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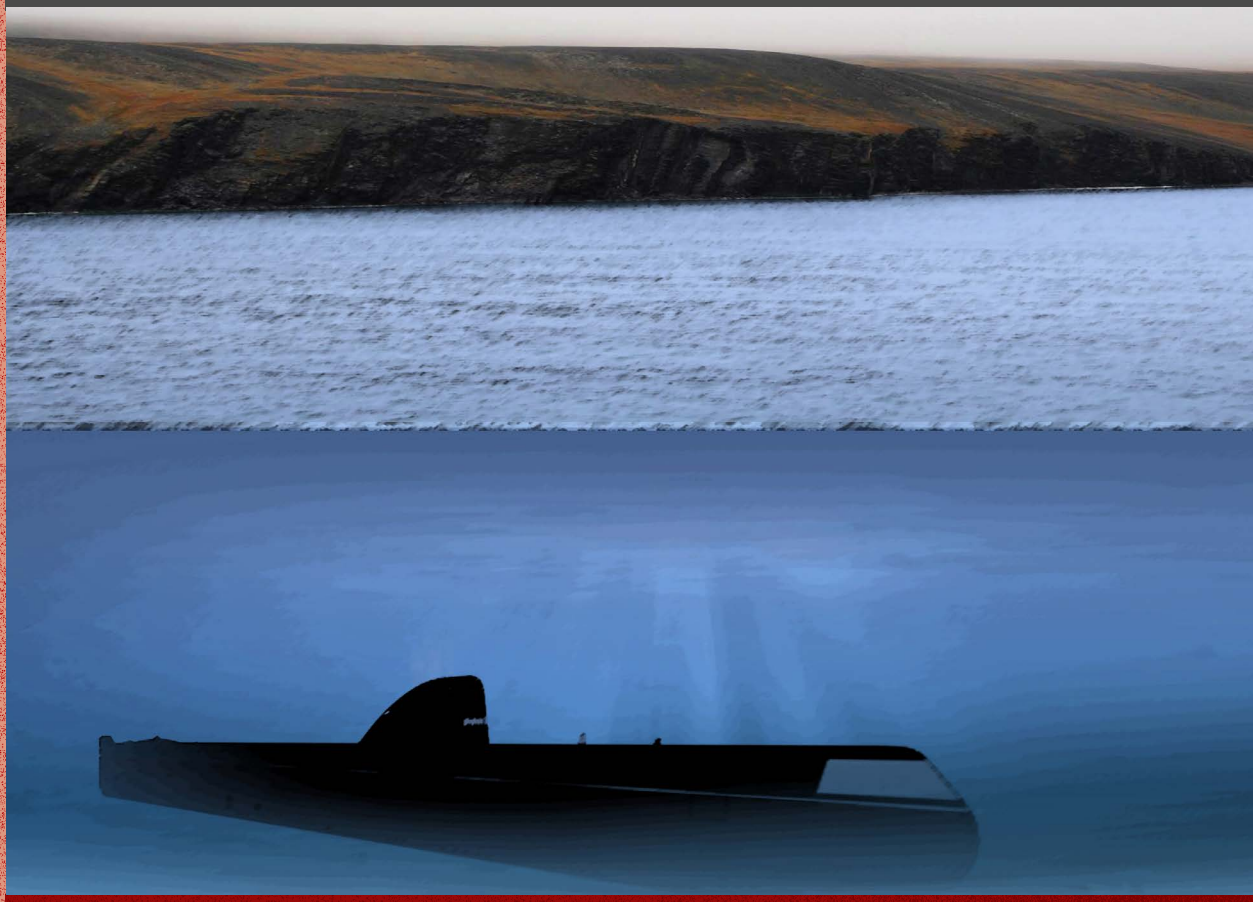
StrålevernRapport 2015:6



Inventory and source term evaluation of the dumped nuclear submarine K-27

Reference:
Hosseini A, Amundsen I, Brown J, Dowdall M, Standring W. Inventory and source term evaluation of the dumped nuclear submarine K-27.
StrålevernRapport 2015:6. Østerås: Statens strålevern, 2015.

Key words:
Dumped objects in the Arctic, Liquid Metal Reactor, LMR, Submarine K-27, Inventory and source term, Criticality scenarios, Recovery of K-27

Abstract:
An overview of existing and /available information regarding the Russian nuclear submarine K-27 is provided in the report. Furthermore, a detailed account of the inventory is given along with consideration of the various conditions under which a spontaneous chain reaction might occur. The amounts and forms of radionuclides that may be released to the environment under given accident scenarios are thereafter characterised.

Referanse:
Hosseini A, Amundsen I, Brown J, Dowdall M, Standring W. Inventory and source term evaluation of the dumped nuclear submarine K-27.
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Language: Norwgian.

Emneord:
Dumpede objekter i Arktisk, LMR, Kildeterm, Atomubåt K-27, Kritikalitet scenarier, Løfting av K-27

Resymé:
Denne rapporten gir en oversikt over eksisterende og / tilgjengelig informasjon om den russiske atomubåten K-27, redegjør for det totale radioaktive materialet ombord, samt vurderer de ulike forholdene som kan føre til en spontan kjedereaksjon. Rapporten karakteriserer mengder og former for radionuklider som slippes ut til miljøet under gitte ulykkesscenarier.

Head of project: Ali Hosseini
Approved:

Per Strand, director, Department of Nuclear Safety and Environmental Radioactivity

73 pages.
Published 2015-04-21.
Cover design: 07 Media.
Printed number: 150 (05-15)
Printed by 07 Media.
Coverphoto: Ali Hosseini

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ISSN 1891-5205 (online)
ISSN 0804-4910

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Inventory and source term evaluation of the dumped nuclear submarine K-27

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Statens strålevern
Norwegian Radiation Protection Authority
Østerås, 2015

Innhold/Contents

1 Introduction

1.1 Legal framework

The Arctic consists of territories of various nations, and as such has no overall, binding legal regime. Although the framework for environmental protection of the Arctic is ultimately constituted by national laws, the region is characterized by international co-operative policy and action. The Arctic Council (see UNEP, 2013) plays an important role in providing a forum within which participating countries can discuss subjects related to governance and Arctic issues. In recent years there has been much focus upon and developments within treaties and conventions related to the Arctic maritime environment, notable examples being UNCLOS - the United Nations Convention on the Law of the Sea (UNEP, 2013) and the Oslo Paris convention OSPAR which covers an area stretching from the east coast of Greenland to the western coast of Novaya Zemlya (OSPAR Commission 2000).

In the 1970s, the practice of dumping industrial wastes, including radioactive wastes, into the world's oceans became a subject of scrutiny by the international community. This engagement resulted in the development of a convention in 1972; the "Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter ", also referred to as the "London Convention "(IMO Secretariat, 1990).

The Convention, which has been in force since 1975, was one of the first global conventions to protect the marine environment from human activities. Its objective is to promote the effective control of all sources of marine pollution and to take all practicable steps to prevent pollution of the sea by the dumping of wastes and other materials. Both Norway and Russia (ratified as the USSR) were some of the first signatories to the convention. The IAEA (International Atomic Energy Agency) was recognized by the Contracting Parties of the London Convention as the competent international body in matters relating to radioactive waste disposal and radiation protection. The mandate of the IAEA under the London Convention was limited to the definition of the *de-minimis* levels of radioactivity for the purposes of the Convention.

Following the provision of technical expertise by the IAEA, in 1993 the Consultative Meeting of Contracting Parties to the Convection made the decision to prohibit the sea dumping of all types of radioactive waste. This resulted in the amendment of the London Convention accordingly. The Russian Federation did not accept the amendment to the Convention on the radioactive waste disposal at sea but agreed to act in the spirit of the Convention (IAEA, 1998).

In 1996, the "London Protocol" was agreed to further modernize the Convention and, eventually, replace it. The London Protocol entered into force on 24 March 2006. It is beyond the scope of this report to provide a view as to whether actions occurred to contravene any agreement or whether particular responsibilities might be incumbent upon signatory parties, the London Convention and Protocol have evolved over the last decades and the documents are, after all, ones requiring the involvement of legal practitioners for correct interpretation. Figure 1.1 shows an overview of the Protocol and Convention parties as of September 2014. The aforementioned documents, nonetheless, provide a context to the subject in hand and would be clearly interpreted as meaning that similar practices of dumping radioactive wastes as those described in this report and using modern day norms would not be acceptable in nowadays.

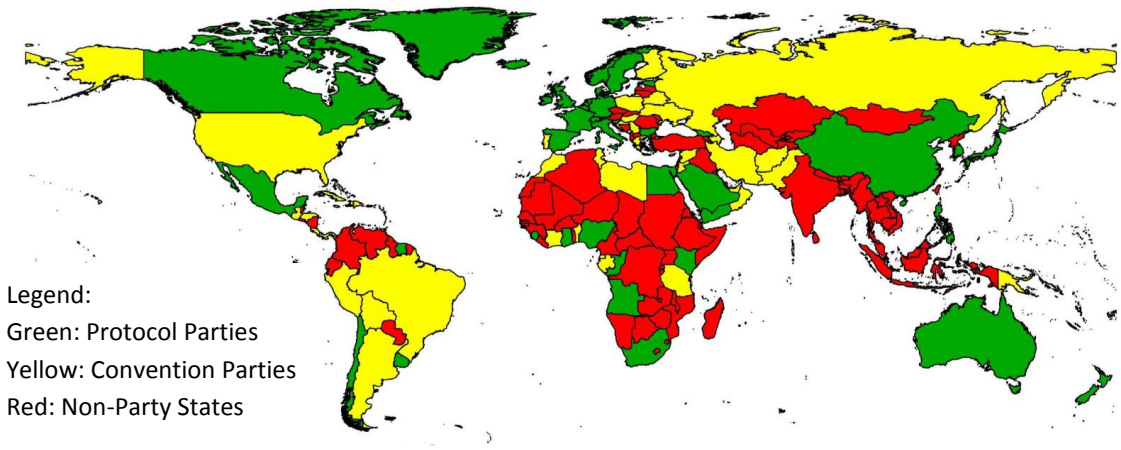


Figure 1.1: Parties to the London Convention and Protocol as of 15 September 2014. Taken from http://www.imo.org/OurWork/Environment/LCLP/Documents/Map%20of%20Parties%20Sept%202014.pdf (28.10/2014).

1.2 Earlier studies

There have been various activities initiated in the last 2 decades or so in relation to assessing the potential impacts following the dumping of radioactive wastes in northern marine environments. In the early 90's, emerging information regarding dumping activities of the former USSR in the shallow waters of the Arctic Seas raised widespread concern especially in countries with Arctic coastlines. To address these concerns, the IAEA launched an international study to assess the health and environmental consequences related to these dumping activities (IAEA, 1998). A similar response, but at a regional level, came in 1992 when the Joint Norwegian-Russian Commission on Environmental Protection established a group of experts to study radioactive contamination in the northern area in general and more specifically to develop a joint plan to investigate the issue of dumping nuclear objects and radioactive waste in the Barents and Kara Seas (JNREG, 1996). In April 1995, the Norwegian Government's Nuclear Action Plan was launched with a key objective to reduce the risk of serious nuclear accidents and radioactive pollution originating in Russia and prevent radioactive and fissile material from falling into the wrong hands.

All these initiatives have resulted in various joint activities (JNREG 2014) including:

- Acquiring information on the handling, storage, discharge and dumping of radioactive material in the northern areas.
- Investigating, through field work, the actual levels of radioactive contamination in the open Kara Sea and at the dumping sites.
- Locating dumped nuclear waste and ascertaining if any leakage of radioactive substances has taken place.
- Undertaking health and environmental risk and impact assessments

The present study falls under the last category of activities.

1.3 This study

Encouraged by the successful experience in defueling a similar submarine (Factory No. 900) and as a response to the increased concern regarding environmental safety, the recovery issue of the K-27 submarine

has become a frequent discussion point in many fora debating the theme of dumped nuclear waste in the Arctic. In this regard, the issue of remedial actions related to the dumped waste in the Arctic Seas was first discussed in IAEA's international study, IASAP (International Arctic Seas Assessment Project), which was launched in 1993 (IAEA, 1998).

It seems, on cursory inspection, therefore that many of issues related to the radiological implications associated with the dumping of K-27 may have already been covered in previous studies. IASAP provided great detail in relation to the inventories of dumped objects, including K-27, and even considered unexpected aspects such as the potential for criticality in dumped reactors and impacts on non-human biota. Nonetheless, there are numerous aspects that have not been covered earlier, most notably the radiological impact of accidents involving a criticality which are highly pertinent in the context of considering recovery operations. Furthermore, much new information has come to light in the more than 15 years that have passed since the last comprehensive studies on dumped objects such as K-27 and modelling capabilities have developed substantially in that time.

Therefore, there is an apparent need to address ongoing concerns regarding radionuclide releases from dumped nuclear waste in the Kara Sea and in particular those potentially occurring from submarine K-27 taking the opportunity to update assessments in relation to new information that has come to light.

With these considerations in mind, the objective of this study was to provide a new human health and environmental impact assessment for the dumped nuclear submarine K-27. The study is based on development of different hypothetical accident scenarios and evaluating possible associated consequences for human and the environment. The focus has been on the assessment of impact at near field (local) and intermediate field (regional) over various time-scales. The results of this study will be presented in 2 reports. The present report is the first of the two and will concentrate on providing an overview of the existing and available facts and information regarding the Russian nuclear submarine K-27, characterising the source term and considering the various conditions under which a spontaneous chain reaction might occur. The findings from the first report will be used in the second report where the focus will be on the modelling of radionuclide advection and dispersion in the environment and subsequent assessment of doses and associated consequences.

The present report is the product of a collaborative effort between the Norwegian Radiation Protection Authority and the Russian Energy Safety Analysis Centre of IBRAE RAN and the Russian Research Centre Kurchatov Institute (KI). While Chapters 5-9 are mainly based on the data and information provided by IBRAE and KI, the remaining chapters have been written making use of various, available sources of information and data. In particular for Chapter 4, background descriptions about the main subject of this study, namely, the submarine K-27 from both a historical and technical perspective, have been collated based on some non-verified sources of information – primarily published articles and internet reports. However this will have no ramifications for the output from this work, i.e. source term characterization, as this is based solely on the work of IBRAE and KI.

2 Dumped radioactive wastes in the Barents and Kara Seas

Regular dumping of liquid and solid radioactive waste in the Arctic was practiced by the former USSR and later by Russia from the early 1960s until the early 1990s. Such oceanic dumping of radioactive wastes, carried out by thirteen countries, occurred in the Atlantic and Pacific oceans during a similar period. Assessments of the total activity of liquid and solid radioactive waste dumped into the Barents and Kara Seas were first made public in a report published in1993 by the Russian Governmental Commission on Issues of At-sea Disposal. The report, commonly referred to as the White Book 1993 (WB 1993) was subsequently revised by the International Arctic Seas Assessment Project (IASAP) in 1993-1996 and then summarised in the IAEA technical document 'Inventory of radioactive waste disposal at sea' (IAEA, 1999a). More recently, the White Book 2000 (Sivintsev et al., 2005) reassessed the information originally published in the White Book (WB 1993) identifiying a number of inaccuracies and omissions. This section gives an overview of some of this updated information.

The total activity of liquid and solid radioactive waste dumped in the Barents and Kara Seas reported by the White Book 2000 is 38801.8 TBq (Table 2.1) (Sivintsev et al., 2005), equivalent to approximately 45% of the total activity of radioactive waste dumped in the worlds oceans. This amount is a slight increase on the previous estimate reported in AMAP (2002) of 36600 TBq. However it is likely that the true figure is somewhat higher as the White Book 2000 (Sivintsev et al., 2005) identified a number of hitherto undocumented dumping operations within the Barents and Kara Seas without being able to provide any information on associated activitites of the dumped waste.

Table 2.1.Total activity (TBq) at time of dumping of different types of radioactive waste dumped in the Arctic region by the Former USSR and Russia as reported in the White Book 2000 (Sivintsev et al., 2005).

Waste type	Total activity at time of dumping (TBq)	Percent of total activity
Reactor units with spent nuclear fuel	21781	56.1
Reactor units without spent nuclear fuel	14802	38.1
Reactor components	20.8	0.1
Low level solid waste	1240.2	3.2
Low level liquid waste	957.8	2.5
Total	38801.8	100

Detailed information on liquid radioactive waste dumped in the Arctic region can be found elsewhere (JNREG, 2014). An activity of 435.2 TBq was dumped into the Barents Sea inside five specifically allocated

areas (Figure 2.1); an additional 522.6 TBq was dumped as a result of operational accidents in the Barents, Kara and White Seas (Sivintsev et al., 2005). Low- and intermediate-level solid radioactive waste (SRW) was dumped principally at eight main locations covering the fjords east of Novaya Zemlya and the Novaya Zemlya trough in the open Kara Sea (Figure 2.2). By volume, the bulk of the SRW dumped consists of waste produced during the operation of the naval ships, icebreakers, and submarines containing nuclear reactors.

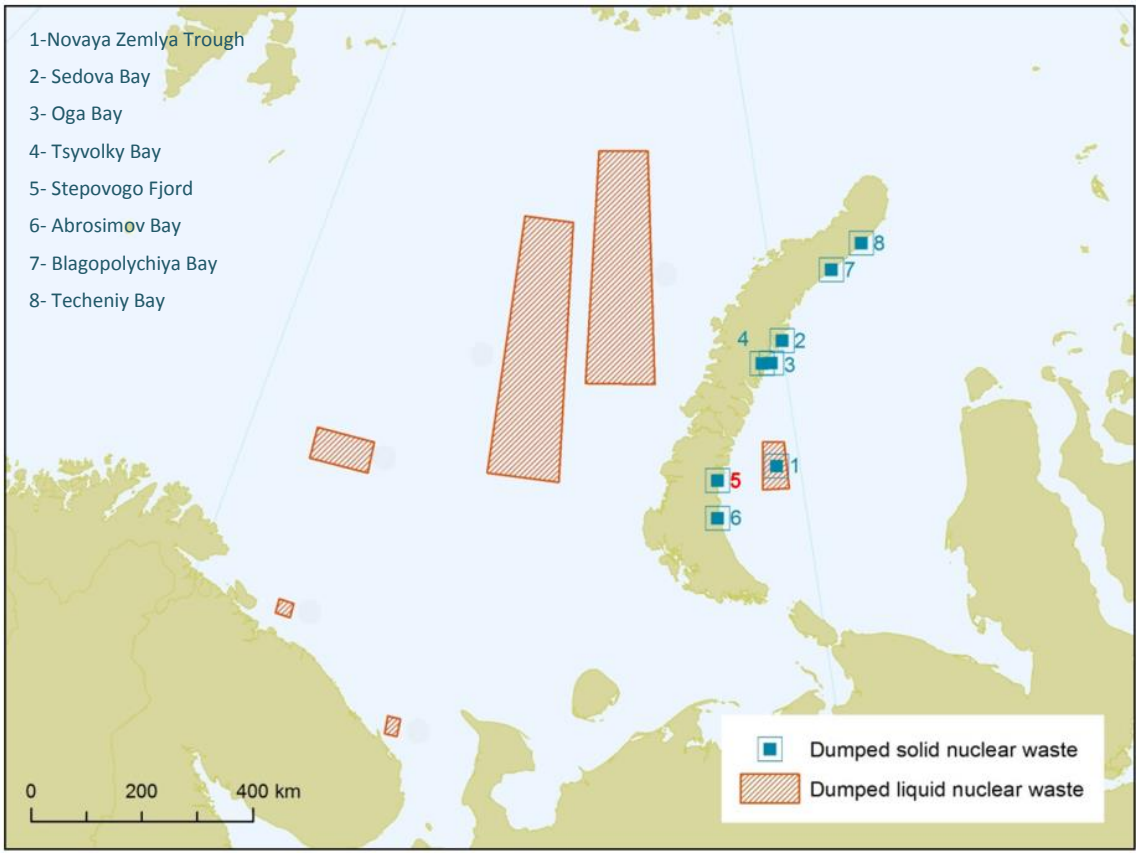


Figure 2.1. Main dumping areas in the Barents and Kara Seas as reported in the White Book 2000 (Sivintsev et al., 2005). Adapted from (JNREG, 2014).

