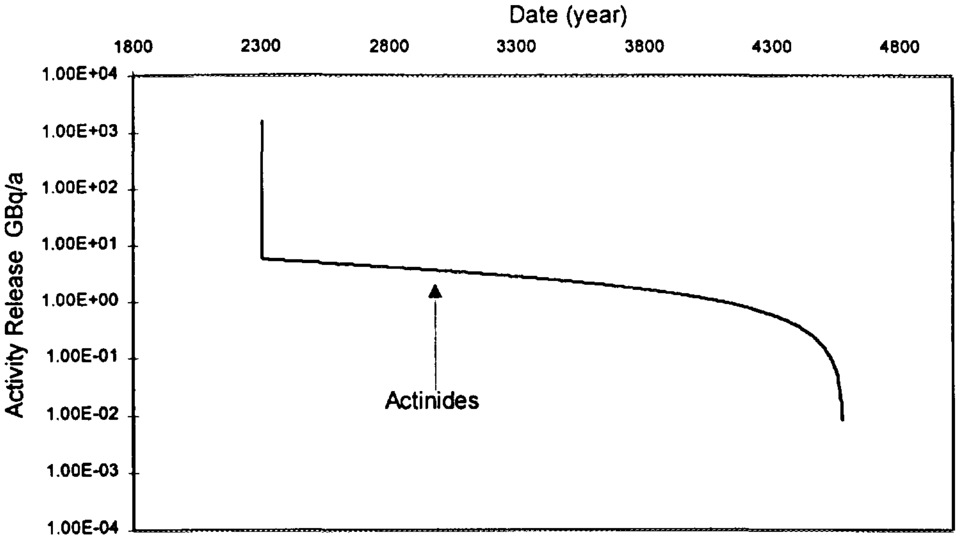

FIG. 30. Tsivolka Fjord, icebreaker reactor compartment and fuel box, actinide release for Scenario A.

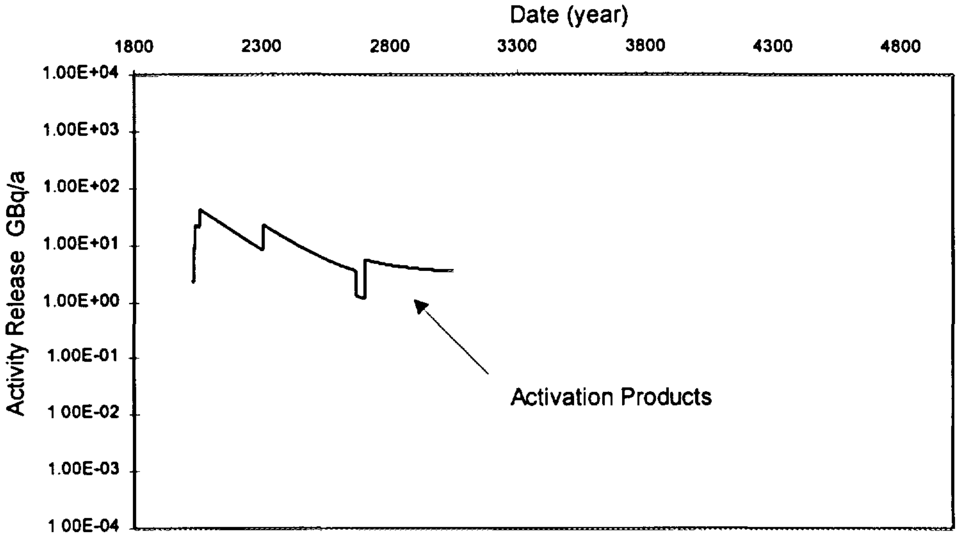

FIG. 31. Tsivolka Fjord, icebreaker reactor compartment and fuel container, activation product release for Scenario A.

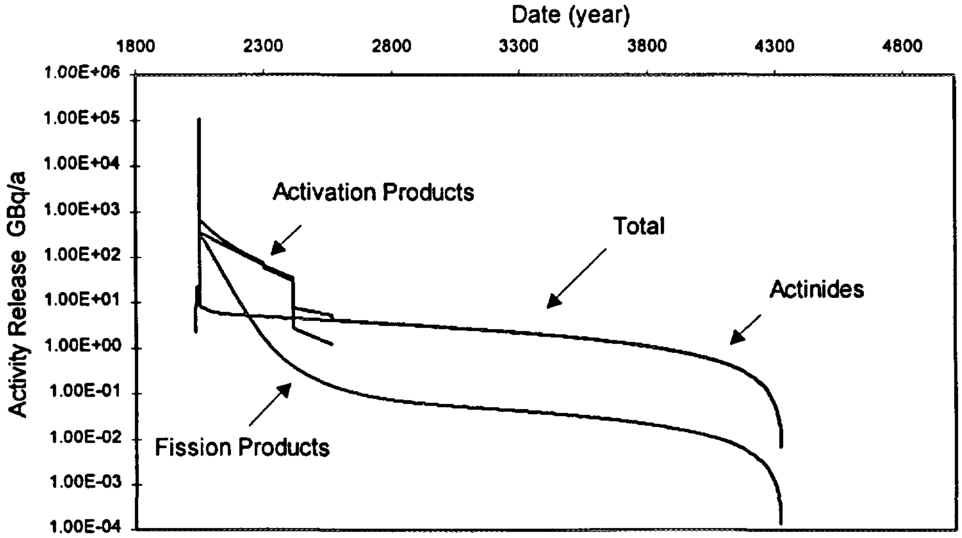

FIG. 32. Tsivolka Fjord, icebreaker reactor compartment and fuel container, total activity release for Scenario B.