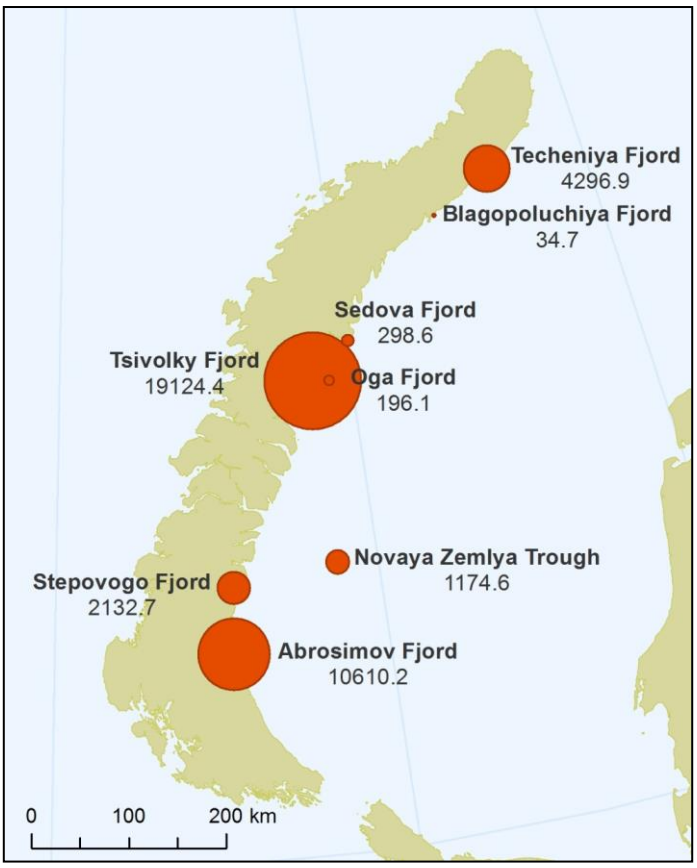


Figure 2.2. Revised estimates of the maximum total activity (TBq) of the dumped solid waste at the time of dumping as reported in the White Book 2000 (Sivintsev et al., 2005). Taken from (JNREG, 2014).

As a rule, low and intermediate-level SRW dumped in the Kara Sea was sealed in metal containers. Large waste objects were dumped separately or inside specially allocated ships, for example within a barge, lighter or tanker. The total estimated activity of low and intermediate-level SRW dumped in the Kara Sea amounts to 1240.2 TBq (Table 2.2), with a further 11.1 TBq having been dumped in the Barents Sea (Sivintsev et al., 2005).

Table 2.2. Low and intermediate-level solid radioactive waste dumped in the Barents and Kara Seas as reported in the White Book 2000 (Sivintsev et al., 2005).

Location	Years	No. of containers	No. of unpacked items	Total activity (TBq)		Remarks
				At time of dumping	In 2000	
Novaya Zemlya trough	1967-1991	4824	561	288.5	112.1	Operational waste and components from the "Lenin"
Sedova Fjord	1982-1984	1100	112	296.6	111.8	Operational waste
Oga Fjord	1968-1983	2190	101	191.2	64.0	Operational waste
Tsivolky Fjord	1964-	5242	166	229.4	53.3	Operational waste

	1978					
Stepovogo Fjord	1968-1975	1917	3	106.0	28.1	Operational waste
Abrosimov Fjord	1966-1981	646	-	55.8	16.7	Operational waste
Blagopoluchiya Fjord	1971-1972	992	2	27.7	7.7	Operational waste
Techeniya Fjord	1982-1988	194	28	33.9	15.9	Operational waste
NW of Kolguyev Island	1978	-	18	2.56	0.83	Various ship components
Barents Sea	1959	-	-	8.55	1.64	Barge with SRW
Total		17105	991	1240.21	412.07	

Reactors and reactor compartments, both with and without spent nuclear fuel (SNF), were also dumped in the Kara Sea with a total activity of about 36.6 PBq (Table 2.3). Unit No. 601 which comprises the reactors of the dumped submarine K-27 in Stepovogo Fjord had an associated activity, at the time of dumping in 1981, slightly in excess of 2 PBq.

In addition, a number of reactor components have been dumped at various locations with a total activity of 20.8 TBq. In Stepovogo Fjord the activity at time of dumping (i.e. 1966) associated with this category of dumped material amounted to 3.7 TBq and was associated with reactor lids. Further details are provided in the report from Joint Norwegian-Russian Expert Group (JNREG, 2014).

A more recent overview (Sarkisov et al., 2009) on radioactive wastes dumped in the Arctic (the Western Arctic Seas) provides the following list of the numbers and types of Russian dumped and / sunken objects: three non-defueled nuclear submarines; one non-defueled reactor (from nuclear submarine Nr 421); one container with a shielding assembly containing SNF (from the Lenin atomic icebreaker); five reactor compartments which belonged to submarines and icebreakers (two of which hold SNF); 19 ships with SRW on board; 735 radioactive constructions and units; and over 17000 containers with radioactive waste. The objects containing SNF are of the greatest potential radioecological hazard among all the radioactive waste dumped in the Arctic seas.

Table 2.3. Total activity at time of dumping in reactor units dumped in the Kara Sea as reported in the White Book 2000 (Sivintsev et al., 2005).

Location	Year of dumping	Unit No.	No. of reactors in unit		Total activity at time of dumping (TBq)
			Without SNF	With SNF	
Abrosimov Fjord	1965	No. 285	1	1	3968
	1965	No. 901	-	2	3644
	1965	No. 254	2	-	1839
	1966	No. 260	2	-	1097
Tsivolky Fjord	1967	OK-150	3	0.6[a]	18891
Novaya Zemlya trough	1972	No. 421	-	1	884
Stepovogo Fjord	1981	No. 601[b]	-	2	2018
Techeniya Fjord	1988	No. 538	2	-	4242
Total			10	6.6	36583

a – SNF was contained in a screening assembly not a reactor.
b – this is the reactor unit associated with sunken submarine K-27

3 Description of Stepovogo Fjord and the joint Russian-Norwegian expeditions

The K-27 lies on the eastern side of Novaya Zemlya in the Stepovogo Fjord. Novaya Zemlya is an archipelago in the Arctic Ocean which separates the Barents Sea from the Kara Sea. It has been a sensitive military area since the Cold War when it was used as a major nuclear weapon test site by the USSR. In addition to being a testing site, Novaya Zemlya has also been used as a dumping site for nuclear objects and radioactive wastes by the USSR. Over the years there have been several expeditions to the dumping sites in the Kara Sea including Stepovogo Fjord and these sites have also been the subject of various studies where possible releases and dispersion of radioactivity have been investigated. In this chapter we first provide a description of the physical attributes of Stepovogo Fjord (especially those considered relevant to the subsequent advection dispersion modelling that will form the subject of the next report) and then briefly explore earlier expeditions to the area.

The term 'fjord' is not actually correct for the main dumping locations along the Novaya Zemlya coast: Abrosimov Fjord, Stepovogo Fjord and Tsivolky Fjord (Figure 3.1). A fjord is commonly defined as a narrow deep valley, carved into a coast line. A fjord usually has an underwater barrier or sill at the entrance so that mixing between the inner part of the fjord and coastal waters off-shore is limited. It is only Stepovogo Fjord that complies, at least to some extent, with these characteristics. However, Stepovogo Fjord, when compared to a 'real' fjord, is quite shallow and the sill is actually marginal. The other two locations are more appropriately described as 'bays' with broad entrances and rather shallow depths.

3.1 A general description of the area

The width of Stepovogo Fjord is mostly less than 2 km, whereas the longitudinal length is about 10 km (Figure 3.2). The maximum depth is about 60 m but the barrier or sill (at the entrance) is only 21 m below the surface. The IAEA TecDoc-1075 (IAEA, 1999b) states that the inner parts of the fjord are quite well aerated, despite the presence of a sill, and that stratification is a pronounced feature in late summer: "*A dissolved oxygen profile measured in the inner part of the fjord indicates good ventilation of bottom waters. In September, the inner basin was relatively strongly stratified as compared to the outer part of the fjord and the nearby coastal area".* The document also provides some valuable information on the hydrography of the fjord: *"In the inner basin, the water temperature went from 3.7°C at the surface to -1.7°C near the bottom while salinity ranged from about 20 psu (practical salinity unit) in the top 10 m to 34.8 psu at the bottom. In the outer part of the fjord temperature varied very little with depth, being in the range of 2.6-3.4°C. Salinities slowly increased from 17 psu to 28 psu in the central part of the outer basin, while at the mouth of the fjord they were constant at 18 psu down to 15-20 m, then gradually increased to 24 psu near the bottom. The fresh water component is thus more important in the outer part of the fjord. The bottom sediments in this area also indicate strong flushing of the bottom*" (IAEA, 1999b).

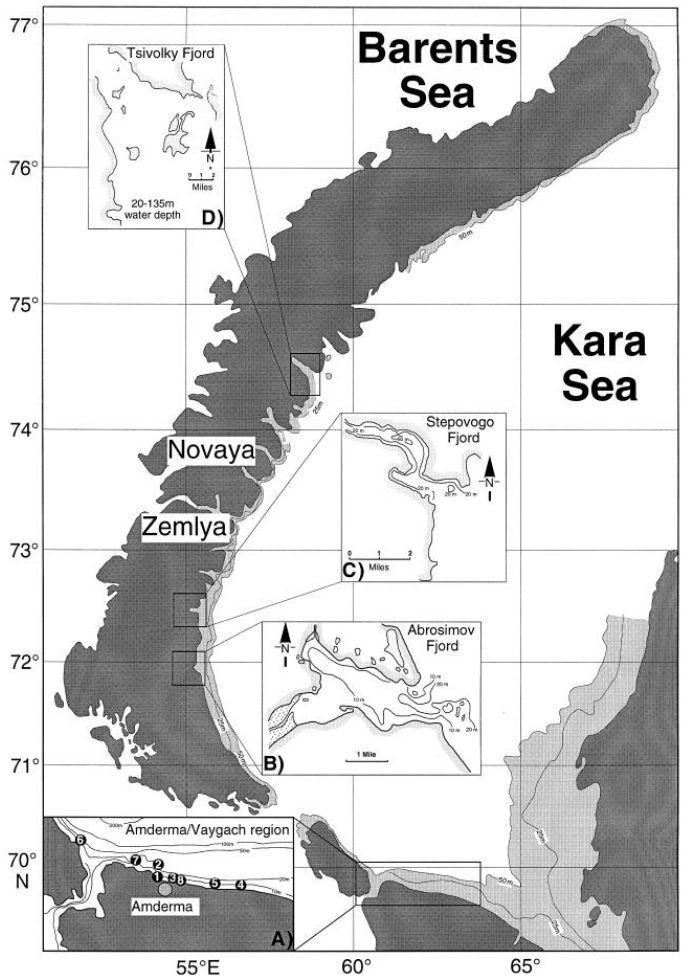


Figure 3.1. Previously investigated and modelled dump site locations along the Novaya Zemlya coast. Taken from Dethleff et al. (2000).

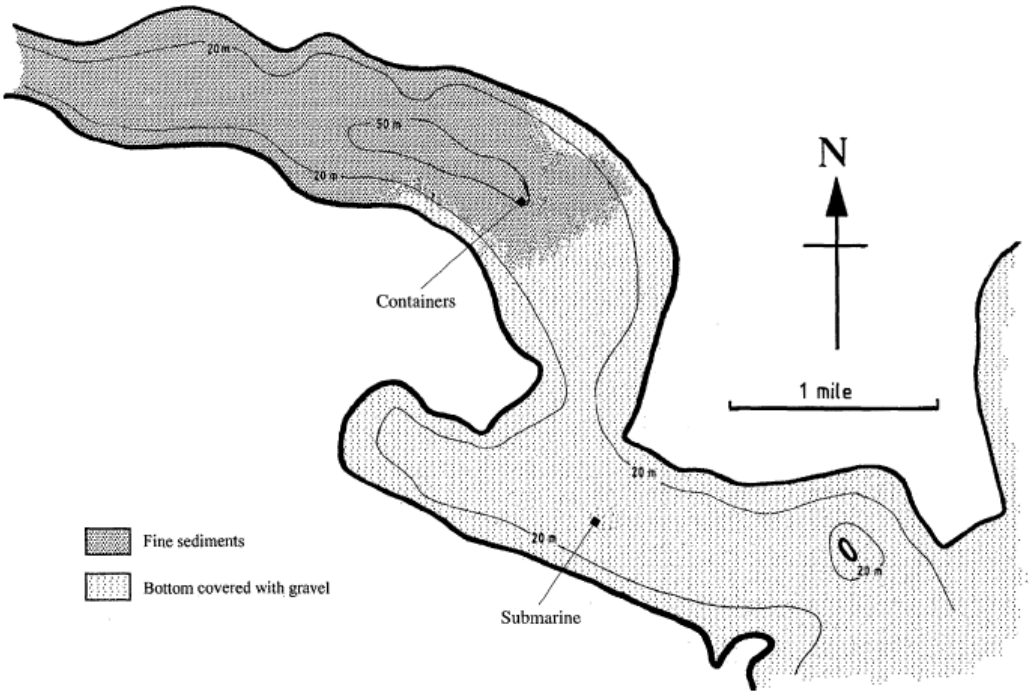


Figure 3.2: Shape, depth structure and sediment composition of Stepovogo Fjord.

Details on the topography and the hydrography of Stepovogo fjord can be found in the reports published by the Joint Russian Norwegian Expert Group following expeditions to dump sites in the Kara Sea (JNREG 1994; Føyn and Nikitin, 1994) and from the IAEA TecDoc-1075 (IAEA, 1999b). More recent information on the hydrography can be found in Stepanets et al. (2006).

3.1.1 Flushing time scenarios

In order to provide basic estimates for the retention times of radioactivity in fjord systems it is useful to have information on flushing times. This can be achieved through the analysis of idealized flushing time scenarios.

The simulations performed by Harms (1997) do not refer to a release of a specific radionuclide but investigate the flushing times of Abrosimov Bay and Stepovogo Fjord (Harms, 1997). Wind was considered to be the most important driving force for the fjord circulation. An average moderate wind speed of 5 m/s was chosen while the models were run until a stationary state of circulation was achieved. The resulting circulation patterns must thus be regarded as average situations, valid for time scales from several days to months. Higher wind speeds (storms etc.) do of course occur in actuality but on much shorter time scales. For the broad Abrosimov Bay four different wind directions were applied in order to investigate differences in circulation and flushing. For the narrow Stepovogo Fjord two longitudinal wind directions were investigated with winds blowing along the fjord.

The simulations were carried out in a barotropic mode (i.e. with homogeneous density) without tides. Ice was also omitted. The stationary solution in circulation was used to drive a simplified version of the dispersion model (i.e. only advection / diffusion without scavenging and radioactive decay). The inner part of the bay/ fjord was initialized with an arbitrary tracer concentration of 100. The outer parts and the open sea (i.e. the Kara Sea) were set to zero. The amount of flushed water was given in % (i.e. 100% = totally flushed). If the value reaches more than 90%, the bay/fjord was regarded as flushed. In that case, the average tracer concentrations in the inner part have decreased to less than 10.

The scenario results suggest that the flushing times for Abrosimov Bay and Stepovogo Fjord are quite short. Abrosimov Bay (Figure 3.3, left panel) can be flushed for almost all wind directions and moderate wind

speeds (5 m/s) within three or four months. This is due to the broad bay-type structure of Abrosimov Bay which allows all wind directions to act more or less equally efficiently.

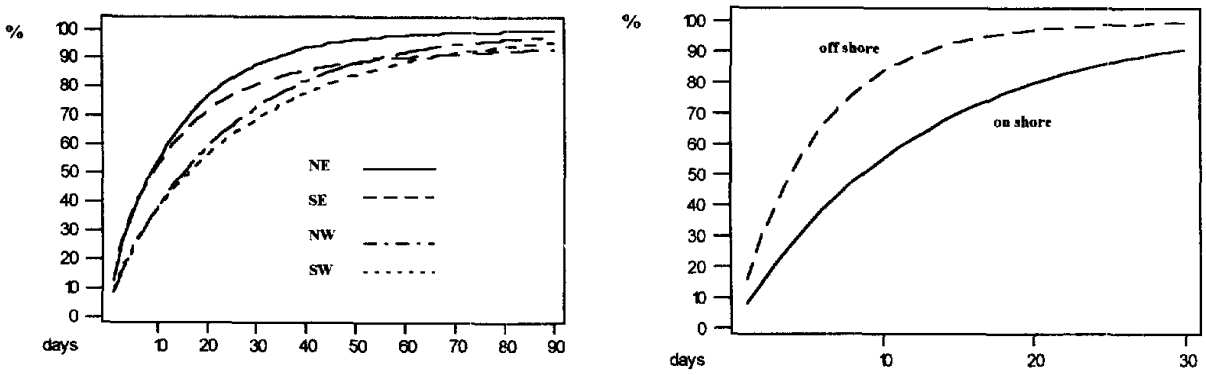


Figure 3.3. Flushing times for Abrosimov Bay (left) and Stepovogo Fjord (right), Harms (1997).

In spite of the fact that there is a sill at the entrance, which forms an obstacle for the circulation, the small Stepovogo Fjord can be flushed much more rapidly (Figure 3.3 right panel). Moderate on- and offshore winds (5 m/s) blowing in the direction of the fjord are most effective for flushing because of the longer fetch (the distance wind travels over water before meeting an obstacle). Typically for these scenarios an offshore wind flushes the Stepovogo Fjord by 90% after 15 days and an onshore wind flushes the fjord after 30 days. The difference is due to a more effective outflow at the surface and an enhanced mixing in the inner part with the off-shore circulation type. The sill at the opening causes a closed vertical circulation cell in the inner part of the fjord which enhances vertical mixing leading to a faster flushing.

Koziy et al. (1998) also performed model scenario experiments for Novaya Zemlya dump sites and calculated very similar flushing times for the bays or fjords. The applied model types and the set-up in these studies are very similar to the model studies performed by (Harms, 1997; Baxter et al., 1998; Harms and Povinec, 1999). This concerns in particular the hydrodynamic model part and its forcing which is essentially the same. However Koziy et al. (1998) included an ice cover during winter and also applied tides. The calculated range of flushing times given by Koziy et al. (1998) is between 10 days in summer and 3.5 months in winter for the small Stepovogo Fjord. The lower bound of this range (summer season) agrees well with the estimations of Harms (1997); Baxter et al. (1998) and Harms and Povinec (1999). The winter flushing times from Koziy et al. (1998) are higher than values estimated by Harms (1997); Baxter et al. (1998) and Harms and Povinec (1999) because the latter did not include an ice cover that reduces transfer of momentum from the wind to the water column in case of thick ice or fast ice in the fjord. For the broad Tsivolky Bay (see Figure 3.1), the flushing times given by Koziy et al. (1998) are in the range of 20 days up to 6 months, depending on the wind direction and season. These values can be compared to the Abrosimov Bay results given in Figure 3.3, left panel. Given that Tsivolky Bay is larger than Abrosimov Bay, the numbers also agree quite well.

However new information regarding the bathymetry of Stepovogo Fjord indicates the existence of 2 sills in the fjord; a shallow one (sill depth 10 - 15 m) in the inner part and a slightly deeper one (20 – 25 m) at the entrance. The K-27wreck is located between these 2 sills at a depth of approx. 30 m.

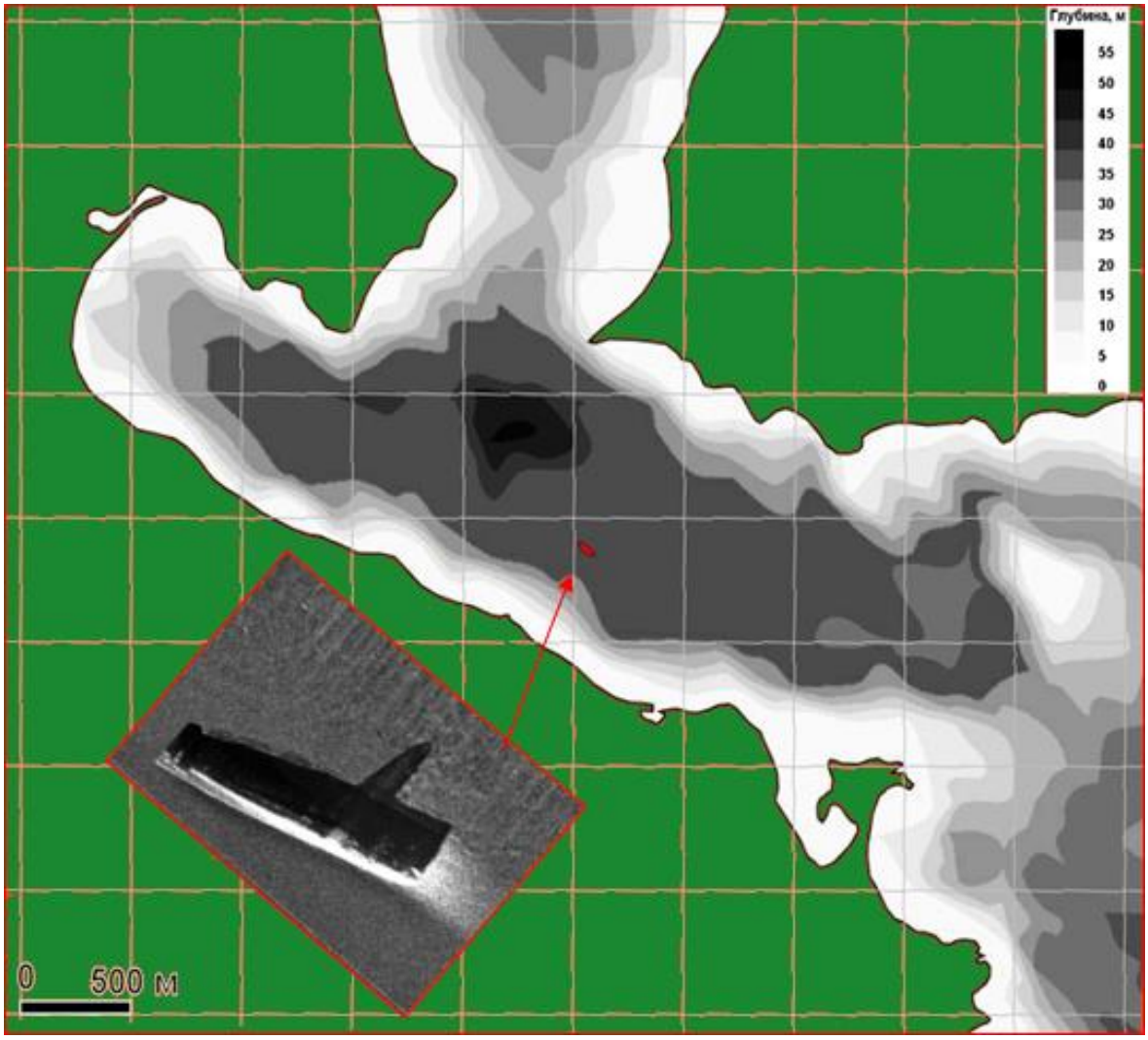


Figure 3.4. Bathymetry of Stepovogo Fjord. Taken from Kiknadze (2013).