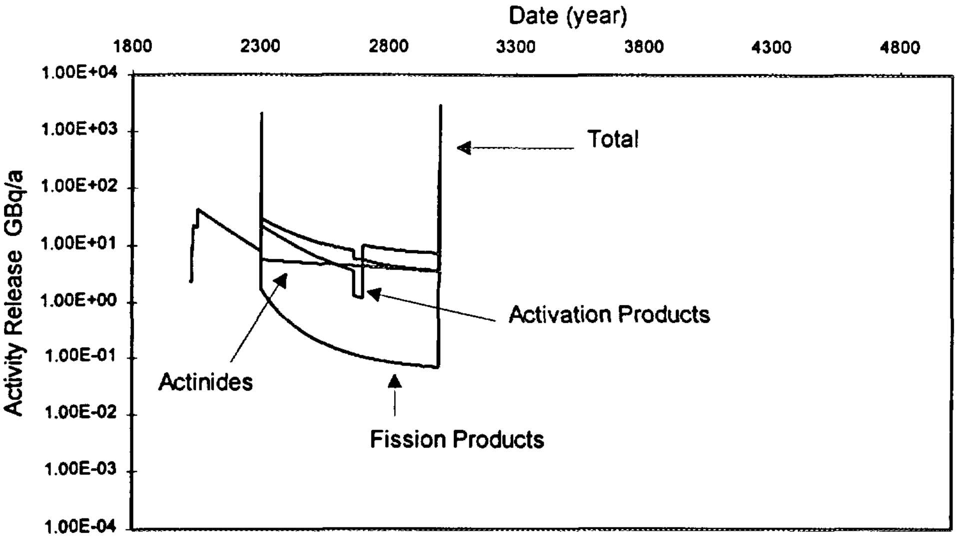

FIG. 33. Tsivolka Fjord, icebreaker reactor compartment and fuel container, total activity release for Scenario C.

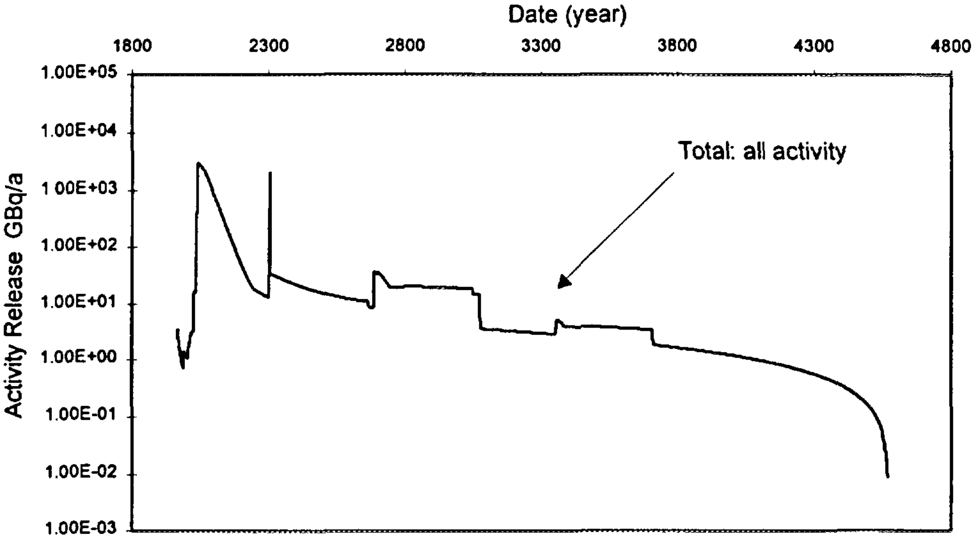

FIG. 34. Kara Sea, all units except 601, total activity release for Scenario A.

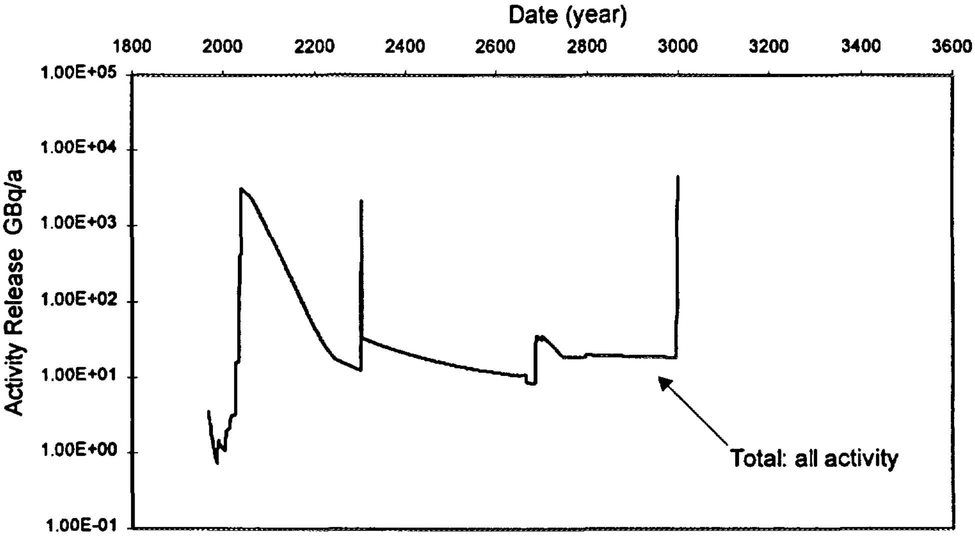

FIG. 35. Kara Sea, all units except 601, total activity release for Scenario C.

4.1.2. Radionuclide inventory

The reliability of the radionuclide inventory centers on the details of the reactor cores, their associated reactor operating histories, and the core models used in the calculations. The reliability of the core details have been described in Section 4.1.1 (Information on the steam generating installation structures and materials). Suffice it to say that the submarine LMR and icebreaker core details are the most reliable. Reactor operating histories obtained from RRCKI [3, 8] and IPPE [7] records for the submarines and icebreaker, when compared with those from Russian Navy records [10], were found to differ by no more than a factor of 2 and on average yielded a comparative ratio of 1.0 ± 37%. As such, the reactor operating histories used in the Group calculations were considered reliable. Core models used in the Group calculations have been discussed in Section 2.4 (Radionuclide Inventories). Models used for the submarine LMRs and icebreaker PWRs represent the actual configurations; models used for the submarine PWRs are assumed the same as that of the icebreaker. In addition, when compared to an independent estimate of the radionuclide inventories prepared by SIAE [10], the Group inventories were determined to represent the best estimate for the IASAP. With respect to potential radionuclide release, the submarine LMR and icebreaker radionuclide inventories are deemed the most reliable, while the submarine PWR radionuclide inventories are deemed the least reliable.

4.1.3. Values of the best corrosion rates

The reliability of the values of the BCRs used is dependent upon the type of material considered. Degradation rates are best estimates based on currently available data, some of which is scant for materials such as the Pb-Bi eutectic and U fuels in their various alloy and oxide ceramic forms. The different types of material may be sub-divided as follows:

(1) *Structural materials.* Extensive literature searches have been undertaken for common structural materials such as SSs and mild steels, and the BCRs employed are believed to be on the conservative side of the range of values examined [23, 29]. Soo [30] reviewed values determined by the IASAP, and considered them reasonable for the bulk and pitting corrosion rates of SS. Sufficient data exists on mild steels (low alloy and carbon) to justify their use and reliability.

(2) *Pb-Bi coolant (LMR only).* In the absence of data on Pb-Bi, the best available alternative was the known corrosion performance of lead in sea water. However, there is a possibility that the behavior of the Pb-Bi eutectic may be different, and some doubt must therefore exist on the reliability of this data. Given the significance of this material with regard to its likely inhibition of fuel corrosion in the core, and indeed with regard to its thermal expansivity and the possible presence of voids in the core, further information on this material is essential.

(3) *Fissile fuel.* Research data on the corrosion of U fissile fuels in sea water is very limited. The best available data has been used for the dissolution rate of the icebreaker oxide fuel.

(4) *Filler materials: Furfurol(F), bitumen, and concrete.* Alexandrov, et al. [27] confirmed the lifetime of Furfurol(F) as 100 years, in a project conducted during the period of the study. Carter [22] helped in assigning values for concrete and bitumen at 100 years as well, but without better information as to their makeup, these must remain as best estimates.

4.1.4. Degree of pessimism used in the models

In order to produce pessimistic but realistic values of radionuclide release, several assumptions were made:

(1) All material corroded is immediately available to the environment, and is regarded as released for the purposes of the IASAP models. Making this assumption negates the need for analyses of solubility and transport mechanisms through the containment, and immediately demonstrates that the IASAP release rates must be pessimistic. In practice, of course, much of the corroded material will be both heavy and insoluble and will remain within the containment, reducing the true release rates. This fact has been discussed in the context of reactor criticality, where it was

assumed that about 80% of the corroded SNF was insoluble and would fall to the bottom of the RPV. Retention of corroded material could further reduce the release rate as its presence may slow or prevent the flow of water through the reactor, retarding or preventing further corrosion, and preventing further release of material.

(2) BCRs used for release rate calculations were chosen to give fast release rates, adding further pessimism to the models. Also, BCRs applied to the SGI materials were constant, whereas some studies indicate that true corrosion rates will decrease with time as a corrosion resistant layer forms on the surface of material under attack.

(3) Fuel pin cladding has been ignored in the IASAP release rate models. Inclusion of the cladding, even when the fuel has been substantially damaged, will retard or reduce the release rates.

(4) The filler materials were assigned a lifetime and rate of degradation, at the end of which they ceased to be barriers to corrosion or transport of corroded material. There is little information available on the long-term behavior of the filler materials in sea water and in a radiation environment; if filler lifetimes are longer, the release rates in the first few hundred years will be less. In reality, the filler material will not simply disappear at the end of its lifetime, and may still retard or prevent release of material.

4.2. SENSITIVITY

By its very nature, the IASAP release rate model is sensitive to the chosen corrosion or degradation rates for the steel components and for the barrier materials. Although an extensive literature study was undertaken and advice sought from materials specialists from the contributing countries, in order to provide the best estimates, nevertheless, steel corrosion rates in the ocean can have a wide spread of values and barrier effectiveness is not well defined.

The model can be run using a range of the rates. An example might be to vary one or more of the parameters, e.g., using a maximum SS pitting corrosion value, and a 500 year Furfurol(F) lifetime, as opposed to the IASAP recommended lifetime of 100 years. To illustrate this, the model was run using three sets of corrosion rates and barrier lifetimes. The rates or lifetimes were chosen at the upper and lower ends of the published or advised values (see Table XVII), and compared to the IASAP recommended values.

Figure 36 shows the result of one such run, using the Abrosimov Fjord units and the release of Scenario A. Using the maximum rates, all the units have corroded away by the year 2370. With the IASAP BCR rates, the time is extended to the year 3075. For the slowest rates, the units survive until about the year 5000. As a general principle, there is a factor of 3 between release rates for the highest degradation rates and the IASAP rates and a factor of 30 between the IASAP rates and the slowest rates. The peak for the maximum rates is 12 000 GBq·a^{-1}, the peak for the IASAP rates is 2700 GBq·a^{-1}, and the peak for the slowest rates is 90 GBq·a^{-1}.

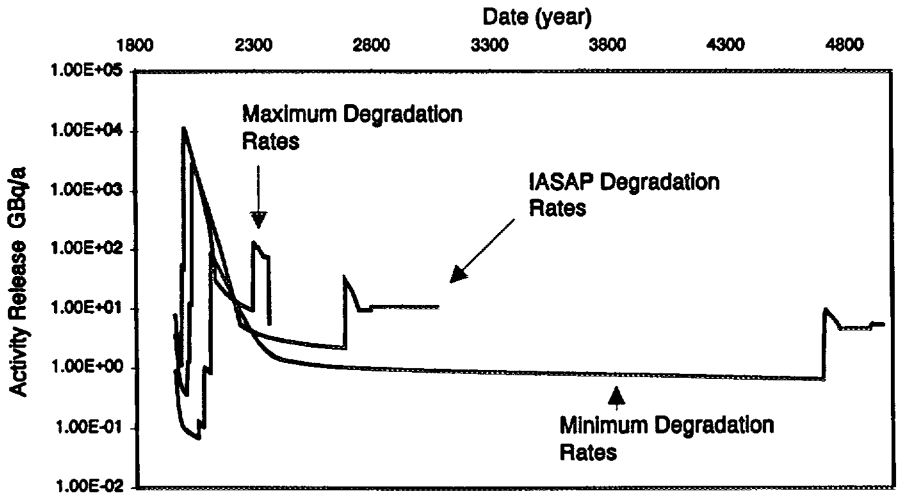

FIG. 36. Sensitivity analysis, Abrosimov Fjord: Scenario A total release rates with maximum, minimum and IASAP rate values applied.