Considering the real bathymetric situation of the Stepovogo Fjord, the earlier findings given above, based on considering just one sill, are still relevant. The sill at the entrance may hamper slightly the outflow of contaminated waters at the bottom of the water column, however, the sill is only 5 - 10 m shallower than the depth of the wreck. It also has to be considered that above the sill the fjord is quite broad which clearly facilitates the outflow of contaminated surface waters.

The effect of the sills is only important for the very near field (i.e. the fjord itself) and has to be considered only on short time scales. Looking at regional mean radionuclide concentrations, e.g. averaged over months or seasons, the influence of the sills becomes less important.

3.2 Previous radioecological investigations in Stepovogo Fjord

The following text summarises the findings from the Joint Norwegian Russian Expert Group (JNREG) in relation to their expeditions to dumping sites off the coast Novaya Zemlya in the early 1990s and has been taken verbatim from JNREG (2014).

"In 1993 and 1994, the nuclear submarine K-27 was located in the outer part of Stepovogo Fjord and despite poor visibility underwater, visual inspections of K-27 with an ROV identified an opening in the submarine hull. In 1994, numerous metal containers were detected and observed in the inner part of Stepovogo Fjord at a

depth of around 50 m. Some of the metal containers were semi-buried in bottom sediments while others were observed with holes in their outer casings.

In the investigations in 1993 and 1994, similar activity concentrations in surface seawater were observed within Stepovogo Fjord as compared to the open Kara Sea in 1992 and 1993, but activity concentrations of Cs-137 and Sr-90 were elevated in bottom water from the inner part of Stepovogo Fjord. In general, activity concentrations in sediments from the outer part of Stepovogo Fjord were uniform and similar to those observed in the open Kara Sea. However, one gross surface sediment sample collected close to the hull of the nuclear submarine K-27 showed a Cs-137 activity concentration (203 Bq/kg) more than one order of magnitude higher than other samples. Activity concentrations of Cs-137, Sr-90, Cs-134 and Co-60 in surface sediments from the inner part of Stepovogo Fjord were highly variable but elevated in some samples taken close to dumped objects. Trends in vertical sediment profiles showed typically higher activity concentrations of Cs-137 and Pu-239,240 in the top 5 cm of sediment cores with decreasing activity concentrations down core. Based on Pb-210 measurements, the sedimentation rate at one station in the inner part of Stepovogo Fjord was estimated to be 1.3 mm/year. The presence of radioactive particles in sediments from Stepovogo Fjord was confirmed by autoradiography and scanning electron microscopy (SEM)."

3.2.1 Conclusions from previous investigations

In conclusion JNREG (2014) notes the following :

"*The Joint Norwegian-Russian Expert Group report (JRNEG, 1996) on the investigations in the 1990s concluded that elevated levels of radionuclides in sediments collected close to dumped objects demonstrated that leakages had occurred and that the highest activity concentrations of Cs-137, Co-60, Sr-90, and Pu-239,240 were observed in sediments collected close to dumped containers in Stepovogo Fjord and Abrosimov Fjord. Although indications of possible leakage from the reactors in K-27 were observed in outer part of Stepovogo Fjord in 1993, this was not confirmed by subsequent sampling in 1994.*

The observed activity concentrations of anthopogenic radionuclides in seawater, sediments and biota in the open Kara sea could be attributed to global fallout from the atmospheric nuclear weapons tests and marine transport of authorised discharges from European reprocessing plants, fallout from the Chernobyl accident and outflow from the Ob and Yenisey rivers. No additional contribution from dumped radioactive waste was observed, though the report recommended a regular monitoring programme due to the potential for leakage from the range of dumped waste in the future.

Follow up investigations in the period 2002 to 2004 found no evidence of additional leakages from dumped objects with SNF including the reactors in K-27, but noted that sediment activity concentrations in Stepovogo Fjord as well as other dump sites had decreased appreciably compared to the 1990s."

3.3 Conclusions from the JNREG 2012 expedition and summarized information

Finally, in the context of the latest expedition to Stepovogo Fjord, and other sites in 2012, the following observations were made and conclusions drawn by JNREG (2014):

"*The nuclear submarine K-27 was observed lying upright and clear of bottom sediments at a depth of around 30 m in the outer part of Stepovogo Fjord and no obvious corrosion damage of the outer hull was observed (see Figure 3.5). Activity concentrations in seawater and sediment in the outer part of Stepovogo Fjord do not indicate that any leakage has occurred from the reactor units of K-27 since it was sunk in 1981. Due to the limited number of containers that were inspected during the investigation, it is not possible to provide an overall assessment of the status of such dumped objects in the inner part of Stepovogo Fjord*."

Tower (front view)

Tower (back view) and rails

Stern hatch

Bow anchor

Figure 3.5. The nuclear submarine K-27 in the outer part of the Stepovogo Fjord. Taken from JNREG (2014).

"Activity concentrations of Sr-90 and in particular Cs-137 in 2012 remain elevated in bottom water from the inner part of Stepovogo Fjord as a result of leakages from dumped containers and subsequent remobilisation of Cs-137 from contaminated sediments. However, activity concentrations of Cs-137 in seawater from all sampling stations in Stepovogo Fjord in 2012 were lower than observed in previous investigations in 1993 and 1994. Sediment cores from the inner part of Stepovogo Fjord revealed that peak activity concentrations of Cs-137 in sediments occurred at deeper sediment depths than was observed in previous investigations in 1993 and 1994. As activity concentrations in surface sediments from the inner part of Stepovogo Fjord remain elevated, it is likely that sediments will continue to act in the future as a source of Cs-137 to bottom water in this part of the fjord. Any future releases from dumped containers lying in deep water in the inner part of Stepovogo Fjord are likely to have limited impacts on the wider marine environment due to the reduced mixing and flushing of bottom water with inflowing Kara Sea water. However, future releases from dumped containers in shallower water in the inner part of Stepovogo Fjord may be more readily transported to the outer part of Stepovogo Fjord and further afield."

Activity concentrations in all biota samples collected in Stepovogo Fjord in 2012 were low and comparable to values reported in the Norwegian Sea and Barents Sea.

A summary of selected information from the 2012 expedition and early studies are presented in Tables 3.1 and 3.2. All data from JNREG (2014).

Table 3.1. Activity concentrations in seawater (<1 µm) from JNREG investigations in Stepovogo Fjord.

		Entrance to Stepovogo Fjord		Outer part of Stepovogo Fjord			Inner part of Stepovogo Fjord		
		1993	2012	1993	1994	2012	1993	1994	2012
Cs-137	S	5.1	1.6 ± 0.2	5.2	8.6	1.6 ± 0.3	5.8	7.4	1.5 ± 0.3
(Bq/m³)	B	6.9	1.7 ± 0.3	8.2	9.5	1.8 ± 0.3	26.5	16.7	15.4 ± 1.2
Cs-134	S	0.12	-	0.14	0.07	-	0.13	0.06	
(Bq/m³)	B	0.16	-	0.21	0.16	-	0.21	0.09	
Sr-90	S	5.5	5.2 ± 1.8	4.9	2.9 - 3.1	3.9 ± 1.3	5.2	2.4 - 2.9	4.3 ± 1.3
(Bq/m³)	B	4.2	3.4 ± 1.6	4.0	4.2 - 4.7	2.5 ± 1.1	24.3	3.9 - 5.0	5.8 ± 1.5
Pu-239,240	S	2.9	1.4 ± 0.7	2.3	3.6	2.3 ± 0.6	1.9	4.7	2.2 ± 0.5
(mBq/m³)	B	18.0	2.4 ± 0.7	2.3	4.5	3.2 ± 0.7	6.0	5.9	4.8 ± 0.8
Am-241	S	-	1.5 ± 0.6	1.5	-	1.3 ± 0.5	2.0	-	1.8 ± 0.5
(mBq/m³)	B	5.3	1.8 ± 0.6	7.7	-	1.1 ± 0.4	-	-	1.5 ± 0.5

S - surface water; B - bottom water.

Table 3.2. Activity concentrations in surface sediments from previous investigations in Stepovogo Fjord.

	Outer part of Stepovogo Fjord				Inner part of Stepovogo Fjord			
	1993[1]	1994[1]	2004	2012	1993[1]	1994[1]	2004	2012
Cs-137 (Bq/kg d.w.)	6.7 - 203	2 - 1670	4.4 - 19.2	4.5 – 11.3	34.3 - 289	4 - 109000	19.5 - 268	17.2 – 58.9
Cs-134 (Bq/kg d.w.)	0.17 - 0.23	0.3 - 2.4	-	-	0.32 - 0.47	0.3 - 39.8	-	-
Co-60 (Bq/kg d.w.)	0.29 - 1.2	<0.1 - 6	-	-	1.6 - 15.2	<0.3 - 3150	-	< 0.5
Sr-90 (Bq/kg d.w.)	0.4	0.4 - 6	-	0.3 – 0.9	2 - 4	1 - 310	-	0.4 – 1.6
Pu-238 (Bq/kg d.w.)	-	0.04 - 0.09	<7.8 - 30.6	0.01 - 0.03	0.03 - 0.06	0.11 - 6.5	<7.9 - 354	0.008 – 0.04
Pu-239,240 (Bq/kg d.w.)	0.37	<0.1 - 6	0.27 - 0.62	0.31 – 0.67	0.73 - 0.79	0.6 - 2.8	0.29 - 1.1	0.36 – 0.76
Am-241 (Bq/kg d.w.)	0.07	-	-	0.14 – 0.31	-	1.1 - 2.0	-	0.14 – 0.36

1 - includes samples taken close to dumped objects with ROV.

4 K-27: Submarine with liquid metal reactors

4.1 Background – K-27 construction and characteristics

During October of 1955, the Council of Ministers of the USSR adopted a resolution regarding the development of nuclear submarines driven by liquid metal cooled (LMC) reactors. The project, denoted as Project 645, commenced during 1957. The original design brief envisaged a 1500 ton interceptor submarine that could travel submerged at 40 kts. To achieve this, a smaller hull and higher power engines were required. A.B.Petrov of the design bureau Malakhit (SKB-143) devised a compact reactor using a liquid metal coolant system. Early designs included lower levels of reactor shielding than would normally be found on conventional nuclear submarines, totally automated engine plants, a small crew (consisting entirely of officers) and a lightweight titanium alloy hull. Later, the designs were altered to increase the displacement of the submarine to 2300 metric tonnes with an increase in the total number of internal compartments. The first LMC submarine to be constructed was denoted as K-27. K-27 was equipped with two VT-1 type LMC reactors (Sullivan et al., 2002).

K-27 was designed by SKB-143 in Leningrad and construction took place at the Severodvinsk shipyards. Intended as a test bed for liquid metal cooled reactors in submarines, the submarine contained 10 compartments (Figure 4.1).

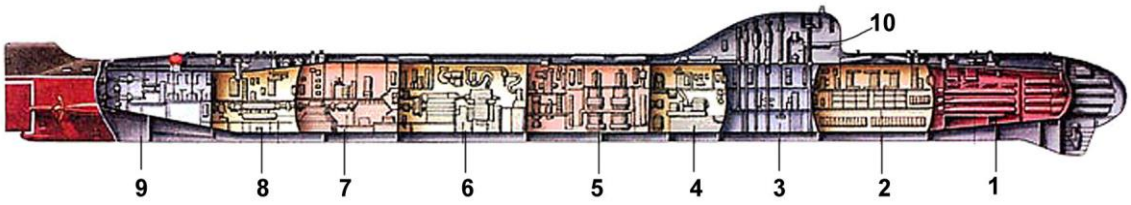


Figure 4.1. Schematic map of the nuclear submarine K-27 compartments. 1- torpedo compartment; 2 – accumulator compartment; 3 – central house; 4 – diesel compartment; 5 – reactor compartment; 6 – turbine compartment; 7 – electromechanical compartment; 8 – dwelling unit; 9 – stem compartment; 10 – deckhouse. Published on the web site http://vnmazurenko.blogspot.com.au/2012/09/27_5.html

In addition to size reduction due to the design of the liquid metal cooled reactor, gains were made by the use of a 400 Hz electrical system as well as the use of smaller battery capacity than would be typical for conventionally cooled reactors. K-27 had an overall displacement of 3410 m^3, was 109.8 m long, 8.3 m wide and had a draft of 5.8 m. The maximum operational depth of the submarine was 300 m with a speed of 29 knots when submerged. The submarine could remain at sea for 50 days and had a full force of 104 crew. The reactors of the K-27 were of the type VT-RM-1 producing 73 MW(t) each for a total thermal power of 146 MW. The submarine had 8 torpedo tubes and a load of 20 torpedoes. The reactors contained 2732 fuel elements in the form of rods, the fuel height being 862 mm with an overall core diameter of 762 mm (Somov, 2011). Fuel rod pellets were made of uranium-beryllium alloy with the rods having a diameter of 11 mm. The rods were housed in El-852 steel tubes (outside diameter 12 mm, wall thickness 0.4 mm) and the space between the rod and tube was internally filled with magnesium. The rods were situated within a triangular core lattice (pitch 13.6 mm) and were surrounded by lead-bismuth coolant with a volume fraction of 31%. The shields for the reactors were 75 mm of beryllium and 200 mm of steel on the sides, 100 mm of beryllium and 295 mm of steel on the bottom with reflectors of 228 mm on the top (Somov, 2011).

The keel of K-27 was laid down in June of 1958 and the submarine was launched on the 1st of April 1962, being commissioned on the 30th of October 1963. Between April and June of 1964 the submarine had its first active service cruise in the mid-Atlantic and between June and August of 1965 the submarine had its second active service cruise, subsequently undergoing medium level repairs during 1966. During the first fuel cycle the reactors had been operated in accordance with pre-set conditions for about 97% of the time (Somov, 2011).

The reactors were refuelled in February and March of 1967. The spent fuel elements of the first cycle were replaced with fresh fuel at this point. Employing data generated during the first cycle as well as information from land based test reactors, the fuel load was increased from 62.5 kg ^{235}U to 91.5 kg ^{235}U and the control rods were replaced with three rod assemblies instead of the single rods initially employed. Each reactor core contained 17 control rod channels (Somov, 2011). During the refuelling, the coolant of the primary circuit was drained but the coolant in the reactor remained. Problems were noted in relation to the steam generators of the port reactor, primarily leakage which resulted in the formation of particulate matter in the coolant which necessitated cleaning of the coolant to avoid blocking of coolant flow in the reactor.

4.2 Accident

On the 24th of May 1968 the submarine was on naval exercises and cruising submerged at maximum power. Prior to this cruise, it had been noted that the coolant required cleaning but this was not conducted. While testing of reactor parameters was underway, power meters of the left board reactor began to oscillate before dropping to 7%. While increasing the reactor power from 83% and the rotation of the main primary circuit pump, an automatic control rod had withdrawn from the reactor for a period of 2 or 3 seconds and 60 – 90 seconds after this the observed power drop to 7% occurred. This power drop indicated damage to the fuel of the reactor. This damage had been caused by localized over heating due to release of poison that had been accumulating in the primary circuit during previous operations as well as the formation of particulate oxides in the coolant due to the previously mentioned steam generating leaks. Based on measurements it was concluded that the damage was located in the upper part of the left board reactor core close to scram rod (safety rod) number 2 and shim rod[1] number 9 at points where elevated coolant temperatures had been observed (Somov, 2011). The Scram thermocouple was located elsewhere and therefore the overheating did not produce any scram (i.e. emergency shutdown) as the situation progressed. Prior to the accident the reactor power was 18700 MW.h and the reactivity margin was approximately 14%(i.e. 14 % above critical state). The margin of reactivity decreased during the accident by some 9% as the fuel was released.

Estimates indicate that approximately 20% of the fuel of the left board reactor was damaged (IAEA, 1997). Radioactive material was released to the primary circuit and subsequently to the reactor compartment. The radiation level in this room was above 100 Roentgen/hour (> 870 mSv/h) and contamination spread to other compartments. The submarine surfaced in an attempt to reduce radiation levels by ventilation even though this increased the return to base time. It was suggested that the submarine remain outside the base but due to considerations of conditions on-board the submarine proceeded directly to the quay. Of the whole crew who were on-board at the time of the accident, 12 received doses in excess of 600 Roentgen (> 5 Sv) and were hospitalized. Five men died, 4 of radiation and one of suffocation in a gas mask. Later figures indicate that perhaps 9 men died of radiation exposure (Oelgaard, 1996).

[1] A control rod used for making occasional coarse adjustment in reactivity of a nuclear reactor.

K-27 remained at Gremikha until 1973. During its stay at Gremikha the operational starboard reactor functioned as a test bed for liquid metal cooled reactors. Steam was pumped into the left board reactor from the starboard reactor to keep the coolant liquid. At the end of this period, the steam supply was removed and both reactors were allowed to cool resulting in solidification of the coolant in both. During this period the submarine functioned as a test bed for decontamination procedures.

4.3 Preparation of the submarine for dumping

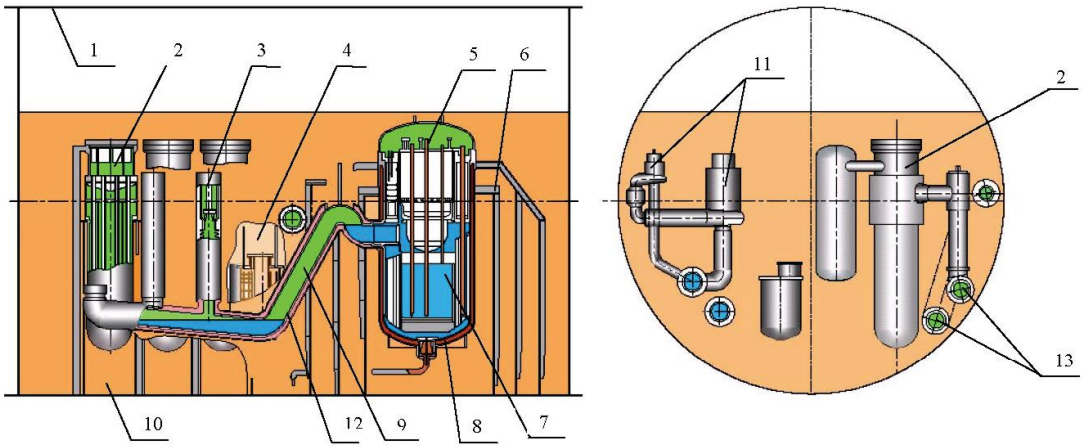
At the end of 1973, the submarine was towed from Gremikha to Severodvinsk. Later, a series of operations were conducted there in order to prepare the vessel for disposal. All frozen lead-bismuth was removed from the starboard reactor primary circuit which had been undamaged in the accident except for the reactor pressure vessel itself. Frozen lead-bismuth in the port reactor primary circuit was left intact. The control rods were fixed within the cores, raising mechanisms were removed and any of the rods extruding above the reactor lids were cut away. The channels that contained control or compensation rods were then filled with furfural[2] (see below) in areas that were not filled with frozen lead-bismuth (see Figure 4.2). Those channels containing the emergency protection rods were completely filled with furfural and then sealed by welding them shut with steel covers on the upper surfaces of the reactor lids. Cylindrical channels around the pressure vessel formed by the addition of cylindrical steel sheets were filled with furfural. The steam generators of the damaged port reactor were pumped full of furfural via the secondary coolant circuit.