5. REACTOR CRITICALITY

The radionuclide release rates predicted by the corrosion model took no account of any change in corrosion rates or fission product inventories due to possible criticality. However, since very little of the fuel had been used in the cores, it was decided to investigate the possibility of criticality being achieved as corrosion progressed. If a reactor core could achieve criticality, this could potentially have affected the predicted radionuclide release rates in two ways:

(1) The energy released by fission would cause an increase in temperature in the core, leading to accelerated corrosion rates and hence release rates;
(2) Criticality over a long period would produce a fresh inventory of fission products, some of which would have short half lives not present in the present inventories.

Three scenarios were reviewed to investigate the possibility of any of the reactor cores achieving a critical state:

(1) Corrosion of a large proportion of the control rod material before the SNF has substantially corroded away. This could have the same effect on core reactivity as control rod withdrawal during a reactor start up, and cause criticality.
(2) The corrosion process or certain forms of attempted remedial action cause a structural change within the core, such as the SNF falling to the bottom of the RPV or control rods being displaced, resulting in some or all of the SNF and core material reaching criticality.
(3) Ingress of water into the core causing an increase in neutron moderation sufficient to cause criticality.

As a very rough estimate, a marine PWR core will reach the end of its useful life once about 40% of the fuel has been used, after which point it must be refuelled. For a core with a fuel load of 50 kg of ^{235}U, this equates to 30 kg remaining. This end of life fuel load gives an indication of how much fuel is required for a critical assembly of a lattice of fuel pins in water. Hence, for all core types, in the following analysis a mass of fuel of less than 30 kg will be considered to be unable to achieve criticality, while a mass of more than that *may* achieve it.

Analysis of the core burn-ups, assuming 1.25 g ^{235}U per GW·d burn-up rate, showed that none of the submarine PWRs with SNF had used more than about 3.5 kg of ^{235}U; hence a typical dumped SGI fuelled PWR core could be assumed to contain about 47 kg of fissile material, or 94% of the original (start of life) fuel load. The right board LMR in submarine factory number 601 was undamaged and so could be assumed to contain around 89 kg of ^{235}U; the left board core lost 20% of its load as a result of the accident, so was assumed to contain 71 kg.

During normal operation, the reactors would have been operated with EPRs fully raised to allow safe emergency shut down of the reactor and some the CCRs at a height which maintains critical condition. This operating rod height must have been able to vary by as much as possible to allow for poison changes; the optimum rod height for the control rods would have been about half core height, or 50% withdrawn. In practice the CCRs may have been operated in groups at different heights to alter the neutron flux profile across the core and compensate for fuel burnup and long term poison changes. No data was available on submarine PWR rod layout, but the LMRs had 3 EPRs and 10 CCRs.A typical operating configuration of the LMRs would have allowed criticality with ten CCRs at around 50% withdrawal and the 3 EPRs fully withdrawn. This equates to 38% of the total mass of rod absorber in the core, or 62% removed. In the following analysis, it is assumed that a core could possibly become critical if 62% of the absorber material were to be removed.

5.1. CONTROL ROD CORROSION

All the corrosion models studied by the IASAP use the control rod channel as one of the primary ingress routes for sea water to the core; thus the control rods are subject to corrosion by water as soon as water ingress occurs. For this scenario to allow criticality, the effective corrosion rates of the SNF and the rods must be such that at least 62% of rod material can corrode away before the fuel mass becomes too low to allow criticality, i.e. below 30 kg. The rods and SNF are assumed to be subject to the same containment and filler k factors, so comparison of the control rod and SNF BCRs will be used in the following analysis. However it should be noted that the time values obtained will be incorrect, as the effective corrosion rate will be two orders of magnitude slower than the BCR owing to the containment k factors.

In the case of the submarine PWRs, all controls rods were assumed to be constructed of SS, enriched with neutron absorbing material. The BCR of the U-Al alloy is 0.03 mm a^{-1}. If, as discussed above, it is assumed that 47 kg (i.e. 94%) of fuel was left at the time of disposal, then 30 kg of fuel will be left when the fuel pins radius has decreased from 5 mm to 4 mm. This will take 33 years at the BCR of U-Al alloy (or 1100 years taking into account the k factors). As discussed above, during normal operation about 38% of the total rod material could be inserted in the core; hence for criticality to occur, about 62% or more of the control rod material must corrode within 33 years. The control rods have a radius of about 10 mm, so to allow 62% to corrode requires a maximum control rod radius after 33 years of 6.2 mm and hence a minimum corrosion rate of 3.8/33 = 0.11 mm a^{-1}, which is not much more than the maximum accepted IASAP SS corrosion rate of 0.1 mm a^{-1}. However, using the minimum IASAP value for the U-Al corrosion rate of 0.015 mm a^{-1}, the assumed minimum critical mass of 30 kg will be left after 44 years. The SS corrosion rate required to allow 62% of the rod material to corrode in 44 years is then 3.9/44 = 0.08 mm a^{-1} which is within the IASAP range of SS corrosion rates. This example suggests that criticality by corrosion of control rods alone is possible if the ranges of values of the IASAP corrosion rates are accepted as realistic.

The icebreaker PWR used UO_2 fuel with a BCR of 0.0011 mm a^{-1}. However, Container C contained only 60% or 20.6 kg of the total fuel from the N2 reactor which is assumed here to be too low an amount for a critical assembly to form.

The LMR reactors had highly enriched fuel and low burnup. Of all the dumped cores, they contain by far the greatest amount of SNF. However, the ten CCRs were encased in Pb-Bi for 500 mm above the core; the Pb-Bi fills the CCR channels and water can only corrode it from above. Hence the water would have to corrode through approximately 500 mm of Pb-Bi before attacking the EuB_6 in the CCR channels, which would take at least 50 000 years at the BCR for Pb-Bi. Using the IASAP assumptions, the 3 EuB_6 EPRs will begin to corrode after 125 years when water penetrates the elliptic shield, bitumen and EPR channels, and will have corroded completely in around 1000 years. However, corrosion of the 3 EPRs alone is insufficient to cause criticality, as the 10 CCRs will still be fully inserted in the core. As discussed in 3.2.2.3, the IASAP corrosion model of the LMRs predicts that water will corrode outwards from the EPR channels, removing SNF and Pb-Bi coolant. Eventually the water will reach the CCRs, which are about 160 mm to 173 mm from the EPRs. This will occur at around 17 000 years after dumping, after which the rods will corrode in about 1000 years. The amount of fuel remaining after 18 000 years can be approximated by taking the ratio of the area of the expanding circles to the total area of the core, as shown in Fig. 18; this ratio is around 0.5, i.e. the corroded fuel represents about half of the total initial amount. Hence at the stage when the CCR material corrodes away, about half the fuel is left in the right board reactor, and about 40% in the left, since approximately 20% of the SNF was transported to the steam generator in the accident. This equates to around 44 kg of fuel in the right board core, and 35 kg in the left. Hence the IASAP corrosion model suggests that at around 18 000 years after dumping the remaining control rods could corrode away, leaving sufficient ^{235}U to form a critical assembly.