Materials which could be used as a solidification agent to prevent contact of seawater with the nuclear fuel of the reactor and subsequent leaching and release of radioactive materials to the environment have to possess special properties. Such material needed to be of low viscosity, homogenous, with a working time of 6-9 hours and a solidification time of less than 15 days. The solid material needed to be resistant to radiation and seawater with good adhesive capabilities, be of low porosity and non-gas generating with a long lifetime and such that criticality would be avoided. The chosen material was a thermosetting polymer based on furfural known originally as Conservant F / “Furfurol F” as sort of a "trade name" (Heiser et al., 1996). The iterations differed in relation to the hardener used in the agent (acids or amines or both), mineral additives such as graphite and quartz and various epoxy resins used as additives. The five different iterations of the material were designated F.1, F.2, F.3, F.4 and F.5. The F.3 variant was used in reactors removed from vessels which had not been defueled – 4 reactors in total dumped between 1965 and 1972. Furfural F.3 was also used in the disposal of the No. 2 reactor of the icebreaker Lenin (Heiser et al., 1996).

The reactors of K-27 were filled with the Furfural F.4 variant (Heiser et al., 1996). The best available information indicates that the F.4 variant consisted of furfural (28.7%), styrole (3.38%), aniline hydrochloride (3.72%), pulverized quartz (62.0%), ethyl alchohol (1.49 %) and cadmium nitrate (0.74%), all values in parentheses defining the percentage by weight of the component (Somov, 2011).

Figure 4.2 shows the scheme of filling reactors, equipment and reactor compartment. For all indicated spaces, apart from emergency protection rods (EPRs), the same preservative composition was used.

[2] Furfural is an organic compound derived from a variety of agricultural byproducts such as oat hulls, corn cobs, wheat bran and sugar cane bagasse.



1 – Strong hull;
2 – steam generator;
3 – valve and bellows expansion joint;
4 – pump tank bay;
5 – reactor;
6 – lead-water shielding tank;
7 – solidifying Pb-Bi-alloy;
8 – furfural with cadmium nitrate;
9 - bitumen with orthoboric acid;
10 – bitumen;
11 – pumps;
12 – heat insulation;
13 – pipelines.

Figure 4.2. Schematic overview of measures taken at different parts of the reactor compartmetn of K-27 prior to its sinking.

To fill the casings of the emergency protection rods (EPRs) the composition of preservative was slightly modified to increase its capacity to absorb neutrons. Orthoboric acid was used as a neutron absorber instead of cadmium nitrate. The amount of orthoboric acid added was calculated allowing for the projected size of a “shrink hole” which could potentially have formed in the core owing to alloy freezing. The absorbing capacity of orthoboric acid was considered to provide conditions of nuclear safety in case of possible water leakage into the shrink hole formed in the preservative.

The internal spaces of other items of equipment of the reactor facility and the voids of the steam generators, pumps and pipelines of the primary circuit and the entire reactor compartment (up to the level of 300 mm above the reactor heads) were filled with bitumen.

Internal voids in the following equipments were also filled with bitumen (Somov, 2011):

- Lead-water shielding tanks LWST (excluding bow sealed volume);
- Geared-turbine TR-2 oil tanks;
- Pumps, steam generators and pipelines of the primary circuit;
- Steam generators of the secondary circuit;
- Buffer capacities of alloy leakage return pumps.

Boric acid was added to bitumen (11% by weight) for filling the portside steam generators of the secondary circuit in order to mitigate against the release of part of the fuel composition to the primary circuit. Calculations showed that the presence of water or other hydrogen-containing substances in the steam generator pipes and accumulation of fuel to a level exceeding 10% of the amount held in the core, might form a critical mass. The calculated amount of boric acid addition was considered to ensure nuclear safety.

Preservative based on furfural flowed incrementally (50–70 kg at a time) from a special loading basin (under a pressure of 3 kg/cm^2) through conduits and nipples welded into the receiving voids in advance. There was a pause after filling of each portion in order to cool the mixture that had heated up owing to an exothermic solidification process. Upon completion of the filling process, the nipples used for filling were kept unpressurized in preparation for the subsequent filling of the compartment with bitumen. The total amount of volume (of equipment and pipelines) filled with preservative based on furfural was ~ 2 m^3 (Somov, 2011).

Bitumen was added at a temperature of 140–170 ºC through hoses running directly from bitumen carriers lifted to the dock-wall. Upon filling of voids with bitumen, conduits used for filling were kept unpressurized. The "mothballing" process required a total amount of 270 m^3 of bitumen BND 90/120 and 5.8 tons of furfural.

Upon filling, all CPS casings were cut, then sealed, and removable core seals were welded. Covers of 2 cm thickness were welded on top of the reactor lids and the space between the reactor lids and the new covers was then filled with bitumen. The void spaces contained within the steam generators, circulation pumps and the reactor compartment itself were then filled with bitumen to a level of 24 cm above the welded covers over the reactor lids.

In 1981 the submarine was towed to Stepovogo Bay at Novaya Zemlya and sunk at a depth of 30 m. A summarised chronology of submarine K-27 is provided in Table 4.1.

Table 4.1. Chronology of the nuclear submarine K-27.

15 Jun 1958:	Laid down
01 Apr 1962:	Launched
30 Oct 1963:	Commissioned
21 Apr – 12 Jun 1964 (52 days):	1st active service voyage in the area of central Atlantic
29 Jun-30 Aug 1965 (60 days):	2nd active service voyage
Autumn 1966:	Medium level repairs
Feb-Mar 1967:	Refueling operations
24 May 1968:	Accident
Jul 1968 to 1970:	Reactor shut-down, coolant frozen
1970 –1973:	Heating the coolant and rising reactor to power levels up to 20% (at Gremikha)
1 February 1979:	Officially decommissioned
Summer 1981:	Reactor compartment was mothballed at "Zvezdochka", Severodvinsk
Sep 1981:	Towage and sinking

4.4 Radiological situation before dumping

Upon completion of all work prior to the disposal of the submarine, gamma dose rates in all compartments of the vessel did not exceed 0.2 μR/s (1.7 x 10^{-3} μSv/s). Neutron dose rates did not exceed natural background levels. Gamma dose rates in the vicinity of the biological shielding on the surface of the light casing, were less than 60 and 65 μR/s (0.52 and 0.57 μSv/s) for the port and starboard sides, respectively. These rates decreased rapidly with distance along the hull. Residual beta contamination of the external and internal

surfaces of K-27 did not exceed 20 particles/cm^2min. No contamination with alpha emitting nuclides was observed (Somov, 2011).

Due to low burnup of cores of the campaign II reactors (~ 5.3% and 5.7 5% of the total design power capacity), the total activity of the dumped submarine was relatively small. According to an evaluation conducted by the Russian Institute of Physics and Power Engineering (IPPE) the total activity of the submarine at the time of dumping, was about 37 kCi. More details regarding the estimation of the K-27 inventory will be presented in the next chapter.

4.5 Remediation plans for K-27 (Russian plans)

More than three decades after dumping of the K-27submarine, the issue of its potential recovery has recently received growing attention both on the Russian side and from international community. In 2011 the Norwegian Radiation Protection Authority (NRPA) and the IAEA Contact Expert Group (CEG) organised a workshop which became an important international gathering to specifically address the issue of dumped objects with spent nuclear fuel in the Arctic Seas. The workshop gave particular attention to challenges related to salvaging of K-27.

In a later event, on 24th of January, 2013, Moscow hosted an interdepartmental seminar to discuss scientific and technical problems and their potential solutions in association with lifting and subsequent handling of the nuclear submarine K-27 (NES, 2013).In the seminar, the Russian Research Centre Kurchatov Institute presented the following stages as the essential steps in the process of handling of the submarine (in the given order):

- Conducting a comprehensive engineering and radiological survey (making use of divers)
- Studying alternative handling scenarios for the submarine:
    - Scenario I: take no action (safety analysis)
    - Scenario II: Entomb the submarine in situ (development and implementation of a design for in-situ isolation)
    - Scenario III: Raise the submarine from the seabed and transport it to land (development and implementation of a design for raising and transportation, disposal, SNF and radioactive waste management)
- Technical and economic assessment of the alternative scenarios (choosing the optimum scenario)
- Implementation of the optimum scenario

According to A. Kramarenko, Director of Science and Research Institute of Rescue and Underwater Technologies of the Russian Naval Academy, three basic options exist for lifting the K-27 submarine off the seabed: using a high-capacity lift vessel, steel lifting pontoons, or inflatable lifting modules (NES, 2013). All these options are technically feasible. An evaluation of these recovery options based on consideration of different factors such as risk, time, cost, availability of hardware and technologies, identified the first alternative as the best option. This option is coherent with international practise of lifting such items (NES, 2013).

Nonetheless, for the time being, there isn't any definitive planned action with regards to recovery of the K-27 submarine from the Russian side.

5 Inventory

5.1 Previous estimates

Fully documented information regarding the inventory of the K-27 submarine has already been provided by two earlier projects, IASAP (Sivintsev, 1993 and 1994, Yefimov, 1994) and ISTC-101 (Lavkovsky, 1999). Calculations performed as part of the IASAP and ISTC-101 projects for evaluation of the activity in the K-27 reactors included the estimation of neutron fields and induced activity in the materials of the reactor and the reactor compartment and the calculation of the activity of fission products and actinides in the cores containing irradiated nuclear fuel. The ISTC-101 calculations took into account the actual operational histories of the reactors according to data from the operator entities and the designers of the reactors (OKBM and OKB ”Gidropress”). Evaluation of neutron fields in the reactor and reactor compartment was conducted using the codes ANISN (Code package ANISN. User manual, Kurchatov INE, 1981) and DOT-III (Code package DOT-III. User manual, Kurchatov INE, 1983) which solve the radiation transport equation using a discrete coordinate method allowing for anisotropic neutron scattering in one-dimensional and two-dimensional geometries, respectively, with division of the neutron power spectrum into 12 groups. Calculations of induced activity were performed by application of SAM (multigroup application for calculation of specific activity of materials) using a library of 233 activation reactions for 37 chemical elements over a range of neutron energies from thermal to 14.7 MeV. Calculation of the accumulation of fission products and actinides in the cores was performed according to Lavkovsky (1999) and Sivintsev et al. (2005). The input conditions employed in the two projects were as outlined in Table 5.1.

Table 5.1. Input conditions employed for IASAP and ISTC assessments of inventory.

Reactor	Operational period				MW·day		Date of dumping	
	Start		End					
	IASAP	ISTC	IASAP	ISTC	IASAP	ISTC	IASAP	ISTC
Port	12.1962	09.1963	05.1968	05.1968	1580	6459	09.1981	1981
Starboard	12.1962	09.1963	06.1968	12.1971	1580	6459		

Note: According to ISTC-101 data K-27 reactors contain spent nuclear fuel of the second load. Its power production is 917 MW·day in each reactor; according to IASAP (IAEA, 1997) data the power production is 875 MW·day. Fuel reloading occurred in March 1967.

Results of both IASAP and ISTC projects in relation to fission products, actinides and activation products are displayed in Tables 5.2, 5.3 and 5.4, respectively. Table 5.5 shows a comparison of inventories estimated by both IASAP and ISTC projects.

Table 5.2. Comparison of fission product inventory estimates (Bq) as produced by both the ISTC and IASAP projects. It should be noted that calculated totals include isotopes not listed discretely in the table. An asterix denotes inclusion of the activity of short lived daughters.

	Project	^{147}Pm	^{137}Cs *	^{90}Sr *	^{134}Cs	^{85}Kr	^{125}Sb	^{151}Sm	TOTAL
Port	IASAP	1.45E+13	8.5E+13	7.4E+13	1.10E+11	8.15E+12	8E+11	2.05E+12	3.18E+14
	ISTC	-	2.43E+14	2.48E+14	1.82E+11	-	-	-	5.3E+14
Star	IASAP	1.45E+13	8.5E+13	7.4E+13	1.1 E+11	8.15E+12	8E+11	2.05E+12	3.18E+14
	ISTC	-	2.43E+14	2.48E+14	1.82E+11	-	-	-	5.3E+14

Table 5.3. Comparison of actinide inventory estimates (Bq) as produced by both the ISTC and IASAP projects. It should be noted that calculated totals include actinides not listed discretely in the table.

	Project	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{241}Am	TOTAL
Port	IASAP	4.73E+09	1.695E+11	3.25E+09	2.55E+10	4.185E+08	6.35E+11
	ISTC	-	2.465E+11	7.60E+09	1.02E+11	3.17E+09	8.45E+11
Star	IASAP	4.73E+09	1.695E+11	3.25E+09	2.55E+10	4.185E+08	6.35E+11
	ISTC	-	2.465E+11	7.60E+09	1.02E+11	3.17E+09	8.45E+11

Table 5.4. Comparison of activation product inventory estimates (Bq) as produced by both the ISTC and IASAP projects. It should be noted that calculated totals include isotopes not listed discretely in the table.

	Project	^{55}Fe	^{60}Co	^{59}Ni	^{63}Ni	TOTAL
Port	IASAP	-	2.55E+14	7.80E+11	7.10E+13	3.655E+14
	ISTC	8.60E+13	3.29E+14	-	4.35E+13	4.78E+14
Star	IASAP	-	2.55E+14	7.80E+11	7.10E+13	3.655E+14
	ISTC	8.60E+13	3.29E+14	-	4.35E+13	4.78E+14

Table 5.5. Comparison of inventories estimated by both IASAP and ISTC projects. Numbers in parentheses indicate inventories in 1994 for the IASAP project and 2000 for the ISTC project. Numbers not in parentheses indicate estimates at the time of dumping.

Project	Actinides TBq	Fission products TBq	Activation products TBq	Total TBq
IASAP	1.27 (1.22)	635 (375)	731 (239)	1367 (615)
ISTC	1.69 (1.53)	1060 (676)	956 (153)	2018 (831)

The earlier estimates of the inventories of the reactors (as shown in the above tables) are clearly quite different. A summary of various estimations is given in Table 5.6 (totals for both reactors):

Table 5.6. A summary of various estimations of total inventory of the K-27 submarine.

Estimated total inventory (PBq)	year	Performed by	Reference
37 kCi (1.37)	1981	IPPE	-
54.5 kCi (2.01)	1981	ISTC-101	Lavkovsky, 1999
22.5 kCi (0.83)	1998	ISTC-101	Lavkovsky, 1999
36.9 kCi (1.37)	1981	IASAP (IAEA)	Yefimov, 1994
16.6 kCi (0.61)	1994	IASAP (IAEA)	Yefimov, 1994
200 kCi (7.4)	1981		WB,1993

A new evaluation of the residual activities of the submarine conducted in 2013 by the Russian Research Centre Kurchatov Institute is shown in Table 5.7. The results indicate that the total residual activity is not exceeding 10 kCi (0.37 PBq).

Table 5.7. New estimates of the residual activities for nuclear submarine K-27 (as of the year 2013).

	Main radionuclides currently present in the reactor core	**Activity (TBq)**
Fission products	$^{137}Cs+^{137m}Ba$, $^{90}Sr+^{90}Y$	270
Control rods	$^{152,154}Eu$	40
Reactor shell constructions	^{63}Ni,^{60}Co	11
Actinides	$^{238,239,240,241}Pu$, ^{241}Am	4.6
Tritium	^{3}H	34

The White Book-2000 (Sivintsev et al., 2005) analyses the sources of discrepancies between the results of the IASAP project and the ISTC project, concluding that the main causes are significant differences in the operational time and power production as well as the use of 1-dimensional and 2-dimensional analytical models in determination of neutron emission. The potential also exists for discrepancies to have been caused by failures to include various aspects related to the design and layout of the reactors.

5.2 Current estimates

For the purpose of conducting an updated assessment of the inventories of the reactors in this work, a new analysis was conducted using modern codes and up-to-date libraries of constants of interaction cross-sections, release, accumulation and radioactive decay. This most up-to-date analysis was undertaken by IBRAE under contract to the NRPA. For assessing the inventory of the reactor cores, it was assumed that both reactors were, for the period prior to 1968, in operation constantly for 120 days with the port reactor operating at 9.3% of its power capacity and the starboard reactor operating at 9.45% of its power capacity. In addition, it was assumed that the starboard reactor operated for 4 days during 1973 at 18% of its power capacity. In assessing the activity of the reactor compartment it was assumed that both reactors operated continuously for 1250 days between 1963 and 1967 with the port and starboard reactors operating at 13% and 13.5% of their power capacities, respectively. The accumulation of long-lived activity during reactor

operation is determined by the decay of fission products, actinides, tritium and the activation of construction materials due to neutron induced reactions.

A 3-dimensional analytical model of the reactor was created in accordance with the design of the reactor and related facilities. Monte-Carlo N-particle Transport code (MCNP-4) was used for estimation of neutron flux density, fission product density, and activation products. Calculations of the activity of generated fission products and induced activity in the construction materials (steel of the reactor vessel and CPS rods), as well as other active isotopes (tritium, europium isotopes), was conducted using modern libraries of evaluated cross sections for radionuclides, taking account of decay chains and radioactive decay properties (Golashvili et al., 1995; Dubinin et al., 2008).

After a cooling period of ~40 years, cobalt is the main source of induced gamma-ray activity in the steel assemblies of the reactor and its compartment. The mass content of cobalt in various steel grades varies over a wide range from 0.019% to 0.081%, based on information provided in (Karabash et al., 1980) where the authors investigated about 20 various steel grades using atomic absorption spectrometry with respect to cobalt content. The average mass content of cobalt in analyzed steels was 0.056%. The estimates of induced activities in the steel constructions of the submarine reactors are in agreement with the findings from Karabash et al. (1980). It is worth noting that the same approach, codes and provision of constants were used for evaluation of nuclear safety in relation to works conducted on reactor retrieval from the submarine factory number 900 (see Section 6.1), preparation for unloading and SNF unloading from the reactor. Generally, activities calculated by employing this approach, showed satisfactory agreement with measurements regularly performed by the nuclear safety service of SevRAO at all stages of works on the submarine factory number 900 (see Section 6.1).

Detailed information concerning the inventories of fission products, actinides and other radionuclides (e.g. neutron activation products) associated with port and starboard reactors, reactor components and associated materials (e.g. cladding), where appropriate, were performed by IPPE and is provided in the Appendix.

A summary of the calculated inventories for various categories of radionuclides associated with port and starboard reactors and for the submarine in its entirety is presented in Table 5.8. Not surprisingly, in view of the similarity in approaches and calculation made in the two analyses, there is good agreement between the estimates made by IPPE (Table 5.8) and those made by the Russian Research Centre Kurchatov Institute (Table 5.7).

Table 5.8. Summary of various potential activities present at the nuclear submarine K-27 (Bq), as performed by IPPE.

		Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
Portside reactor	Fission products	1.56E+14	1.49E+14	1.38E+14	1.05E+14	6.59E+13
	Actinides	1.58E+12	1.54E+12	1.40E+12	1.24E+12	1.01E+12
	Tritium	4.44E+12	3.96E+12	3.00E+12	1.70E+12	5.55E+11
	Activation products	1.37E+13	1.34E+13	1.25E+13	1.13E+13	9.88E+12
	Europium in CPS rods	1.48E+13	1.32E+13	1.01E+13	5.92E+12	2.07E+12
	SUM	1.91E+14	1.81E+14	1.65E+14	1.25E+14	7.94E+13
Starboard reactor	Fission products	1.67E+14	1.59E+14	1.48E+14	1.12E+14	7.07E+13
	Actinides	1.69E+12	1.65E+12	1.50E+12	1.32E+12	1.09E+12
	Tritium	4.74E+12	4.26E+12	3.22E+12	1.81E+12	5.92E+11
	Activation products	1.41E+13	1.38E+13	1.31E+13	1.17E+13	1.02E+13
	Europium in CPS rods	1.58E+13	1.41E+13	1.07E+13	6.33E+12	2.18E+12
	SUM	2.03E+14	1.93E+14	1.76E+14	1.33E+14	8.47E+13
TOTAL, Bq		3.94E+14	3.74E+14	3.41E+14	2.58E+14	1.64E+14
TOTAL, Ci		*1.06E+04*	*1.01E+04*	*9.22E+03*	*6.99E+03*	*4.43E+03*

6 Possible consequences of long-term stay under water

The reactors of K-27 belong to an intermediate spectrum reactor class (i.e. using intermediate neutrons without any need for moderators). Their cores contain quite a large amount of highly enriched uranium. In the case of water or other hydrogen containing substances penetrating into the core, the neutron spectrum degrades and the potential for occurrence of a criticality increases. In the event of penetration of a sufficient amount of water, conditions leading to a self-sustaining nuclear reaction may arise in the core. In order to prevent different parts of the reactors coming into contact with seawater, in addition to physical barriers, protective material such as bitumen and furfural were used during the disposal operation (see section 4.3). At the time of dumping, there was no reliable information regarding stability periods for the main protective barriers (preservatives, fuel cladding, absorbers in Pb-Bi alloy, construction and elements of the primary circuit, joint welds etc.) under conditions of long term contact with the seawater. The quality of containment technology used and its long-term integrity has been considered in this chapter.

To model releases from the dumped objects, IASAP (IAEA, 1997) considered different scenarios of which one was related to the release rates produced by corrosion processes (Scenario A). However, from the outset of the work it was realized that a detailed and accurate prediction of radionuclide release rates was impossible due to the limited availability of relevant and reliable information. In order to acquire data it was decided to make use of stated assumptions based on consideration of the worst case, i.e., fastest rates of release. In the following we summarize the findings of IASAP with respect to K-27:

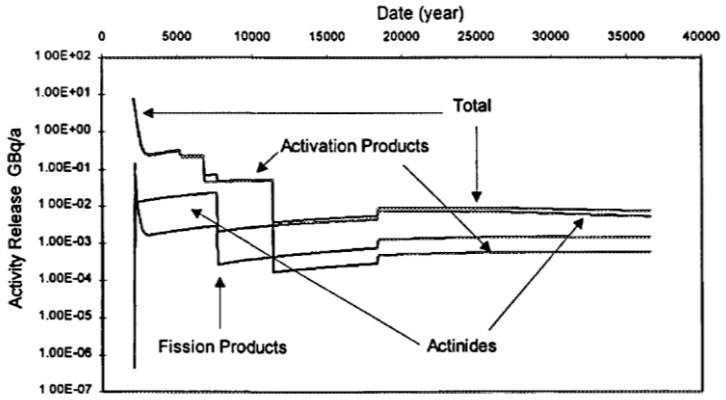


Figure 6.1. Total activity release for K-27 at Stepovogo Fjord according to IASAP Scenario A. Taken from IAEA, 1997.

Overall Scenario A release rates are shown in Figure 6.1. No release of active material is expected before 2105. At that time, fission products and actinides in the corroded left board steam generator were predicted to appear at a rate which is less than 0.1 MBq/a. This will be augmented by a contribution of about 8 GBq/a due to the corrosion of the outer surfaces of the activated reactor pressure vessels.

Approximately 200 years after dumping, fission products and actinides from both reactor cores are predicted to join the release stream and the total release rate was predicted to be of the order of 5 GBq/a. By the year 3000, the rise in release rate from the SNF and thermal shields of the reactors exceeds the fall due to decay, and the release rate rises until the year 5200 when the emergency cooling tubes merge to form an annulus. The release rate then varies as the shields external to the emergency cooling tubes corrode away by the year 6800 and the left board steam generator loses all its SNF by the year 7500. By this stage, the release rate has fallen to 0.07 GBq/a.

During the years 7500 to 11400 the release rate remains steady until the thermal shields corrode away by the end of this period. Release rates then continue at 4 MBq/a from the SNF and the Pb-Bi coolant, as the reflector starts to corrode. Due to complete corrosion of the reflector by the year 18400, a small rise in releases occurs and by that time the water will have reached the outside of the core. Release then continues at a rate of 7 MBq/a until all the SNF has corroded away in the year 36580.

It has been stated by the IASAP experts that the reliability of the parameter values (Best Corrosion Rates) used in their study was dependent upon the type of material considered and the availability of data at the time of the study. Data availability has been a major challenge especially for materials such as the Pb-Bi eutectic alloy and U fuels in their various alloy and oxide ceramic forms. Unfortunately, access to reliable data especially with regards to long-term material properties remains a formidable challenge.

A means of testing model predictions regarding the integrity of protective barriers and the potential for radionuclide release, at least in the short term, involves the acquisition of environmental samples. It was therefore somewhat disconcerting when a sediment sample from the sea-floor close to the K-27 submarine taken by the Joint Norwegian–Russian Expert Group in 1993 showed the presence of the europium isotopes ^{152}Eu and ^{154}Eu and high levels of ^{137}Cs (JNREG, 1996). It was suggested at that time that these radionuclides might have originated from the reactors of K-27 in contrast to the results from model runs made by IASAP which indicated an initial release much later into the future. Observations made during later expeditions (see Section 3.2 of this report) have not led to a reconfirmation of earlier cruise findings drawing instead the conclusion that leakages from K-27 have probably not occurred (indicating that enhanced levels of contamination have most probably originated from other sources in inner Stepovogo Fjord).

The following sections attempt to collate various relevant data on the long-term behaviour of protective barrier materials used to prevent water leakage into the K-27 reactor cores. Here the issue of corrosion and integrity of coolant and barriers in the K-27 has been reconsidered in more detail and in light of new and updated information.

6.1 Submarine Factory Number 900: a source of new and relevant information

Following the experience with the K-27 submarine, the Soviet navy developed the Lyra class attack submarines (Alpha class submarines according to NATO classification) of which seven were commissioned between 1971 and 1981 (NES, 2013). The first vessel of this class was the nuclear submarine designated factory number 900 (K-377) which was commissioned in 1971. It later suffered a reactor accident in 1972 when a leak developed between the secondary water-steam circuit and the primary liquid metal circuit. This resulted in a high pressure which caused a rupture of the piping of the primary circuit and subsequent leakage of two tons of liquid metal coolant into the reactor compartment (Reistad and Ølgaard, 2006). As a result of the accident the coolant solidified and, as re-melting of it was not possible, the submarine was taken out of service.

In 1974, the reactor compartment was cut out of the submarine and sent to Severodvinsk. In 1985 the reactor compartment was filled with furfural and bitumen and made ready to be dumped in the Kara Sea. However, dumping of the reactor did not take place as it was not in line with the commitment imposed by the London Convention which was signed by the former Soviet Union (see Chapter 1). After almost 27 years, the reactor compartment of the submarine factory number 900 was finally defueled and dismantled at Gremikha in 2012 (NES, 2013). Because the same mothballing technologies and preservatives were used for the reactor compartment of the nuclear submarine number 900, information on the state of the protective barriers and experience gained in fuel unloading provided an invaluable contribution to any process involving decisions on K-27.

6.2 Corrosion of reactor vessel and primary circuit pipelines

Shells and primary circuit pipelines of K-27 reactors are made of stainless steel 1Kh18N9T or similar grades. During long-term underwater storage of K-27 they may face a two-sided corrosion attack. These steel components interact with lead-bismuth alloy or preservative (bitumen or furfural) from the inside and with preservative (bitumen or furfural) or seawater (in case of leakage through the protective barriers) from the outside.

An earlier study (Ignatiev et al., 2008) has shown that interaction of stainless steel and lead-bismuth alloy is of a galvano-chemical nature. Corrosion speed is ~3 μm/year, and the process depends weakly on temperature and other environmental parameters.

It is likely that the interaction of stainless steel and preservative materials (bitumen or materials based on furfural) is absent or is very slow at K-27. Observations of test samples from the dismantled reactor of the submarine factory number 900, which, as noted earlier, had been mothballed using the same technology as the one used for K-27, show very little interaction between preservatives and stainless steel components. Therefore, corrosion of stainless steel owing to interaction with preservatives can be neglected.

Interaction of stainless steel and seawater at K-27 is only possible in the event of water leakage through the protective barriers. Seawater is a well aerated (4-10 mg/l O_2) neutral (pH=7.3-8.6) electrolyte with high conductivity (0.5-6.7 Ω/m) owing to the presence of salts with a salinity ranging from 2 (bay waters, seas at the location of river outlets) to 40 ‰. Average salinity of the open seas and oceans is 35 ‰. In Stepovogo Fjord water salinity varies from 31 ‰ near the surface to 34 ‰ at the bottom.

Salts in seawater are mainly sodium, magnesium, calcium and potassium chlorides and sulphates. Owing to its high content of chlorides, seawater has the ability to depassivate metals. Seawater corrosion of metals proceeds according to an electrochemical process, mainly with oxygen depolarization and mixed diffusion-kinetic cathode control.

Corrosion pits and pitting along with general uniform corrosion are typical for the destruction of metals in seawater. The average speed of steel corrosion in seawater, calculated on the basis of weight loss, is from 0.05 to 0.20 mm/year, and an average speed of pit corrosion is up to 1 mm/year. A number of factors affect the speed of corrosion of metals in seawater:

- Seawater movement increases the speed of corrosion of most metals (under the conditions expected for K-27, stagnant contact of seawater with stainless steel is more likely to be the norm).
- Water temperature does not affect the corrosion speed mainly due to the effect of two competing processes: on the one hand, a rise in temperature speeds up oxygen diffusion and reactions in anode and cathode components; on the other hand, it decreases solubility of oxygen and, therefore, speed of corrosion.
- Contacts of different metals (e.g. at welding or soldering joints) in seawater result in corrosion acceleration for the more electronegative metal and suppression (or inhibition) for the more electropositive metal. The corrosion behaviour of contacts of different metals was not analysed when K-27 was constructed. Preliminary comparison of steady state potentials of materials used in reactors of K-27 shows that both electropositive and electronegative contacting metals are present. Therefore, in case of long-term contact with the seawater, there is a high risk of significant corrosion at locations of welding and soldering.

Taking these factors into account, a corrosion rate of 0.05 mm/year seems reasonable to adopt for the speed of corrosion of stainless steel (on the basis of weight loss) in case of K-27 long-term contact with the

seawater. This value has thus been used in further estimations. It is worth mentioning that IASAP considered a best corrosion rate of 0.02 mm/year for stainless steel in their study (IAEA, 1997).

6.3 Long-term behavior of bitumen

Bitumen is one of the main preservatives used at K-27 to create a reliable barrier against water leakage. Internal spaces and voids associated with partitions of steam generators, pumps and pipelines of the primary circuit, lead-water shielding tanks, steam generators of the secondary circuit and the entire reactor compartment (up to a level of 300 mm above the reactor lids) were filled with bitumen. Boric acid, comprising 11% by weight, was added to the bitumen for filling the portside steam generators of the secondary circuit. Mothballing of K-27 required a total amount of 270 m^3 of bitumen BND 90/120 (see Section 4.3).

The chemical composition of bitumen is very complex. The elemental composition of the different forms of bitumen vary within the following ranges: carbon - 70-80%, hydrogen – 10-15%, sulphur - 2-9%, oxygen - 1-5%, nitrogen - 0-2%. Bitumen contains these elements as hydrocarbons and compounds with sulphur, oxygen and nitrogen. Bitumen can also contain saturated hydrocarbons with various molecular masses from C_9H_{20} to $C_{30}H_{62}$.

In terms of structure, bitumen is a colloidal system with asphaltenes dispersed in a medium of oil and resin. Bitumen asphaltenes are dispersed as particles of 18-29 μm size, each particle being surrounded by an envelope of decreasing density, from heavy resin to oil.

Bitumen is hydrophobic, weather-resistant, soluble in organic solvents, highly non-rigid and tends to soften as temperatures increase towards the melting point. The most significant property of bitumen and bitumen materials is their resistance to aggressive agents such as seawater, alkalis and acids. All of these properties justify the selection of bitumen and its use as the protective barrier to prevent water leakage at K-27.

However the purpose of this section is to provide an evaluation of the ability of bitumen in retaining its protective properties with time and under the influence of various environmental factors (temperature, seawater, ionizing radiation etc.).

With time, a process of aging and changing of physical properties and chemical composition occurs in bitumen, because the components of bitumen change spontaneously and/or under the impact of the environment. The substances partially turn into other substances: oils turn into resins, and resins turn into asphaltenes. The process of oil turning into resin is slower than the process of resin turning into asphaltene, so with time the amount of asphaltenes in bitumen increases. With sufficient accumulation of asphaltenes, bitumen loses its plasticity, its fragility increases, wrinkles and breakages appear and deformation occurs. The rate with which these changes occur might vary from 8 to 30 % within 25 years. In addition, bitumen oxidizes with time due to polymerization and adjoining free oxygen. Therefore, the main functional properties of bitumen as a protective material against water leakage also deteriorate, in particular, its water repellent nature decreases (by 3-5 % for the same time period). Observations made in 2010-2011 indicated high fragility of the bitumen extracted from the reactor compartment of submarine factory number 900 (which was mothballed with the same technology as K-27).

Metal corrosion caused by bitumen is another long-term process which should be considered upon using it as the protective barrier against water leakage. Bitumen corrosion causes a destructive change in metals at locations of contact with bitumen, because interaction of bitumen and water causes slow oxidation processes and the breakdown of bitumen with formation of acid decay products (mainly sulphuric acid). Acid decay products are soluble in water and may lead to high concentrations of acid on the surface of metal even in small amounts of liquid, which corrodes metal. The speed of steel corrosion by this process is up to 120 μm within 25 years.

However, bitumen corrosion of metals can only start under conditions where moisture is present at the point of contact between metal and bitumen. Bitumen extraction from the submarine factory number 900 did not reveal any traces of bitumen corrosion on the reactor vessel, on the outlet fittings of the pipelines and internal surface of the reactor compartment.

Therefore, taking into account all data on bitumen aging, which is a slow process, bitumen in K-27 remains a reliable protective barrier preventing water leakage for the moment and in the immediate future. IASAP assumed a lifetime of 100 years for bitumen (IAEA, 1997).

6.4 Long-term behavior of preservatives developed based on furfural

A radiation-resistant preservative substance, based on furfural, was specially developed and used for protecting of SNF in K-27 reactors from coming in contact with the seawater after dumping (see Section 4.3). The preservative was supposed to last for a long period (initial estimates were 100 years, but later reduced to 50 years), had high radiation resistance (up to 245 MGy) and shrinkage loss up to 1%. Some 5.8 tons of the preservative were used in the process of mothballing of the submarine (see Section 4.3). Studies on the long-term behaviour of furfural are scarce and not well documented in the literature.

Radiation resistance of the furfural based preservative was studied by IPPE through irradiation of the samples placed in the reactor vessel 27/VT (a surface test bench) after unloading of the core. Control measurement of samples occurred annually after irradiation with a dose of 200 MGy with surveillance of weight change, geometry of the samples and gas release. The measurements did not show any significant changes of weight and geometry of samples and gas release, which led to the conclusion that the furfural preservative used at K-27 had the desired radiation resistance.

In 1985, during mothballing of the submarine factory number 900, 5 test samples of the furfural based preservative were manufactured. Three samples were pipes made of stainless steel with a diameter of 1.5 inches and their internal space filled with the preservative. Two other samples were of “pipe-in-pipe” shape with the space between the pipes filled with the preservative. Each sample had a length of about 100 mm. Test samples were stored at room temperature at a dark location. During preparation activities prior to work on the submarine factory number 900, samples were inspected. This showed that in two samples, adhesion between the preservative and the pipe wall was not rigid. The samples “rattled” when shaken, the preservative could move (for ~2 mm) under force, but it did not leave the pipe.

These results led to the assumption that the furfural based preservative loses adhesion with time. Therefore, a gap can appear between the preservative and the construction and water can penetrate into it.

Dismantling of the reactor retrieved from the submarine factory number 900 in 2011 provided valuable data on the long-term properties of the furfural based preservative (see Section 6.1). During this work, the preservative was extracted from the outlet fitting of the pipelines, upper reactor chamber and steam-heating jacket (total ~300 kg). It turned out that the preservative in the outlet fitting of the pipelines and upper reactor chamber was easy to drill, break, crumble and scrape from the internal walls of the items. With respect to these properties, it resembled dried cement wash with a high content of sand.

Cutting of the steam-heating jacket indicated that only half of the internal space was filled. Filling of the mothballed space was not complete, probably owing to insufficient fluidity of the preservative. However, adhesion of the preservative to the reactor vessel and jacket body was high. Even significant force applied to remove the untouched jacket failed, long cuts having to be made. Removal of the preservative required the application of significant force involving the breaking and scraping of the preservative.

Current available data regarding the long-term protective properties of the furfural based preservative are, therefore, contradictory. Consequently, a pessimistic assumption would be that such preservative loses its protective properties with time and hence cannot be regarded as reliable with regards to providing

protective barriers against water leakage. In the work conducted by IASAP a lifetime of 100 years was assigned to furfural (IAEA, 1997).

6.5 Uranium-beryllium fuel and Fission Products release

Available experimental data on the release of fission products from uranium-beryllium fuel are very scarce. Existing estimates are usually based on theoretical models of the release of fission products from the fuel.

Several mechanism for the release of fission products exist: thermal diffusion, radiation-enhanced diffusion, direct release (direct recoil), and vaporization.

Usually, when modelling fission products release from metal fuel, the fuel is considered as a composition of spherical grains of diameters equal to the average size of a fuel grain. The pore space or volume is considered as a system of interconnected channels with an exit to the surface of the fuel rod.

Fission products release from the fuel to the pore volume depends on many parameters, including the material of the sub-layer, the technology of manufacturing of the fuel rod (grain size), the fuel rod operation mode (speed of fission, temperature, burn-up, swelling, width of pores) and construction of the fuel rod (presence of false cladding, material of sub-layer).

Diffusion is the main mechanism of gaseous fission product release from the fuel grain. The diffusion rate is determined as the total of the rates of thermal and radiation-enhanced diffusion, which are $2 \cdot 10^{-12}$ cm^2/s and $2.6 \cdot 10^{-17}$ cm^2/s, respectively.

Parameterisation of the diffusion model was done with experimental data obtained at the test bench KM-1 for the fuel rods with U-Be. In contrast to fuel rods of K-27 reactor, the tested fuel rod was surrounded with a sodium sub-layer.

During the experiment, the casing of the test sample was pierced to measure the amount and composition of gas in fuel rods. Table 6.1 displays calculated and measured amount of Kr and Xe (per cm^3) in a compensatory volume (CV) of a fuel rod.

Table 6.1. Comparison of evaluation of stable gas release from the fuel; calculated vs. experiment.

	Calculated number of atoms N_0 formed	Calculated number of atoms released from the fuel N_{calc}	Measured number of atoms in CV of the sample – N_{exp}	$\frac{N_{calc}}{N_0}$,%	$\frac{N_{exp}}{N_0}$,%
Xe	2.12E+20	1.83E+20	1.80E+20	86.3	84.9
Kr	4.00E+19	3.45E+19	< 3.54E+19	86.2	<86.2

7 Spontaneous Chain Reaction (SCR): initiation possibilities and development

In this section factors and situations which are relevant to the development of an uncontrolled chain reaction will be identified and various criticality scenarios based on the considered plans for handling of the submarine will be investigated.

IASAP (IAEA, 1998) defined scenarios to investigate the possibility of any of the dumped reactor cores achieving a critical state. With regards to the K-27 submarine, the ingress of water into the core due to corrosion of the Pb-Bi coolant was considered as a probable scenario for the ignition of an SCR. Ingress of water into the core will increase neutron moderation and hence increase reactivity (a measure of how close an assembly of nuclear fuel is to criticality). In particular, in the case of K-27, corrosion of the coolant allows water to enter the core and surround the fuel pins, which eventually results in increasing the core's reactivity.

According to IASAP (IAEA, 1998) "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 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 United States of America) would be 44 MBq". In the following we will have a much shorter time perspective in mind upon consideration of different accident scenarios involving a spontaneous chain reaction.

7.1 Conditions for the SCR initiation

Situations may arise where reactor criticality might change and as a consequence of that a spontaneous chain reaction (SCR) is initiated. Such situations are most probable in the following cases:

- Ingress of water into the core;
- Relative displacement of fuel and absorbers resulting in the reduction of the compensation capacity of the control and protection system (CPS), which occurs with a certain delay, or in process of the core destruction.

As has been mentioned in Chapter 4, the K-27 reactor compartment was mothballed prior to dumping. The mothballing process created special protection barriers against sea water penetration into the core and sea water contact with SNF and other highly radioactive materials. In addition, special measures were taken during the course of mothballing to prevent changes in the relative position of fuel and absorbers to ensure the lowest sub-criticality of the system. Hence, the second situation detailed above will not be of much relevance for the case of K-27.

As discussed in Chapter 6, protracted underwater storage might result in the loss of the barrier protection capacity of the material used. This is especially the case for the preservatives based on furfural and, in a more general sense, relevant for structure surfaces having contact with Pb-Bi alloy, welded connections etc. as discussed below.

Water might penetrate directly into the reactor core through gaps which have appeared owing to the thickness decrease of fuel cladding and CPS casing. The strength of the resin-metal connection decreases with time owing to the difference in their thermal expansion coefficients followed by appearance of shearing and connection breakdown.

Welding connections can be considered impermeable only in a provisional sense. Welding materials which were used during the construction of the reactors were not chosen based on the consideration of their

durability in case of long term storage in the submerged position, probably because such scenarios would have been difficult to envisage at the time of construction of the submarine. Therefore, it can be assumed that the welding connections are the most vulnerable spots with respect to seawater corrosion and contact with preservatives. This assumption is partially confirmed by data obtained during defueling of the reactor of submarine factory number 900, when it was discovered that the caisson of the reactor vessel contained about 230 l of water.

The lead-bismuth alloy is the most reliable protection barrier. However, the alloy interaction with the fuel cladding, owing to physico-chemical interaction, causes ruptures. Conservative estimates based on calculations and experiments (Ignatiev et al. 2008) showed that the thickness of the fuel cladding and CPS cases might, under certain conditions, decrease as rapidly as 3 μm/year. Alloy freezing might result in cavity formation, because thermal contraction in this case reaches about 1.9 %.