5.2. STRUCTURAL CHANGES

5.2.1. Corrosion of the stainless steel supporting structure

Alterations in core structure caused by corrosion are, by their very nature, complex and difficult to quantify. An example of a scenario which may lead to criticality would be disintegration of the

Furfurol(F) filler after its assumed lifetime and subsequent corrosion of the SS structure holding the SNF, causing the fuel pins to fall into the bottom of the RPV. If sufficient U were left in the fuel pins, and the configuration of the fallen pins formed a suitable arrangement in a small enough volume, a critical assembly could result. The possibility of this would also depend on the amount and arrangement of other materials such as water and corroded control rod material, and would be extremely difficult to model accurately. The rest of this section attempts to determine whether enough U could arrive in the bottom of the RPV to allow the possibility of criticality, but does not attempt to estimate the probability of the U forming a critical assembly. There is little information available about the configuration and thickness of the SS supporting structures; an estimated thickness of 3 mm will be used in this analysis.

Again an assumption is made that 47 kg of ^{235}U remains in the core. In order to leave 30 kg in the pins to provide a critical mass, the pins must only have lost 17 kg of SNF. This will occur if the fuel pin radius has decreased to 4 mm when the SS structure corrodes away.

However, this does not take into account the SNF already corroded. Although the IASAP models assume that all corrosion products are released to the environment, and so provide the most pessimistic release rates, in practice a large fraction of the SNF is likely to be insoluble and remain in the RPV, collecting at the bottom as loose material. The IASAP corrosion model assumed 20% of the fission products in the alloy fuel were more soluble, leaving 80% effectively insoluble. If 80% of the U is assumed to collect in this manner, then an additional amount is available to provide the critical mass. If the IASAP submarine PWR Greenwich FORTRAN model is run with the SS corrosion rate set at the maximum value, and the U-Al alloy rate set at the minimum, the 3 mm thick SS structure corrodes away by about the year 2700. By then the fuel pin radius has decreased from 5 mm to 3.2 mm, losing 60% of the mass of the fuel pin. This could still provide well over 30 kg of SNF at the bottom of the RPV, as shown in the following calculation:

(1) 60% of the mass of the fuel pins has corroded - 80% of this is insoluble, so 60% × 80% = 48% is lying loose on the bottom of the RPV;
(2) 40% remains in the fuel pins, which fall to the bottom, giving a total of 88% of the mass of the fuel pins at the bottom, and;
(3) 47 kg or 94% of the SNF is not burnt up, so 94% × 88% = 83% of the start of life fuel load is left on the bottom, which gives a mass of 41.5 kg.

It should be noted that even without the failure of the SS supporting structure the corroded U could fall to the bottom of the core, where it could collect in a compact mass which may be less affected by the presence of neutron absorbers such as the control rods. Hence, assuming the corroded U is able to fall past the remaining core structure and the remnants of the Furfurol(F), the possibility of a critical mass being formed from corroded SNF cannot be ruled out.

This analysis does not take into account any cladding on the SNF, which will increase the probability of enough U remaining in two ways:

(1) the cladding will slow the corrosion of the SNF, so more will remain once the SS structure corrodes;
(2) the structural integrity of the fuel pins may depend on the cladding: if the cladding is corroded, the SNF in the form of pellets or small pins may fall to the bottom. The cladding is unlikely to be more than 1 mm thick, so will corrode away before the SS structure, and hence could allow the SNF to fall to the bottom sooner, where more SNF remains.

The presence of neutron absorbing material will increase the critical mass. In the worst case, however, such material may corrode rapidly and cease to affect the core in the time estimated for the rods to corrode.

5.2.2. External corrosion

Corrosion to the external parts of the RPV that weakens its structure has the potential to cause the whole assembly to collapse or topple. The consequences of such an occurrence are difficult to predict. In the worst case, the control rods could be displaced out of the SNF and a critical mass achieved, possibly fast enough to cause prompt criticality (a situation where the increase in reactivity is sudden enough to produce an uncontrolled rise in the fission rate and a rapid, possibly explosive, release of energy). Submarine hull and SGI supporting structures are heavy gauge mild steel, and at an assumed thickness of 25 mm and BCR of 0.08 mm a^{-1} (taking into account the effect of biofouling and corrosion from both sides) could fail after 300 years. The probability of the RPV toppling and the control rods coming out of the core will depend on the configuration of the other structures within the reactor compartment and core.

Any proposed remedial actions which involve lifting or moving the RPV will have to take into account the possibility of the resulting structural change causing criticality: the most catastrophic scenario would be the RPV turning over during transport to shore and the control rods falling out of the core under gravity.

5.3. INGRESS OF WATER

In the case of the submarine PWRs and icebreaker, the SNF is initially surrounded with Furfurol(F), which can be assumed to degrade in 100 years, allowing water to enter the core and surround the fuel pins. This will increase the reactivity of the core; however, in normal operation the control rods will have been designed to produce a safe shut down condition, and with all rods present there is no risk of criticality. Combined with control rod corrosion and compaction of SNF under its own weight at the bottom of the RPV, however, the presence of water will increase the probability of criticality.

The LMRs were never designed to operate with a water moderator: the reactors in operation were intermediate (i.e., unlike in a PWR, the neutrons did not have to be fully slowed down to thermal energies to perpetuate the chain reaction), with Be acting as the neutron moderator. If the Pb-Bi were to be replaced by water, a substantial increase in reactivity would result as the higher slowing down power (a measure of the efficiency with which a material slows down neutrons) of the water would produce a much larger thermal neutron flux and hence greater fission rate in the highly enriched U.