The reactor protection barriers might be broken suddenly owing to natural disasters (e.g., earthquakes) or malicious acts like a terrorist act, or owing to natural degradation. In a case involving the destruction of the protection barrier, water could penetrate into the reactor core. Water acts as a neutron moderator and hence increases the reactivity of the reactor. The starboard and the portside reactors reach their critical states when the amount of water that has penetrated reaches 5-6 and 18-20 l, respectively.

The most probable water penetration pathways into the core are through the primary-circuit nipples and the steel channels of CPS passing through the whole reactor removable core and attached to the reactor lid (see Figure 4.2). During the mothballing of the submarine, 14 CPS channels out of 17 were filled by the Pb-Bi alloy. The remaining 3 CPS channels originally filled with the shielding gas were filled with furfural and orthoboric acid (26.9 % weight fraction). The internal volume of 1 CPS channel in the core (including that associated with the filling absorber and preservative) is about 1 l; therefore, the total CPS volume is ca. 17 l. In this case, the total volume of the gap between the channel wall and preservative, which is the free space for liquid accumulation, would hardly exceed several percent of the total volume. Thus, a conservative estimate gives 100 ml for one channel or 1.7 l for all CPS channels. Hence, the conclusion would be that the total water that can penetrate through the possible gaps between the internal walls of the CPS channels and filling absorbers is evidently insufficient to cause an SCR.

The main fraction of the liquid required to initiate an SCR might penetrate into the core in case of leak tightness failure in the reactor primary circuit. In addition, moisture might condense from air filling gaps, cracks, and contraction cavities inside the core and the primary circuit.

7.2 Possible SCR locations at the K-27 submarine

- The core of the starboard reactor, where the probability of an SCR should be considered to be greatest. This reflects the facts that the smallest volume of water would be required for SCR initiation, and the resulting SCR power is the highest with respect to possible consequences.
- The core of the portside reactor, which was damaged by the accident in 1968 and where about 20 % of its fuel was released into the primary circuit. Compared to the starboard reactor core, an SCR in the portside reactor core is less probable owing to larger initial reactor sub-criticality – the SCR initiation would require the penetration of a water volume into the core 3-4 times larger than for the starboard reactor core. The consequences of the SCR in this core are also less significant owing to the lower amount of radionuclides accumulated in the portside reactor core.
- The lower part of the portside steam generator accumulated about 20 % of the fuel released from the core. Calculations show that the presence of water or other hydrogen-containing materials in the steam generator pipes and aggregation of about 10 % of fuel in the inter-pipe space could cause a local critical mass. An SCR initiation in the steam generator is less probable and its possible consequences are the lowest. Nevertheless, an SCR is possible in the case of water penetration into

the steam generator from the secondary circuit and wash-out (dissolution) of boric acid from the bitumen which fills the steam generator.

As the probability of occurrence of an SCR and its associated consequences are highest for the starboard reactor core, we will only consider possible scenarios of SCR initiation related to this reactor.

7.3 Ingress of water and initiation of SCR: two scenarios

The speed with which an SCR progress and the consequences that follow depend on the speed of water penetration into the core and other conditions, as will be discussed below through the consideration of various scenarios.

7.3.1 Scenario 1 – slow water penetration into the core

The bitumen layer above the reactor might lose its integrity due, for instance, to cracking or any of the other reasons previously mentioned (see Section 6.3). In addition, welded joints of the primary-circuit and the CPS nipples might also be damaged by corrosion. As a result, water could start to penetrate slowly into the core. The penetration might last a long time - several months or even years. The speed of water ingress through the gaps between the walls of the pipe channels and filling preservatives can reach several milliliters per day. Accumulation of 5-6 l of water results in efficiency decrease of the CPS elements, which could create conditions for an SCR initiation.

In this case, the SCR would have virtually no environmental impact. According to the IPPE estimates, energy release in this kind of event would reach 10^{18} fissions (32 MJ). The energy value is sufficient to cause fuel heating, and the Pb-Bi alloy would reach its melting temperature (125 °C). The moisture which had penetrated would be ejected beyond the core space along the same gaps through which it had penetrated, and the SCR would be terminated. In addition, steam would eject certain amounts of Pb-Bi alloy from the core, which would partially block the water penetration path.

In this scenario, as a consequence of the SCR, certain parts of the fuel rods would likely lose integrity. We assume that the unsealed rod fraction would be 1 %, which exceeds the permitted value for the full-power reactor operation by a factor of about 10. This part of the fuel would release short-lived nuclides produced by the SCR (predominantly noble gases and iodine isotopes) and fission products accumulated in the fuel. However, activity release from the fuel rods would be of a relatively short duration, because the Pb-Bi alloy melted by the SCR would cool and prevent further activity release from the fuel. During this short time period, the produced nuclides would not have enough time to diffuse from the rod cores to the cladding in any significant amount. The released nuclides carried by steam would enter the reactor compartment and would be partially adsorbed by the walls of the gaps along the water penetration pathways. All the released activity would be localized in the reactor compartment volume. The dominating part of the activity would originate from radioactive noble gases (RNG) and volatile (iodine) gaseous fission products.

Such an accident, with SCR initiation and limited access to water, is only possible following the lifting of the submarine and draining the reactor compartment, when additional water ingress to the core would be absent.

Further development of this kind of scenario might proceed according to two scenarios that have different time characteristics.

In the first variant, the released steam energy is sufficient for the local destruction of the bitumen layer, which would block the water penetration paths for a certain time. In this case, the water penetration into the core would remain slow, which would result in repeated initiation of the SCR as soon as the next portion of liquid would enter the core (possibly, through other paths), and the whole event would repeat itself. It is impossible to estimate with sufficient accuracy the time interval between such successive initiations.

In the second variant, high steam pressure would expand gaps along pathways of water penetration. Owing to the cooling effect of water, the gaps would remain opened for free ingress of water to the core. In this case, the event would progress according to scenario 2 (below).

Currently, we cannot estimate which one of these two variants is more probable. Therefore, both variants have been considered to be equally probable.

7.3.2 Scenario 2 – fast water penetration into the core

The SCR considered in the previous section expands the water penetration pathways and thus increases the speed of water ingress into the core. As a result, energy release and water steam pressure of the SCR would increase, which results in a greater expansion of the water penetration pathways. The Pb-Bi alloy partially ejects water steam and leads to decrease in the reactivity. Steam pressure from the reactor side suppresses water ingress into the core, and the SCR fades. Steam would then transport a certain amount of coolant from the core thus increasing the free space that might be filled with a new portion of penetrated water. Upon cooling, water again penetrates into the core with higher speed and in greater amount. Thus, the SCR runs in a pulse mode with increasing energy release in each successive pulse.

According to IPPE estimates (Gromov et al., 1999), the SCR energy release for a water ingress speed of 0.2-0.5 l/min reaches about 10^{19} fissions (320 MJ). The energy value is sufficient for the complete meltdown of the alloy in the core, destruction of a certain part of fuel rods and activity release from the damaged fuel. The submarine vessel remains safe, and all activity is carried away from the core by steam (predominantly, short-lived RNG and iodine isotopes) and would be localized inside the reactor compartment.

Acceleration of water ingress into the core increases the frequency of SCR initiations. For an ingress speed of 0.2 and 0.5 l/min, the amount of water required for the SCR initiation (6 l) is accumulated in the core during at least 30 and 12 minutes, respectively.

It is worth noting that the speed of water ingress into the core might reach the levels considered above in the course of lifting the submarine, as a result of mechanical tension in the ships construction elements followed by the damage (cracking) of preservatives. However, water ingress during lifting would also decrease due to a pressure reduction (up to complete draining of the reactor compartment). Therefore, the repeated SCR pulses would stop during submarine lifting. This means that the maximum possible value of the energy release that can be assumed for a potential SCR during lifting would be equal to 10^{19} fissions.

The location of the submarine in deep water provides, in the case of the considered accident, for continuous water ingress into the core, which facilitates the progress of an emergency. For the worst case scenario, elements of the core might emerge and come to the surface of the melted coolant. Owing to such stratification of the core, the CPS efficiency decreases. In addition, certain absorbing elements in the CPS channels might be destroyed. If these channels are already filled with water, the generated steam might eject the CPS elements outside the core, which would further increase the reactivity.

As the reactor exceeds criticality the developed SCR pulse, enhanced by the closed volume, results in the damage of the submarine hull and could result in the release of fission products into the marine environment. The damage of the core would render the initiation of a recurring SCR impossible.

7.4 Maximum energy release due to the SCR

We have seen that several variants of initiation and development of an SCR are possible. Therefore, it is difficult to choose unambiguously an SCR scenario by which to evaluate the SCR consequences.

However, by adopting a conservative approach and assuming a worst case scenario based on the maximum possible energy released, it is possible to estimate the environmental impact related to the occurrence of an

SCR in the core of K-27. Such a situation might occur for the submarine in its current state at a depth of 30 m, when water ingress under hydrostatic pressure is a possibility.

In Chapter 8, we consider all possible emergency scenarios at various stages of lifting and transportation. For all these situations, the activity release from the core into the environment is substantially lower than for the case of "maximum SCR" which will be discussed below.

7.4.1 General description of the maximum SCR

By "maximum SCR" we mean an SCR with maximum power (highest fission number) where the energy released is high enough to cause the destruction of the reactor compartment and cause instantaneous release of activity into the environment. Such a situation only exists for the submarine in a submerged position and when the SCR event progresses according to the events considered in scenario 2 (see Section 7.3.2).

In order to estimate the SCR power, use can be made of the data obtained by IPPE in investigation of nuclear safety regarding storing the SNF unloaded from the surface test benches 27/VM and KM-1, which were the prototypes of the reactors of the nuclear submarine designs 645 (K-27) and 705 (Alfa class submarines), respectively (Gromov et al., 1999).

The IPPE analyzed possible occurrences of SCRs as a function of the amount of penetrated water (from 1 to 12.6 kg) into the spent removable core of the KM-1 test bench. The analysis considered two values for the speed of water ingress - 0.2 and 0.5 l/min. For the worst case (water ingress speed of 0.5 l/min) the total number of ^{235}U fissions reached 2.2×10^{19} within 2000 s, which is equivalent to an energy release of about 700 MJ. The energy generated is sufficient to cause the fuel to heat up to 470 °C with melting of the Pb-Bi alloy and boiling of the penetrated water. The maximum value of thermal power in the core reached about 218 MW for 586 s, which exceeds the reactor rating power by a factor of 1.46.

The thermal power of the KM-1 reactor and fuel load with respect to ^{235}U are approximately twice as large as the corresponding parameters of the reactor VT RM-1 installed in the nuclear submarine K-27. Therefore, we can assume in the first approximation that for K-27 the «worst» SCR would release energy in the order of about 320 MJ. The TNT (Trinitrotoluene) equivalent of the released energy in such an SCR is 77 kg. It is also worth noting that, owing to the long time required for the SCR development, the energy fraction applied to the destruction of the protection barriers would be even lower than the evaluated value.

In addition, we can also make an estimation of the maximum possible SCR using available data from publications. The White Book-2000 (Sivintsev et al., 2005) considers accidents involving SCR initiation for critical assemblies, research and experimental reactors, and equipment used in the production of nuclear materials during the time period 1945-1999. The review indicates that in 60 accidents which occurred all over the world, the maximum energy released due to the occurrence of the SCR was within the interval of 10^{16}–10^{20} fissions. It is worth noting that these values correspond to the whole pulse series or to a reactor 'runaway'. The energy released in the individual pulses never exceeded 10^{19} fissions.

The most serious accident involving SCR initiation resulting in the destruction of the reactor vessel and subsequent contamination of the environment occurred in 1985 in the course of reloading of the reactor of the submarine K-431 in the Chazhma Bay, in the Far East region of Russia. Other severe accidents at nuclear submarines with the SCR initiation did not result in any significant radiation environmental impact.

The SCR accident occurred in the reactor of type VM-A within a first-generation nuclear submarine moored at the pier of the shipyard in Chazhma Bay. As a result of a thermal explosion, a fresh fuel assembly was thrown out from the reactor vessel. Measurements of the activation nuclides indicate that the estimated number of fissions during the SCR reached 5×10^{18}(Gusev and Dmitriev, 1978). Further investigations performed by the Russian Research Institute of Kurchatov based on the experimental data obtained after the accident on the

critical assembly (Kolobashkin et al., 1989) concluded that this value (5×10^{18}) would be the maximum possible number of fissions for the SCR in the Chazhma Bay accident.

The overall conclusion is that the value of 10^{19} fissions is a likely estimate regarding the potential SCR power in the starboard reactor of K-27. This energy however is insufficient to destroy the K-27's strong hull and causing damage to the core to such an extent that no pulsed SCR would occur. The reasons are both the design of the LMC reactors and additional protection barriers (furfural, bitumen) inserted into the reactor compartment prior to dumping. As we have already mentioned, the actual maximum value of the thermal power in this case would exceed the rated power of the reactor by a factor of about 1.5. Such a release of energy would only destroy certain part of the fuel rods.

On the other hand, the SCR released energy might increase owing to the closed volume effect, which prevents the ejection of steam from the core space, and owing to the accelerated water ingress to the core (above 0.5 l/min). In addition, several successive SCR pulses are possible, which would result in the complete destruction of the protection barriers. However the presence of these additional factors is regarded as being highly unlikely.

Therefore, we adopt a conservative approach and assume a maximum energy release from the SCR equal to 10^{20} fissions (3.2 GJ). In this case, the core and the submarine vessel would be destroyed.

It is also worth noting that «conservative» estimates of the SCR power affects the evaluation of the radiation consequences mainly with respect to the short-lived radionuclides produced directly during the SCR. Activity of the released long-lived fission products accumulated in the SNF depends to some extent on the SCR power and is mainly determined by the percentage of damaged fuel rods. The dominant contribution to the long-term contamination of the environment comes from the long-lived fission products, mainly ^{137}Cs and ^{90}Sr, which in turn determine the radiological consequences of the accident.

7.4.2 Qualitative evaluation of the emergency situations involving an SCR

We can determine other probable scenarios of emergencies with SCR initiation in the course of lifting of the submarine. However, their consequences would be lower than that of the maximum possible accident.

The maximum possible accident could occur if the submarine remains at the dumping location, at a depth of 30 m, for a prolonged duration (decades rather than centuries). As already mentioned, the probability of such an accident in the near future is equal to the probability of the «moderate scenario» with aperiodic occurrences of weak SCR pulses (with intervals from several months to several years) caused by the slow penetration of water into the core. At the same time, over a long-term perspective, this «moderate» scenario might also develop into a maximum possible accident.

Mechanical tensions in the submarine construction generated in the process of lifting might result in damage (cracking) of the preservatives used as protection barriers. As a result, water ingress into the core would be accelerated. However, the lifting would take limited time, while the water inflow to the core would decrease with pressure reduction up to the complete draining of the compartment. Thus, a repeated SCR would have no time to occur, and we can take the intermediate value 10^{19} as the maximum energy release. In this case, the submarine hull would remain intact, and the released activity (mainly RNG and iodine) would be virtually completely localized within the submarine.

8 Release scenarios related to a potential raising and transporting of the submarine

As already mentioned in Chapter 4 (see Section 4.5), the issue of raising and transporting K-27 to land for defueling has been discussed by Russian specialists as a potential option regarding handling of the submarine. In the present study, accident scenarios related to the raising of the submarine from its current position to the surface and its transportation to Gremikha for dismantling have been developed. With regards to accident scenarios, several alternatives can be considered depending on where and how a hypothetical accident will take place and whether the subsequent releases occur under water or at the water surface.

With regards to the type of fission products which could be released, all possible accident scenarios can be divided into two categories based on whether these scenarios involve the occurrence of an SCR or not:

- No SCR (mechanical destruction): release of previously accumulated long-lived fission products along with some gaseous fission products from the fuel in the event of cladding destruction;
- SCR occurs: in addition to previously accumulated long-lived nuclides, newly produced fission products by the SCR are also released (mainly short-lived, gaseous, and volatile radioisotopes).

However if the type of release is of concern, we can again divide the possible situations into two groups depending on whether a possible accident has the potential to destroy the reactor compartment or not:

i) Destruction of reactor vessel and reactor compartment

This situation might occur both with and without an SCR:

- An SCR with a power of 10^{20} fissions would destroy the reactor vessel and the reactor compartment hull, and all active fission products would enter the environment directly and instantly. This damage might only occur in the underwater position.
- Reactor vessel and reactor compartment hull can be destroyed without an SCR at all management stages, similar to the previous case, owing to accidents such as those caused by a navigation wreck or a terrorist act.

ii) Reactor vessel destroyed but reactor compartment hull is intact

These situations can occur either when the submarine is underwater or at the surface. In the underwater position, the SCR is more probable at the start of lifting activities. Moving the submarine might initiate penetration of condensed moisture or leakage water into shrink gaps in the core. The amount of the penetrated moisture might be sufficient for an SCR ignition.

Such a senario can be envisaged also in the case of accidents where no SCR is involved. Such situations are rather unlikely, but might occur during the lifting of the submarine owing to mechanical damage, when the ship keels over and then bumps against the sea bottom or an underwater rock. This could result in the reactor dropping inside the reactor compartment hull.

8.1 Justification and choice of release scenarios

All combinations of various scenarios for emergency situations are depicted in Figure 8.1. Irrespective of the current geographical location of the submarine, the fission products release process would mainly depend on the SNF impact level (internal (SCR) or external mechanical, for example, destruction of the reactor

compartment hull with damage of the reactor vessel and fuel elements) and the submarine position in the water column (underwater - at the bottom or in course of lifting-, in the submerged position, or at the surface). The latter factor determines the ratios of the activity released into air and water.

In the following, three main 'typical' scenarios will be considered for the calculation of the release parameters:

- reactor vessel and reactor compartment hull destroyed, instant activity release into the environment;
- reactor vessel destroyed, reactor compartment hull intact, SCR may or may not occur, slow activity release into water or atmospheres;
- fire during spent removable-core retrieval.

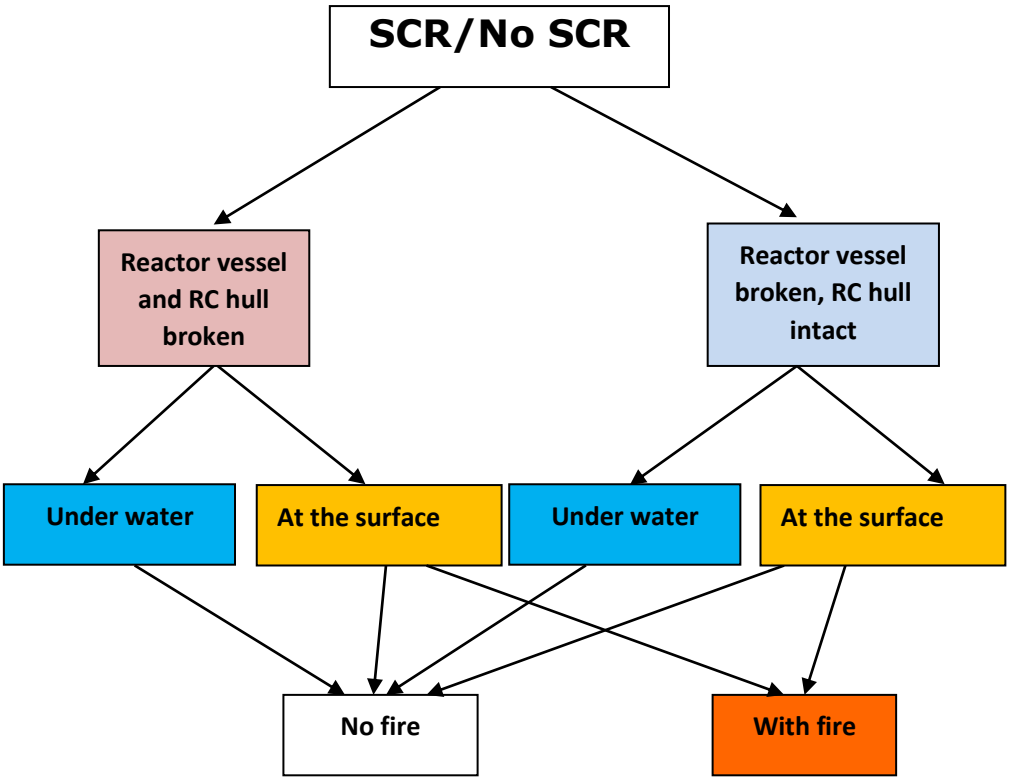


Figure 8.1. Various considered scenarios of emergency situations with or without an SCR involvement. RC stands for reactor compartment.

8.2 Evaluation of characteristics of the releases for a scenario with destruction of both reactor vessel and reactor compartment

The rate of fission products releases from the reactor vessel depends on the level of destruction of the SNF cladding which would be determined by the power of the SCR explosion or the extent of the mechanical impact on the reactor vessel during an accident.