To allow water ingress, however, the Pb-Bi must corrode. This was assumed to occur at the IASAP BCR of 0.01 mm a^{-1}, ten times the rate of the SNF corrosion, so corrosion would remove the Pb-Bi preferentially and replace it with water. The above discussion in 5.1 of control rod corrosion concluded that around 50% of the fuel may be left once water has corroded away all control rod material after about 18 000 years. There are other factors which this analysis did not take into account:

(1) the preferential corrosion of the Pb-Bi, so more fuel pins will be left than the 50% estimated above;
(2) the presence of water will further increase the reactivity, leading to less fuel being required for criticality than was necessary during normal operation, and;
(3) the Pb-Bi corrosion rate has a fastest IASAP value of 0.1 mm a^{-1}, which would reduce the time to reach the CCRs to around 2700 years, or around the year 5000.

Hence it is possible that owing to the combination of corrosion of the CCRs, the presence of water, and amount of fuel remaining, that as soon as about the year 13000 the reactor cores in submarine factory number 601 could slowly achieve criticality.

The IASAP radionuclide release model also assumes that because the Pb-Bi mixture has a negative coefficient of thermal expansivity, the volume of Pb-Bi has not decreased and remains in close contact with the SNF and SS structures, preventing water from seeping through any gaps and coming in contact with the SNF. Prior to dumping there appeared to be no gaps [9]. However, the LMRs were dumped in cold Arctic water, and further thermal contraction may have occurred. If the Pb-Bi volume has reduced, making water ingress in this manner possible, each fuel pin could eventually be surrounded with a layer of water which could cause a larger increase in reactivity. This would then allow other methods of reactivity increase to occur:

(1) corrosion of the Pb-Bi layer around the CCRs leading to earlier corrosion of the CCRs;
(2) corrosion of the SS supporting structures causing the SNF to fall into a smaller volume at the bottom of the RPV, and;
(3) corrosion of the SS cladding of the fuel rods could result in the fuel pellets being released. If the Pb-Bi has corroded sufficiently, they could fall to the bottom of the RPV, possibly forming a critical mass.

Suffice it to say, the thermal expansion or contraction of the Pb-Bi has a large impact on the potential for criticality, and the time at which it occurs.

5.4. CONSEQUENCES OF CRITICALITY

Each of the three scenarios alone is unlikely to cause a critical assembly to be formed; however, they are not mutually exclusive, as, for example, control rod corrosion is caused by water ingress. The combination of two or more ways in which reactivity of the dumped SGI cores can increase hence makes the possibility less remote, particularly for the reactors of submarine factory numbers 421, 901 and 601, which have the most SNF remaining and the higher enrichments.

If criticality is possible and occurs through slow corrosion of the control rods or water ingress, or a combination of the two, the approach to criticality will be extremely slow and the possibility of prompt criticality and any kind of explosion or structural damage can be entirely ruled out. Instead, the onset of criticality will cause a slow rise in fission rate, and an increase in SNF temperature. The conditions in the core are likely to be such that the heat generated is easily dissipated, particularly as the rate of reactivity addition is so slow, and the temperature is unlikely to rise significantly. This may cause a slight rise in corrosion rates and increased flow through the RPV. Since the cores will be water moderated, the rise in temperature will probably cause a reduction in reactivity, and there is a possibility that self regulation could occur. As corrosion continues, the critical state is likely to be short lived compared with the total lifetime of corrosion, with further structural corrosion and loss of SNF leading to a reduction in reactivity and eventual subcriticality. Such behavior is unlikely to have a significant effect on total release rates.

A structural change resulting in prompt criticality is an extremely remote possibility, but cannot be ruled out. If it occurred due to corrosion of the structure resulting in toppling and control rod displacement, the large release of energy that follows would cause the structure to further disintegrate and radionuclide release would be accelerated, possibly causing the total remaining inventory to be released to the environment. If it occurred due to attempted remedial action which involved movement of the RPV, the consequences could be much greater.

A prompt criticality with core disassembly in the far distant future would involve very little radioactivity compared to the present radionuclide inventory in these cores. By the year 2700, nearly all of the current fission product inventory in the cores would have decayed. Also, the amount of fission products produced in a prompt critical excursion is relatively small. For example, the amount of ^{137}Cs generated in a 10^{18} fission criticality excursion (about the same as the SL-1 accident in the USA) would be 0.044 GBq [31].

5.5. FURTHER WORK

The above discussion has highlighted several areas which require further work to quantify the risk of criticality occurring. Areas to be studied should include:

(1) detailed information on the structure and configuration of the cores, including the dimensions and sites of the welds in the SS supporting structures;
(2) detailed modelling of the corrosion processes in the core, applying better estimates of corrosion rates to the core materials;
(3) criticality calculations for various possible configurations of the SNF, once it has dropped to the bottom of the RPV;
(4) the behavior of the Pb-Bi eutectic thermal expansivity over the temperature range 125°C to 0°C; and
(5) analysis of the change in neutron spectrum and reactivity as water enters the core of submarine factory number 601, when introduced as a consequence of bulk corrosion of the Pb-Bi, or if item (4) above suggests the Pb-Bi has shrunk and allowed water to enter prior to bulk corrosion.

6. REMEDIAL CONSIDERATIONS

6.1. REMEDIAL MEASURES

Before any remedial measures are taken, it is assumed that a site survey will be undertaken to establish the structural condition of the SNF containers and evidence of radionuclide leakage. In view of their relatively low levels of activity, this section does not address the issue of remedial measures for non-SNF radioactive material. Also, no attempt is made to assess the *need* for remedial measures, as it is beyond the remit of the Group. However, an estimate is provided below of the physical condition of the SNF containers to assist in any assessment of necessary or feasible actions. It is up to other expert groups to assess whether the environmental impact, technologies, and cost considerations render such actions necessary and or feasible.

Potential remedial measures might include:

(1) reinforcement of the existing containment barriers around the SNF, or
(2) recovery of the dumped SNF for land storage.

6.1.1. Reinforcement of the existing containment barriers

To inhibit any monitored leakage from the SNF containers or enhance current containment arrangements, it may be possible to pump fillers inside existing barriers. Additional barriers may also be erected around the SNF containers, such as capping materials applied to potential leakage paths. Factors to be considered include:

(1) The risk of a disturbance to the SNF containers that may damage existing barriers, perhaps inducing leakage and weakening containment structures.
(2) The risk of a disturbance causing a change in the orientation of the SNF in the RPVs sufficient to displace the CCRs and initiate criticality, to the extent that prompt criticality may occur. The resulting power excursion from prompt criticality could cause significant and immediate radionuclide release to the environment and prejudice the safety of personnel at the scene. Even if slow criticality occurs, the increase in temperature may accelerate the corrosion rates and cause an early breakdown of the containment barriers. If this occurs with the SNF in sealed containers, they may be ruptured by an accompanying rise in pressure, leading to premature breakdown of the barriers to radionuclide release.
(3) The risk of a disturbance causing displacement of the SNF in the RPVs due to the corrosion-weakened or damaged condition of the fuel supporting structures. In this case, the possibility of a critical mass being formed at the bottom of an RPV, beneath the CCRs, cannot be ruled out. This could result in either slow or prompt criticality, similar to the situation discussed in the previous paragraph for displacement of the CCRs.

6.1.2. Recovery of the dumped spent nuclear fuel for land storage

In the event that a decision is made to recover the dumped SNF to a land disposal site, the factors to be considered are rather more wide ranging:

(1) The risk that, through a deterioration in their condition, the SNF containment barriers fail during recovery operations, releasing radionuclides either at the scene or during subsequent recovery and transportation to the land disposal site.
(2) The risk that a disturbance to the fuel supporting structures or a change in the orientation of the SNF induces criticality, as described in items (2) and (3) of Section 6.1.1 (Reinforcement of existing containment barriers).
(3) The nuclear safety hazard associated with the handling and transportation of the SNF to its final disposal site, including the radiation dose to personnel involved throughout the process.
(4) A comparison of the potential hazard to the environment of the SNF in its land disposal site against retention at its sea disposal site.

6.2. STRUCTURAL INTEGRITY OF THE SPENT NUCLEAR FUEL CONTAINERS

Table XVIII provides an estimate of the structural integrity of the SNF containers, at the present time (1996) and in the year 2015, and implications for recovery purposes. This represents a theoretical assessment only, and should therefore not suggest remedial action without confirmation of the actual condition of the SNF containers through site survey.

7. CONCLUSIONS

(1) For the reactors and their associated SNF dumped in the Kara Sea, the isotopes making up the radionuclide inventory and their activities were calculated on the basis of the information on quantity, enrichment, and burnup of the nuclear fuel. These inventories were grouped into three categories: fission products, activation products, and the actinides.

The total activity in 1994 was calculated to be 4.7×10^{15} Bq. Fission products made up 86% of this total, activation products 12%, and 2% came from the actinides. The major component of this total inventory comes from the container containing the SNF from the icebreaker N2 reactor.

It should be remembered that the inventory finally established by the Group for all these units was considerably lower than that originally published in the White Book. The White Book quoted 88×10^{15} Bq as the total for fission and activation products in the dumped objects. The Group established that the original inventory at the time of dumping for all the long-lived radionuclides was 37×10^{15} Bq.

(2) Using best estimates of the corrosion rates of the SGI materials, barrier lifetimes, information on the structures and containment of the SNF and methods used for dumping, a computer model was developed. Calculations were made for the release of radioactivity into the Kara Sea, using several scenarios.

Release rates for all the radionuclides were calculated from objects in each fjord. As an example, using the best estimate scenario and summing the contribution form all the fjords with PWRs, it was shown that a release rate peak of 3000 $\mathrm{GBq \cdot a^{-1}}$ occurs around the year 2040 when the PWR containments are partially breached, and there is another peak of 2100 GBq in the year 2305, when the icebreaker SNF container corrodes open. However, for a large part of the time, release rates lie between 20 $\mathrm{GBq \cdot a^{-1}}$ and 2 $\mathrm{GBq \cdot a^{-1}}$. Some very low levels of activation product releases might be expected, a decade or so into the next century from the corrosion of the outer walls of the RPVs.

TABLE XVIII. ESTIMATES OF THE CONDITION OF DISPOSED SPENT NUCLEAR FUEL CONTAINERS FOR REMEDIAL ACTION CONSIDERATIONS

Factory number and reactors	Disposal date	Disposal site	Containment structures	Condition		Conclusions
				1996	2015	
901 2 PWR	1965	Abrosimov Fjord	a. Reactor compartment (RC) - hull b. Mild steel bulkheads c. Reactor pressure vessel (RPV)	a. 25% corroded - sound b. Unsound c. Little corroded - sound	a. 40% -weakened b. Unsound c. 5% corroded - sound	RC bulkheads now (1996) believed ineffective as containment barrier, It may not be possible to use hull for recovery purposes. RPV remains intact.
285 1 PWR[1]	1965	Abrosimov Fjord	a. RC - hull b. Mild steel bulkheads c. RPV	a. 25% corroded - sound b. Unsound c. Little corroded - sound	a. 40% -weakened b. Unsound c. 5% corroded - sound	RC bulkheads now (1996) believed ineffective as containment barrier, It may not be possible to use hull for recovery purposes. RPV remains intact.
OK-150	1967	Tsivolka Fjord	a. Pontoon b. Outer container - steel box with welded lid c. Inner container - steel clad concrete box with welded lid	a. Unsound[2] b. 30% corroded - sound c. 20% corroded - sound	a. Unsound[2] b. 45% corroded - sound c. 35% corroded - weakened	Pontoon may not be used for recovery purposes. Containers remain intact.
421 1 PWR	1972	Novaya Zemlya Depression	a. Barge b. Concrete enclosure bulkheads c. RPV	a. Unsound[2] b. 25% degraded c. Little corroded - sound	a. Unsound[2] b. 45% degraded c. 5% corroded - sound	Barge may not be used for recovery purposes. Effectiveness of concrete corrosion barrier likely to be severely degraded. RPV remains intact.
601 2 LMR	1981	Stepovoy Fjord	a. Submarine hull b. Bitumen inside reactor compartment c. RPVs and steam generators	a. 10% corroded - sound b. 15% degraded - sound c. Sound	a. 25% corroded - sound b. 35% degraded - cracked c. Sound	RC and RPV remain intact. Effectiveness of bitumen barrier considered severely degraded. Otherwise no structural constraints on remedial actions.