For a protracted stay in an underwater position, the discharge of activity present within the reactor compartment (irrespective of how this occurred) would depend on the state of the environment. Even for a low velocity and stable near-bottom flow (below 0.1 m/s) a weak water flux would still exist. It is also necessary to take into account the thermal energy of the SCR which would be distributed in the reactor vessel. Heat carried away to the reactor compartment, would cause convection, thus enhancing activity dispersal through the compartment and its transport by water.

These considerations apply especially to the scenario involving sinking of the submarine during its transport to Gremikha. For example, in the Barents Sea (average depth 300 m) surface flows with general counterclockwise circulation dominate. In certain locations, the flow speed reaches 2-5 m/s and decreases exponentially with growing depth.

Upon core destruction, fuel elements come into contact with seawater, which results in release of fission products owing to corrosion. According to experimental data, the average annual relative rate of fission products release is equal to 1 %. The fission products activity in the SNF of the starboard reactor is 4.11 kCi (0.15 PBq), the average annual input of radionuclides (mainly, ^{137}Cs+^{137m}Ba and ^{90}Sr+^{90}Y) into the marine environment at the K-27 dumping location, at Stepovogo Fjord, would therefore be ~40 Ci (1.5 TBq).

8.2.1 Possibility and consequences of an SCR in the lifted position

The probability of an SCR occurrence in the lifted position at the surface or on land is considered only hypothetically. The SCR power would be significantly lower than the one in the underwater position.

According to existing estimates, such a hypothetical SCR might result in sudden power increase equivalent to 10^{18} fissions. The energy from this power increase is sufficient to heat fuel, melt coolant, and damage the sealing in certain parts of fuel rods. With the passage of gaseous fission products, melted Pb-Bi nuclides would be released from the depressurized fuel elements to the internal space of the reactor vessel. As soon as the sealing of the reactor vessel is also broken (by SCR), radioactive products can enter the reactor compartment and further to the atmosphere. The discharge would be mainly dominated by radioisotopes of Kr and I.

The release of the nuclides newly produced by the SCR pulse would be negligible. Therefore, the gaseous ^{85}Kr and volatile ^{129}I accumulated during the previous reactor operation would dominate the release. The activity of accumulated ^{129}I would be almost five orders of magnitude lower than that of ^{85}Kr, and the radiological consequences of the iodine release would be thus negligible compared to ^{85}Kr.

Hence it is sensible to assume that the release of radioactivity beyond the fuel space can take place as a result of such a hypothetical SCR accident scenario but this would be limited to ^{85}Kr which would have accumulated during the previous reactor operation. To make a conservative estimate we can assume that all this accumulated activity would be released from the reactor into the atmosphere instantly.

8.3 Evaluation of a scenario involving the accidental releases for the case of destroyed reactor vessel and intact reactor compartment

It has already been mentioned that two situations leading to the loss in integrity of the reactor vessel (one with the additional loss of hull integrity) might occur, with or without the initiation of an SCR, and that both cases might occur in the underwater state and at the surface (leading to 4 possible accident configurations).

An SCR of power up to 10^{18} fissions would destroy the reactor vessel and fission products of all types would be released to the environment in two steps: first from the reactor to the reactor compartment, next from the reactor compartment to the environment. In other words, the activity release rate decreases while the release duration grows. An SCR while underwater is more probable at the start of the lifting operation. The submarine motion might initiate penetration of condensed moisture or leakage of water into the core. Provided that a sufficient amount of water gets into the core, it might initiate an SCR.

Reactor vessel destruction with depressurization inside the reactor compartment is also possible without the occurrence of an SCR. This might happen as a result of a blow from the bottom, for example, in the case of a break during lifting or during transportation, e.g. similar to what happened in the case of the sunken nuclear submarine B-159 in the Kola Bay.

8.3.1 Accident scenarios at the surface

When the submarine is positioned at the surface, i.e. during transportation or dock operation, the probability of an event involving an SCR decreases, but it is still possible. The low probability of an SCR reflects the limited access to water in this position. The absence of water in the compartment would reduce the impact on the reactor compartment hull, and hence increase the probability of the reactor compartment remaining intact. The level of total activity released and also the types of nuclides in this case are similar to the previous SCR scenario.

For accident scenarios with the submarine located at the surface with its compartments drained, the SCR probability is minimal. In the case of initiation of such a hypothetical SCR, its power would be minimal and would not exceed 10^{18} fissions. In this case mainly volatile fission products would be released into the atmosphere, and a certain portion of the release would be localized within the submarine's hull.

8.3.2 Accident scenarios involving losing the submarine

For the case of losing the submarine during lifting and the potential destruction of the reactor vessel without an SCR, the activity releases into the water would be determined by the state of the fuel cladding. Depending on its state (degree of damage), the corrosion of the cladding in seawater would take from 6 months to 1 year. Upon destruction, fission products would be washed away from the fuel into the reactor compartment and further into the water as in the case of an SCR. Thus, the activity release would be dominated mainly by long-lived fission products. The annual average nuclide release (mainly ^{137}Cs+^{137m}Ba and ^{90}Sr+^{90}Y) into the marine environment at the submarine dumping location would reach ~ 40 Ci (1.5 TBq).

It is also necessary to consider an accidental scenario related to the sinking of the submarine in deep water (over 100 m), for example during its transportation to Gremikha somewhere in the Barents Sea. In this case the possible scenarios are similar to those considered in relation to the current dumping location. Owing to higher hydrostatic pressure and possible mechanical damage of the reactor vessel and the submarine hull, the probability of the «worst case scenario» with the destruction of the submarine hull increases. The activity released to the environment is the same as in the case of the maximum power SCR accident with the only exception being that all released radionuclides would be localized in the seawater column.

8.4 Accident scenarios with the SCR initiation: a summary

To summarize, regardless of the submarine's geographical position, several different release pathways could be envisaged based on consideration of various possible locations of the submarine in relation to the water column (in shallow water, in deep water, at the surface) and in conjunction with various possible accidental scenarios. Table 8-1 displays all possible combinations of accident scenarios including SCR initiation during lifting of the submarine. In Chapter 9, we will consider the quantitative characteristics of the releases.

Table 8-1. Variants of the accident scenarios with the SCR initiation during lifting and transportation of the K-27 submarine.

Scenario	Depth, m	Maximum number of fissions for SCR	SCR type	Release localization
At the dumping location	30	10^{18}	Periodic, with period from months to years	RC*, water
		10^{20}	Fast with destruction of the reactor vessel and RC hull	Water, air
Lifting	0-30	10^{19}	Single, impulsive	RC, water, air
Loss in deep water (during transportation)	>100	10^{20}	Fast with destruction of the reactor vessel and RC hull	water
		10^{18}	Periodic, with period from months to years	RC, water
At the surface	0	10^{18}	Single, impulsive	RC, water, air

*RC stands for reactor compartment

8.5 Evaluation of accidental releases for the fire scenario

With regards to the development of a worst case scenario which involves outbreak of a fire we have considered a rather unlikely case of a plane crash during the retrieval of the spent fuel at Gremikha. In this case the SNF is in its most vulnerable position with respect to heavy mechanical damage.

With several tons of aviation fuel on board, it is highly probable that fire would cause the release of activation products associated with the core structures— fuel cladding, fuel grids, screens.

In order to make estimates of the released activity in this case, use has been made of the available data from empirical measurements and calculated characteristics of accidents which have occurred earlier, i.e. accident with SCR and fire such as the Chazhma Bay accident in 1985 and Chernobyl accident in 1986.

Over the course of the initial explosion of the aviation fuel, fragments of the core are ejected, and a short-term release, saturated by aerosols, occurs above the object. The temperature rises virtually instantaneously and the SNF reaches its melting point over the same timescale.

The dispersion of heavy fuel fragments and fire consequences are considered less significant compared to aerosol release with respect to radionuclide transport over long distances, and is not, therefore, considered further.

The time variation in the ascension velocity of the mass center of an aerosol release for two considered wind velocities *v* (5 and 15 m/s) is shown in Figure 8.2. It is apparent that wind velocity significantly affects the release trajectory: the higher the wind velocity, the lower and longer is the trajectory. The calculated initial ascending velocity is approximately 9 m/s. At wind velocity 5 m/s, the cloud reaches its highest point (h ~ 153 m) approximately in one minute; at the wind velocity 15 m/s, the cloud reaches a maximum height (h ~ 70 m) in 30 s.

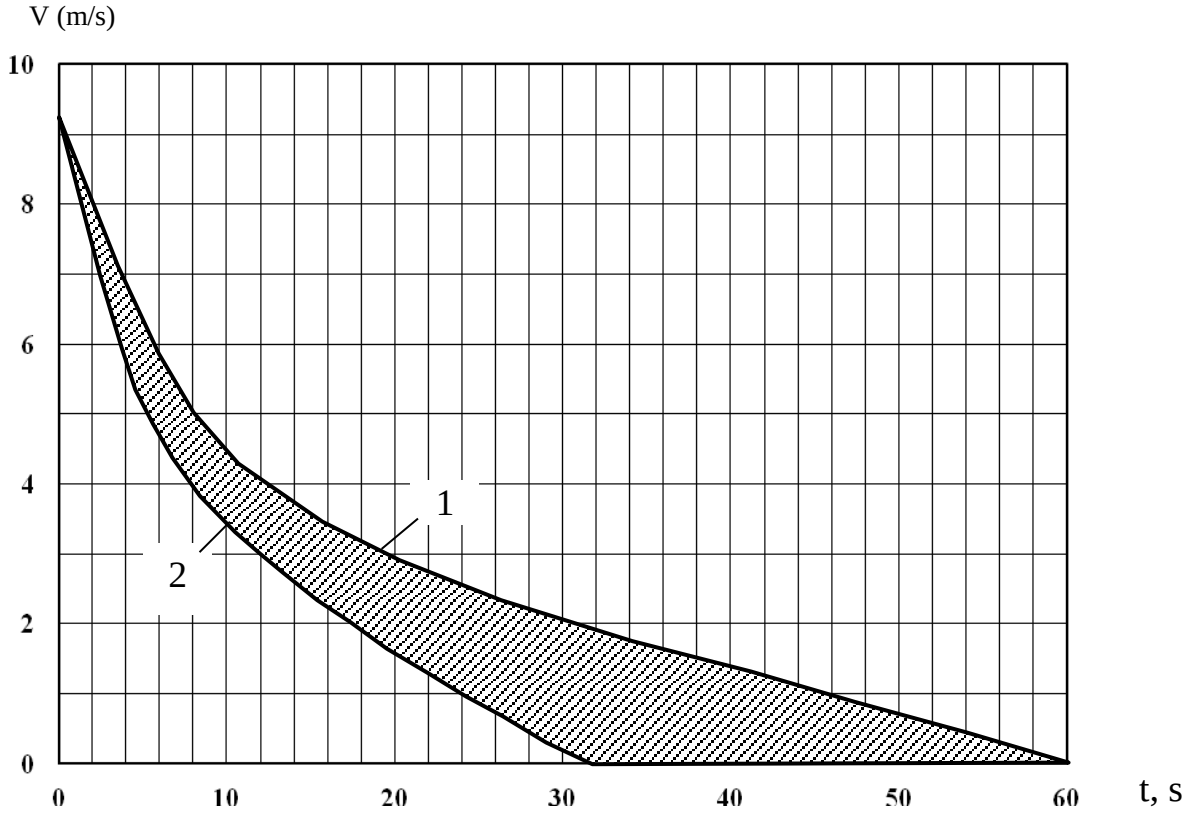


Figure 8.2.Time (t) dependence of the ascending velocity (v) of the mass center of the steam-air cloud. Curves 1 and 2 correspond to the wind velocity 5 and 15 m/s, respectively.

According to estimates based on the measurement of the soil contamination in the radioactive trace formed as a result of the accident in Chazhma Bay, the mass fraction of fuel carried away by the fire is approximately 1 % of the total fuel load. In comparison, the fuel fraction deposited near the 4th block of the Chernobyl NPP as a result of the SCR accident with subsequent fire is approximately 3.5 %. It is worth noting that this estimate is valid only within one order of magnitude accuracy.

Using a conservative approach and allowing for the possibility that a fire can be started and thereafter propagates upon retrieval of spent removable-core from the reactor vessel(a situation under which the spent removable-core is more vulnerable) we assume that the fire products would carry away in the aerosol cloud up to 15 % of the total activity of the core.

The total activity of the starboard SRC at the end of 2013 is about 5 kCi or 0.185 PBq (Table 5.8). Depending on the state of the PS SRC during unloading, fission products in its core might also be affected to some degree by a hypothetical fire. Thus, it is considered prudent to increase the total activity value by a factor of 1.2-1.3. In this manner, the total activity in the aerosol cloud formed by a fire during SRC retrieval might reach 1 kCi (37 TBq). For determination of the types of radionuclides and amounts of activity released, a fraction of 15 - 20 % can be applied to the activities listed in Tables A1-A10 in the Appendix for the considered year.

9 Source term

A necessary step in considering any scenario is the characterization of a source term, i.e. the quantity and types of released radionuclides (and also their physical and chemical properties) and associated release patterns.

The issue of the source term related to K-27 has already been addressed in an earlier work conducted by IAEA (IAEA, 1997). The work was comprehensive and detailed. It involved discussing characteristics of the different compartments of the reactor, reactor operating history, disposal related measures, protective barriers (both initial constructional ones and those provided prior to dumping), and the likely behavior of the barriers with time in the marine environment.

However, since the conduction of IAES's work in the 90's, some new information has become available, both through further research and also through work conducted on the similar submarine (i.e. submarine factory number 900, see Section 6.1). Additionally and as already touched upon in the introduction, the present analysis has developed beyond IAEA's work by looking at the issue of criticality in more detail and by considering potential releases to the atmosphere.

We are aware of the fact that a detailed and accurate prediction of the source term is almost impossible as this requires precise knowledge of a number of factors, inter alia, core design, fuel matrix, cladding, containment and power production history. Unfortunately these information are not easily available and hence in order to do an assessment we have to rely on assumptions. However, it is obvious that these assumptions should be based on the most accurate and up to date available knowledge. The source term defined below is considered to fulfill this criterion.

9.1 Evaluation of quantity and types of radionuclides generated by an SCR of maximum power

Considering an accidental scenario which involves an SCR with a power of 10^{20} fissions, the code ORIGEN-2 was used to calculate the amounts of fission products produced.

Table 9.1 displays the total activities of the most significant fission products created in the core as a result of an SCR pulse of 10^{20} fissions. Taking into consideration the possibility of several successive SCR pulses with step-by-step destruction of the reactor compartment, a minimum calculation time of one hour has been assumed.

Table 9.1. Total activities of the main fission products created in the core of the starboard reactor of K-27 upon the occurance of an SCR of 10^{20} fissions (Bq).

Time	1 hour	3 hours	12 hours	1 day	7 days	28 days
RNG(Kr, Xe)	7.73E+14	2.85E+14	9.03E+13	4.88E+13	5.51E+12	3.44E+11
Iodine, bromine	1.30E+15	7.22E+14	1.68E+14	7.33E+13	4.44E+12	3.03E+11
Ru, Te	1.69E+15	2.86E+14	2.62E+13	1.34E+13	3.22E+12	4.70E+11
^{140}La	9.51E+10	2.29E+11	7.70E+11	1.35E+12	2.93E+12	1.02E+12
^{140}Ba	4.03E+12	4.03E+12	3.96E+12	3.85E+12	2.78E+12	8.84E+11
^{143}Ce	3.36E+13	3.40E+13	2.82E+13	2.19E+13	1.06E+12	2.68E+07

^{143}Pr	5.14E+10	1.98E+11	7.81E+11	1.39E+12	2.72E+12	9.69E+11
^{141}Ce	1.49E+11	5.40E+11	1.29E+12	1.44E+12	1.29E+12	8.25E+11
^{144}Ce	1.58E+11	1.58E+11	1.57E+11	1.57E+11	1.55E+11	1.47E+11
^{144}Pr	1.53E+11	1.57E+11	1.57E+11	1.57E+11	1.55E+11	1.47E+11
^{147}Nd	1.73E+12	1.80E+12	1.76E+12	1.71E+12	1.17E+12	3.10E+11
^{89}Sr	9.21E+11	9.84E+11	9.81E+11	9.73E+11	8.95E+11	6.73E+11
^{91}Sr	1.14E+14	9.88E+13	5.11E+13	2.14E+13	5.96E+08	6.88E-08
^{91}Y	3.57E+10	1.30E+11	4.63E+11	6.73E+11	7.70E+11	6.03E+11
^{93}Y	7.81E+13	6.85E+13	3.68E+13	1.62E+13	8.25E+08	7.84E-07
^{95}Zr	8.44E+11	8.58E+11	8.55E+11	8.51E+11	7.99E+11	6.36E+11
^{97}Zr	6.29E+13	5.81E+13	4.00E+13	2.45E+13	6.70E+10	7.03E+01
^{95}Nb	5.55E+08	1.96E+09	8.21E+09	1.65E+10	1.07E+11	3.11E+11
^{137}Cs	4.70E+09	4.70E+09	4.70E+09	4.70E+09	4.70E+09	4.70E+09
^{137m}Ba	4.44E+09	4.44E+09	4.44E+09	4.44E+09	4.44E+09	4.44E+09
^{90}Sr	4.70E+09	4.70E+09	4.70E+09	4.70E+09	4.70E+09	4.66E+09
^{90}Y	1.79E+08	2.76E+08	6.85E+08	1.17E+09	3.96E+09	4.66E+09
Total, Bq	4.06E+15	1.56E+15	4.52E+14	2.32E+14	2.81E+13	7.66E+12
Total, Ci	*1.10E+05*	*4.22E+04*	*1.22E+04*	*6.27E+03*	*7.59E+02*	*2.07E+02*

Table 9.1 shows that the total activity of the fission products created by an SCR would decrease from being one order of magnitude above residual activity (10 kCi) at the start to a level lower than the total residual activity within one day. A month later, the produced activity would be within 5 % of the total residual activity. Activities of the SCR-produced long-lived nuclides ^{137}Cs and ^{90}Sr are 4 orders of magnitudes lower than the activities of the same nuclides contained in fuel.

9.2 Evaluation of quantity and types of radionuclides released following an SCR of maximum power

The SCR pulse in the core would result in the local destruction of some fuel rods. In the absence of data on SCR events in LMC reactors, it is impossible to determine unambiguously the fraction of the damaged fuel rods. Expert judgment can be made based on consideration of the scenario of the nuclide release in the submarine accident which occurred in the Chazhma Bay (Sivintsev, 2000). The scenario assumed that the SCR, with an energy release of 10^{19} fissions, resulted in the destruction of 10% of the reactor fuel.

It is worth noting that the Pb-Bi alloy in the LMC reactors provides an additional protection barrier which prevents fuel rods destruction and hence nuclide release from damaged fuel. On the other hand, in our case, the energy release is one order of magnitude higher. Taking these points into consideration, it can be assumed that the fraction of the damaged fuel in the core would be up to 30 %.

Based on the experience gained from the Chernobyl accident, it is known that the release of fission products upon an SCR is determined by their volatility. According to generic Russian data (Pologikh, 2001) the relative

release fractions of various nuclide groups from the fuel, upon the occurrence of an SCR, is as given in Table 9.2.

Table 9.2. Relative release fractions of various nuclide groups from the fuel, upon the occurrence of an SCR.

Nuclide group	Release fraction, %
RNG (krypton, xenon)	100
Iodine, bromine	50
Cesium	20
Ruthenium, tellurium	10
Other	5

Therefore, allowing for a damaged fuel fraction of 30 %, the relative releases of various nuclide groups in the release from the K-27 submarine are given in Table 9.3.

Table 9.3. Relative releases of various nuclide groups in the discharge from the K-27 submarine.

Nuclide group	Release fraction, %
RNG (krypton, xenon)	30
Iodine, bromine	15
Cesium	6
Ruthenium, tellurium	3
Other	1.5

Table 9.4 displays activities of the short-lived radionuclides released to the sea as a result of the occurrence of an SCR with the energy of 10^{20} fissions at the starboard reactor of K-27.

Table 9.4. Total activities (Bq) of the short-lived fission products released into the sea from the starboard reactor of K-27 upon the event of an SCR of 10^{20} fissions power.