[1] A defueled PWR is also included in this RC.

[2] Judged unsound on the basis that the pontoon and barge are known not to have been built for this specific purpose and are believed to have been old and at the end of serviceable life.

Assumptions: a. The concrete and bitumen have effective lifetimes of 100 years, with a linear decrease in effectiveness over this period.
b. Best corrosion rates (BCRs), as used in the Source Term Working Group models.
c. A survey of the dump sites will be necessary to assess the actual condition of the discarded objects prior to any decisions on remedial actions.

The LMRs, with their very slow corrosion rates and large mass of solidified Pb-Bi coolant, were treated differently and separately. Using this second model, it was shown that the release rate peaks at 8 $\mathrm{GBq \cdot a^{-1}}$ by the year 2105 but for most of the remaining long lifetime, release rates stay around 7×10^{-3} $\mathrm{GBq \cdot a^{-1}}$.

For this best estimate scenario, the calculations were extended until all components had corroded away; for the PWR units, this was by the year 4570, for the submarine LMRs, the reactors will take until the year 36 600 to disintegrate.

With another postulated scenario, glacial action in the year 3000, it was calculated that 4500 GBq would be released to the Kara Sea simultaneously from all the PWRs and their SNF, and 2100 GBq from the LMRs in that year.

The release rate data, for all the radionuclides, from all the units in the Kara Sea was then submitted to the Environmental Modelling and Radiological Assessment Working Group of the IASAP for dispersion and dose calculation modelling.

(3) The errors, uncertainties, and conservatism in the model were discussed, nothing amongst others, the assumption made that all activity is released for immediate dispersal to the sea. This makes the estimated release rates perhaps overly pessimistic, as much of the corroded material will slump to the bottom of containment structures, or be buried in surrounding sediments and therefore not passed into the fjords for circulation into the Kara Sea and beyond.

Corrosion rates were best estimates from the literature but still leave some uncertainties, especially the rate assigned to the solidified Pb-Bi of the LMRs.

Any future studies should attempt to obtain and use actual corrosion rates and onsite observations of barrier material effectiveness, from samples of the actual objects themselves, providing such investigation does not breach the containment barriers.

(4) The risks of a criticality incident in the dumped fuelled reactors following corrosion of the components or remedial action was considered. The possibility of such an incident was shown to be very low, but could not be ruled out. This should be borne in mind if remedial action is contemplated.

(5) With regard to the availability of design information for the submarine and icebreaker SGIs, a great deal is still unknown about the support structures for the reactor cores, thermal shields, and RPVs. While sufficient information is known about the disposal of the SNF from the icebreaker as to make this less of a concern, the lack of this information is most significant for criticality studies and remedial action evaluation of the submarine RPVs that contained SNF.

Sufficient information has been provided for the submarine LMR and icebreaker PWR cores to develop preliminary core models; however, information on the submarine PWR cores has been so limited as to require several assumptions with respect to their composition and layout. Less is know about the U-Al alloy fuels of the submarine PWRs than is known about the icebreaker fuel and its configuration within the RPV. Thus, inventories and release rates associated with the submarine PWRs, and specifically those containing SNF, have the greatest relative uncertainty and would likely require further and more detailed evaluation should remedial actions be considered.

(6) Future work should also include regular onsite investigation of the integrity of the objects, looking for any leaks which have opened earlier than anticipated by the model. Firstly, the condition of the welds which seal important leakage paths should be investigated. These would include the icebreaker SNF container lid weld, the CCR cap welds on the top of the PWRs, and the state of the concrete capping over unit 421. Hatches into RCs might provide access for small

remotely operated vehicles to look for radionuclides leaking from the SNF and for activation products corroding from the internal steels of the PWRs.

(7) Of course these disposal methods for damaged reactors would not be advocated now. From the available information however, this study has shown, that the containment methods employed by the dumping teams should be effective. Assuming that the Group has modelled the barrier strategy correctly and there is no disturbance to the objects, release rates will be very low, for most of the time these structures remain corroding away on the Kara Sea floor.

ABBREVIATIONS

BCR	best corrosion rate
CCR	control or compensation rods
ECT	emergency cooling tubes
EPR	emergency protection rods
IASAP	International Arctic Seas Assessment Project
IPPE	Institute of Physics and Power Engineering, Russian Federation
LMR	liquid metal reactor
PWR	pressurized water reactor
RC	reactor compartment
RPV	reactor pressure vessel
RRCKI	Russian Research Center "Kurchatov Institute"
SG	steam generator
SGI	steam generating installation
SIAE	State Institute for Applied Ecology, Russian Federation
SS	stainless steel
SNF	spent nuclear fuel
TFC	technical fuel channel

DEFINITIONS

control or compensation rods	A movable part of the reactor in the form of a rod with high neutron absorption cross-section, which itself affects reactivity and is used for reactor control.
coolant	Either water or Pb-Bi in this report, that is used to remove heat from the nuclear fuel.
criticality	The condition of fissile material, e.g.,nuclear fuel, when a fission chain reaction takes place. Subcritical means that the fission rate decreases. Supercritical means that the fission rate increases.
emergency cooling tubes (LMR only)	Tubes penetrating the RPV through which water may be passed to remove post-shutdown decay heat from the reactor core.
emergency protection rods	CCRs reserved for emergency return to a subcritical condition.
prompt criticality	The uncontrollable instantaneous increase in the fission rate of a fissile material.
reactivity	The parameter giving the deviation from criticality of a nuclear chain reacting medium.
reactor compartment	The space within a ship or submarine that is dedicated to the components of the SGI.
reactor pressure vessel	A cylindrical steel vessel containing the nuclear fuel, CCRs, EPRs, and ECTs (LMR only) which penetrate through the RPV head or lid.
slowing down power	The effectiveness a substance has to slow down (moderate) fast neutrons.
spent nuclear fuel	Irradiated fuel not intended for further use in reactors.
steam generating installation	A collective term for the entire reactor plant contained within the RC.
steam generator	A vessel in which the heat of fission is transferred from the coolant to water and subsequently converted into steam for propulsion and electric power.
technical fuel channel	An assembly of a specific number of fuel pins in the reactor of the icebreaker *Lenin*.

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