Time	1 hour	3 hours	12 hours	1 day	7 days	28 days
RNG(Kr, Xe)	2.32E+14	8.55E+13	2.70E+13	1.47E+13	1.65E+12	1.03E+11
Iodine, bromine	1.95E+14	1.08E+14	2.52E+13	1.10E+13	6.66E+11	4.55E+10
Ru. Te	5.07E+13	8.55E+12	7.84E+11	4.03E+11	9.66E+10	1.41E+10
^{140}La	1.43E+09	3.43E+09	1.15E+10	2.03E+10	4.40E+10	1.53E+10
^{140}Ba	6.07E+10	6.07E+10	5.96E+10	5.77E+10	4.18E+10	1.33E+10
^{143}Ce	5.03E+11	5.11E+11	4.22E+11	3.28E+11	1.59E+10	4.03E+05
^{143}Pr	7.73E+08	2.96E+09	1.17E+10	2.08E+10	4.07E+10	1.45E+10

Time	1 hour	3 hours	12 hours	1 day	7 days	28 days
^{141}Ce	2.24E+09	8.10E+09	1.93E+10	2.16E+10	1.94E+10	1.24E+10
^{144}Ce	2.36E+09	2.36E+09	2.36E+09	2.36E+09	2.33E+09	2.21E+09
^{144}Pr	2.30E+09	2.36E+09	2.36E+09	2.36E+09	2.33E+09	2.21E+09
^{147}Nd	2.59E+10	2.70E+10	2.64E+10	2.56E+10	1.75E+10	4.66E+09
^{89}Sr	1.38E+10	1.48E+10	1.47E+10	1.46E+10	1.34E+10	1.01E+10
^{91}Sr	1.71E+12	1.48E+12	7.66E+11	3.21E+11	8.95E+06	1.03E-09
^{91}Y	5.37E+08	1.95E+09	6.96E+09	1.01E+10	1.15E+10	9.07E+09
^{93}Y	1.17E+12	1.03E+12	5.51E+11	2.43E+11	1.24E+07	1.18E-08
^{95}Zr	1.27E+10	1.29E+10	1.28E+10	1.28E+10	1.20E+10	9.55E+09
^{97}Zr	9.44E+11	8.73E+11	5.99E+11	3.68E+11	1.01E+09	1.05E+00
^{95}Nb	8.33E+06	2.94E+07	1.23E+08	2.47E+08	1.60E+09	4.66E+09
Total. Bq	4.82E+14	2.06E+14	5.55E+13	2.75E+13	2.64E+12	2.61E+11
Total. Ci	*1.30E+04*	*5.58E+03*	*1.50E+03*	*7.43E+02*	*7.13E+01*	*7.04E+00*

Along with the short-lived radionuclides, the long-lived fission products accumulated in SNF during submarine operation would be released (see Tables A1 and A2 in Appendix). The activities of these radionuclides, released to the marine environment immediately following an SCR event, are given in Table 9.5.

Table 9.5. Total activities of the long-lived fission products, accumulated in the fuel of the starboard reactor of the K-27 submarine and the released fractions into the sea following the event of the SCR of 10^{20} fissions power.

Nuclide	T ½	Activity as of 2013 (tab. A2), Bq	Release fraction	Activity, Bq	Activity, Ci
^{79}Se	6.5E+4 years	3.45E+08	1.50%	5.18 E+6	1.40E-04
^{85}Kr	10.72 years	1.06E+12	30.00%	3.18E+11	8.59E+00
^{87}Rb	5.56E+10 years	2.32E+04	1.50%	3.48E+02	9.41E-09
^{90}Sr	29.2 years	3.90E+13	1.50%	5.85E+11	1.58E+01
^{90}Y	64.26 hours	3.90E+13	1.50%	5.85E+11	1.58E+01
^{93}Zr	1.5E+6 years	2.24E+09	1.50%	3.36E+07	9.08E-04
^{93m}Nb	13.6 years	1.87E+09	1.50%	2.80E+07	7.58E-04
^{107}Pd	6.5E+6 years	1.18E+07	1.50%	1.77E+05	4.78E-06
^{121}Sn	27.05 hours	6.74E+09	1.50%	1.01E+08	2.73E-03
^{121m}Sn	52 years	9.09E+09	1.50%	1.37E+08	3.69E-03
^{126}Sn	1.0E+5 years	3.01E+08	1.50%	4.51E+06	1.22E-04

Nuclide	T ½	Activity as of 2013 (tab. A2), Bq	Release fraction	Activity, Bq	Activity, Ci
^{126}Sb	12.4 days	4.18E+07	1.50%	6.25E+05	1.69E-05
^{126m1}Sb	19 min	3.01E+08	1.50%	4.51E+06	1.22E-04
^{129}I	1.6+7 years	2.47E+07	15.00%	3.7 E+6	1.00E-04
^{135}Cs	2.3E+6 years	1.50E+09	6.00%	9.00E+07	2.43E-03
^{137}Cs	30 years	4.43E+13	6.00%	2.66E+12	7.18E+01
^{137m}Ba	2.552 min	4.18E+13	6.00%	2.51E+12	6.78E+01
^{147}Pm	2.623 years	5.95E+09	1.50%	8.92E+07	2.41E-03
^{151}Sm	90 years	1.79E+12	1.50%	2.69E+10	7.26E-01
^{152}Eu	13.2 years	5.02E+05	1.50%	7.55E+03	2.04E-07
^{155}Eu	4.68 years	7.81E+10	1.50%	1.17E+09	3.17E-02
Total		1.67E+14		4.80E+12	1.30E+02

As can be seen from Table 9.4 and 9.5, one hour after the accident, the total activity of the released short-lived fission products (4.82E+14 Bq) is higher by 2 orders of magnitude than the total activity of the released long-lived fission products (4.80E+12 Bq). A week later, the activities become equal, while a month later the main contribution to the radiation contamination of the environment would come only from the long-lived fission products accumulated in the SNF, and the dominating contribution (over 90 %) would come from ^{137}Cs+^{137m}Ba and ^{90}Sr+^{90}Y. The fraction of the short-lived fission products produced by the SCR would decrease to 7 %.

9.3 Dispersion of radioactivity following an underwater release

Core melting during an SCR event results in the blowout of overheated steam containing fission products. Bubbles ejected to the surrounding water would pass cracks and ruptures in the protection barriers produced by the explosion-like expansion of the steam-gas mixture produced in an SCR.

The bubbles of overheated steam would ascend as they grow in size with decreasing hydrostatic pressure. Under such conditions, bubbles would be instable in shape, which would result in their dispersing into smaller bubbles (Ametistov et al., 1982). Ascension of steam bubbles in cold seawater would proceed according to the following scheme: their initial ascent is accompanied by an increase in size, then the steam-gas mixture is cooled down and at certain depth, size growth is swapped for size decrease up to the ultimate collapsing of the bubbles. The collapse depth depends on several parameters - initial temperature of the steam-gas mixture, bubble size and water flux turbulence and temperature.

If the maximum possible SCR occurs at a depth of over 100 m, the gas discharge, according to expert estimates, would not reach the surface owing to complete dissolution in seawater. At lower depths, in particular, at the dumping depth of K-27 (30 m), certain part of the gaseous fission products would be released into the atmosphere.

The maximum possible SCR (10^{20} fissions) would release 3.2 GJ energy. However, upon an SCR event, energy would be released gradually rather than instantaneously - it takes from several seconds to hours. Therefore, only a certain fraction of the released energy would cause the destruction of the protection barriers. The conservative assumption is that this fraction is within 20% of the total energy release, which is equivalent to

640 MJ (equivalent to ~ 160 kg of TNT). In cases of SCRs which results in a situation where the submarine hull breaks down, the gas bubble would disperse into smaller parts.

The rise in water level from an underwater explosion at a depth of 30 m from the maximum SCR would be within the range of 1-2 m. Therefore, near-surface discharges would occur. The bubbles that reach the surface would burst and eject a gas-steam mixture. Thus, radionuclides released into the atmosphere from the reactor during the SCR have the form of aerosol.

The distribution of released radionuclides between water and atmosphere depends on the material volatility and solubility. For example, noble gases are weakly soluble in water. Other nuclides can form easily soluble and insoluble fractions (Ignatiev et al., 2008). As shown in Table 9.6, cesium, iodine, strontium, and antimony belongs to the group which easily form soluble compounds.

Table 9.6. Nuclide release distribution as a function of various physical forms.

Element	Fraction of total content, %		
	ion	Colloid	suspension
Cesium	70	7	23
Iodine	90	8	2
Strontium	87	3	10
Tellurium	45	43	12
Antimony	73	15	12
Molybdenum	30	10	60
Ruthenium	0	5	95
Cerium	2	4	94
Zirconium	1	3	96
Yttrium	0	4	96
Niobium	0	0	100

Thus, in order to account for the dispersion of the underwater discharge, we introduce additional distribution coefficients of the fission products fraction released into the atmosphere and remaining in the water column:

Table 9.7. Atmospheric release fractions for various nuclide groups.

Nuclide groups	Atmosphere release fraction, %
RNG (krypton, xenon)	90
Iodine, bromine	20
Cesium, strontium, antimony	10
Ruthenium, tellurium	5
Other	2

The activities of the short-lived radionuclides released into the atmosphere and remaining in water, respectively, from the instantaneous release upon the occurrence of an SCR with a power of 10^{20} fissions in the starboard reactor while the submarine is located at a depth of 30 m are shown in Tables 9.8 and 9.9. Similarly, the distributions for the long-lived fission products are displayed in Table 9.10.

Table 9.8. Total activities of the short-lived fission products released into the atmosphere from the instantaneous release from the SCR in the K-27 starboard reactor with 10^{20} fission power and the submarine located at the depth of 30 m, Bq.

Time	1 hour	3 hours	12 hours	1 day	7 days	28 days
NRG(Kr, Xe)	2.09E+14	7.70E+13	2.43E+13	1.32E+13	1.49E+12	9.29E+10
Iodine, bromine	3.89E+13	2.17E+13	5.03E+12	2.20E+12	1.33E+11	9.07E+09
Ru, Te	2.53E+12	4.29E+11	3.92E+10	2.01E+10	4.85E+09	7.07E+08
^{140}La	2.86E+07	6.85E+07	2.31E+08	4.07E+08	8.81E+08	3.06E+08
^{140}Ba	1.21E+09	1.21E+09	1.19E+09	1.15E+09	8.36E+08	2.66E+08
^{143}Ce	1.01E+10	1.02E+10	8.44E+09	6.55E+09	3.19E+08	8.07E+03
^{143}Pr	1.55E+07	5.92E+07	2.35E+08	4.18E+08	8.14E+08	2.91E+08
^{141}Ce	4.48E+07	1.62E+08	3.85E+08	4.33E+08	3.89E+08	2.48E+08
^{144}Ce	4.74E+07	4.74E+07	4.74E+07	4.74E+07	4.66E+07	4.40E+07
^{144}Pr	4.59E+07	4.74E+07	4.74E+07	4.74E+07	4.66E+07	4.40E+07
^{147}Nd	5.18E+08	5.40E+08	5.29E+08	5.11E+08	3.51E+08	9.32E+07
^{89}Sr	1.38E+09	1.48E+09	1.47E+09	1.46E+09	1.34E+09	1.01E+09
^{91}Sr	1.71E+11	1.48E+11	7.66E+10	3.21E+10	8.95E+05	1.03E-10
^{91}Y	1.07E+07	3.92E+07	1.39E+08	2.02E+08	2.31E+08	1.81E+08
^{93}Y	2.35E+10	2.06E+10	1.10E+10	4.85E+09	2.48E+05	2.35E-10
^{95}Zr	2.53E+08	2.58E+08	2.57E+08	2.55E+08	2.40E+08	1.91E+08
^{97}Zr	1.89E+10	1.75E+10	1.20E+10	7.36E+09	2.01E+07	2.11E-02
^{95}Nb	1.67E+05	5.88E+05	2.46E+06	4.96E+06	3.20E+07	9.32E+07
Total, Bq	2.51E+14	9.93E+13	2.95E+13	1.54E+13	1.63E+12	1.05E+11
Total, Ci	*6.78E+03*	*2.68E+03*	*7.99E+02*	*4.18E+02*	*4.41E+01*	*2.85E+00*

Table 9.9.Total activities of the short-lived fission products remaining in water from an instantaneous release from the SCR in the K-27 starboard reactor with 10^{20} fission power and the submarine located at the depth of 30 m, Bq.

Time	1 hour	3 hours	12 hours	1 day	7 days	28 days
NRG(Kr, Xe)	2.32E+13	8.55E+12	2.70E+12	1.47E+12	1.65E+11	1.03E+10
Iodine, bromine	1.56E+14	8.66E+13	2.02E+13	8.81E+12	5.33E+11	3.63E+10
Ru,Te	4.81E+13	8.14E+12	7.44E+11	3.81E+11	9.18E+10	1.34E+10
^{140}La	1.40E+09	3.36E+09	1.13E+10	1.99E+10	4.33E+10	1.50E+10
^{140}Ba	5.96E+10	5.96E+10	5.85E+10	5.66E+10	4.11E+10	1.30E+10
^{143}Ce	4.92E+11	5.00E+11	4.14E+11	3.22E+11	1.56E+10	3.96E+05
^{143}Pr	7.59E+08	2.90E+09	1.15E+10	2.04E+10	4.00E+10	1.42E+10
^{141}Ce	2.20E+09	7.96E+09	1.89E+10	2.12E+10	1.90E+10	1.21E+10
^{144}Ce	2.32E+09	2.32E+09	2.31E+09	2.31E+09	2.28E+09	2.16E+09
^{144}Pr	2.25E+09	2.31E+09	2.31E+09	2.31E+09	2.28E+09	2.16E+09
^{147}Nd	2.54E+10	2.65E+10	2.59E+10	2.51E+10	1.72E+10	4.55E+09
^{89}Sr	1.25E+10	1.33E+10	1.32E+10	1.32E+10	1.21E+10	9.10E+09
^{91}Sr	1.54E+12	1.34E+12	6.88E+11	2.89E+11	8.07E+06	9.29E-10
^{91}Y	5.25E+08	1.91E+09	6.81E+09	9.92E+09	1.13E+10	8.88E+09
^{93}Y	1.15E+12	1.01E+12	5.40E+11	2.38E+11	1.21E+07	1.15E-08
^{95}Zr	1.24E+10	1.26E+10	1.26E+10	1.25E+10	1.18E+10	9.36E+09
^{97}Zr	9.25E+11	8.55E+11	5.88E+11	3.61E+11	9.88E+08	1.03E+00
^{95}Nb	8.18E+06	2.88E+07	1.21E+08	2.42E+08	1.57E+09	4.55E+09
Total, Bq	2.32E+14	1.07E+14	2.60E+13	1.20E+13	1.01E+12	1.55E+11
Total, Ci	*6.26E+03*	*2.90E+03*	*7.03E+02*	*3.25E+02*	*2.73E+01*	*4.20E+00*

Table 9.10. Total activities of long-living fission products released into atmosphere and remaining in water from the instantaneous release following an SCR in the K-27 starboard reactor of 10^{20} fission power and the submarine located at a depth of 30 m.

Nuclide	Activity (Bq)	
	Atmosphere	Water
^{79}Se	1.04E+05	5.07E+06
^{85}Kr	2.86E+11	3.18E+10
^{87}Rb	6.96E+00	3.41E+02
^{90}Sr	5.85E+10	5.25E+11
^{90}Y	5.85E+10	5.25E+11
^{93}Zr	6.73E+05	3.29E+07
^{93m} Nb	5.62E+05	2.75E+07
^{107}Pd	3.54E+03	1.73E+05
^{121}Sn	2.02E+06	9.92E+07
^{121m} Sn	2.73E+06	1.34E+08
^{126}Sn	9.03E+04	4.44E+06
^{126}Sb	6.25E+04	5.62E+05
^{126m1}Sb	4.51E+05	4.07E+06
^{129}I	7.40E+05	2.96E+06
^{135}Cs	8.99E+06	8.10E+07
^{137}Cs	2.66E+11	2.39E+12
^{137m} Ba	2.51E+11	2.26E+12
^{147}Pm	1.78E+06	8.73E+07
^{151}Sm	5.37E+08	2.63E+10
^{152}Eu	1.51E+02	7.40E+03
^{155}Eu	2.35E+07	1.15E+09
Total, Bq	9.20E+11	5.76E+12
Total, Ci	*2.49E+01*	*1.56E+02*

As seen from Table 9.10, the main fraction of long-lived radionuclides would remain in seawater. The activity of the long-lived fission-products ^{137}Cs and ^{90}Sr in seawater for an instantaneous discharge at a depth of 30 m exceeds the activity in the atmosphere by an order of magnitude.

10 Concluding remarks

Conducting environmental and health impact assessments requires acquisition of data on the amounts of radioactivity released to the environment, application of various models to simulate the spread and movement of contamination and synthesis of information via exposure estimates and inference on consequences. Two major steps in this process are the estimation of inventory of the submarine and the characterization of the concomitant source term. While the former aims at making an evaluation of the present activity in the K-27 reactors, the latter characterises the quantity and types of released radionuclides (and also their physical and chemical properties) and associated release patterns.

In addition to providing an overview of the existing and available facts and information regarding the Russian nuclear submarine K-27, the current report presents results from the most up-to-date assessment of the inventories of the submarine's reactors. The results indicate that the total inventory of the submarine is about 3.74E+14 Bq (as of the year 2015), of which about 82% is fission products, 8% europium, 7% activation products, 2% tritium and 1% actinides. The report also discusses various release scenarios based on considering different hypothetical accidents including those involving a criticality.

Furthermore, essential inputs for the second part of the study are provided in the present report where the focus will be on the modelling of radionuclide advection and dispersion in the environment and subsequent assessment of doses and associated consequences. The outcomes of the second part of the work will be presented in a new report.

Acknowledgements

The present report is the product of a collaborative effort between the Norwegian Radiation Protection Authority and the Russian Energy Safety Analysis Centre of IBRAE RAN and the Russian Research Centre Kurchatov Institute (KI) under the contract No. M16-13/22. All new calculations and data with regards to characterisation of the source term and investigating the criticality issue along with the development of associated scenarios were conducted by IBRAE and KI. The contribution from our Russian colleagues is gratefully acknowledged. The authors would also like to thank Ingo Harms from the University of Hamburg and Michael Karcher from O.A.Sys Ocean Atmosphere Systems GmbH for their valuable contribution to Chapter 3.

11 Acronyms

CPS	Control and protection system
IASAP	International Arctic Seas Assessment Project
IPPE	State Scientific Center of the Russian Federation – A. I. Leypunsky Institute for Physics and Power Engineering
IWST	Iron-water shielding tank
LMC	Liquid-metal coolant
LTS	Long-term storage
LWST	Lead-water shielding tank
NIKIET	Joint Stock Company “N.A. Dollezhal Research and Development Institute of Power Engineering”
NPF	Nuclear power facility
OKB “Gidropress”	Joint Stock Company “Experimental and Design Organization “Gidropress”
RNG	Radioactive noble gases
SCR	Spontaneous (self-sustained) chain reaction
SFA	Spent fuel assembly
SGU	Steam-generator unit
SNF	Spent nuclear fuel
SPMDB “Malakhite”	Public Corporation “St. Petersburg Marine Design Bureau “Malachite”
SRC	Spent removable core
SRW	Solid radioactive waste

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13 Appendix

Detailed radionuclide inventories associated with reactors and reactors components

The tables presented below display data derived from calculations of the activity of long-lived fission products (Tables A1 and A2), actinides (Tables A3 and A4) and tritium (Tables A5 and A6), europium in control rods (Tables A7 and A8), and steel reactor constructions (Tables A9 and A10) for the port and starboard reactors. Tables A1 and A2, displaying data on fission products, show not only the data for long-lived nuclides, but also data on nuclides with short and medium half-lives, as they are the products of radioactive decay of the respective long-lived isotopes and make a significant contribution to the total activity (primarily these are Ba-137m and Y-90). The tables show data on accumulation and activity of heavy actinides with no regard to ^{235}U and ^{238}U.

Table A1. Activity of long-lived fission products in the portside reactor, Bq.

Nuclide	T ½	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
$_{34}Se^{79}$	6.5E+4 years	3.32E+08	3.32E+08	3.32E+08	3.32E+08	3.32E+08
$_{36}Kr^{85}$	10.72 years	9.95E+11	8.75E+11	6.30E+11	3.31E+11	9.11E+10
$_{37}Rb^{87}$	5.56E+10 years	2.17E+04	2.17E+04	2.17E+04	2.17E+04	2.17E+04
$_{38}Sr^{90}$	29.2 years	3.65E+13	3.47E+13	3.09E+13	2.43E+13	1.52E+13
$_{39}Y^{90}$	64.26 hours	3.65E+13	3.47E+13	3.09E+13	2.43E+13	1.52E+13
$_{40}Zr^{93}$	1.5E+6 years	2.09E+09	2.09E+09	2.09E+09	2.09E+09	2.09E+09
$_{41}Nb^{93m}$	13.6 years	1.75E+09	1.78E+09	2.02E+09	2.46E+09	2.66E+09
$_{46}Pd^{107}$	6.5E+6 years	1.10E+07	1.10E+07	1.10E+07	1.10E+07	1.10E+07
$_{50}Sn^{121}$	27.05 hours	6.30E+09	6.10E+09	5.07E+09	4.89E+09	3.73E+09
$_{50}Sn^{121m}$	52 years	8.50E+09	7.85E+09	7.30E+09	6.30E+09	4.80E+09
$_{50}Sn^{126}$	1.0E+5 years	2.81E+08	2.81E+08	2.81E+08	2.81E+08	2.81E+08
$_{51}Sb^{126}$	12.4 days	3.91E+07	3.91E+07	3.91E+07	3.91E+07	3.91E+07
$_{51}Sb^{126m1}$	19 min	2.81E+08	2.81E+08	2.81E+08	2.81E+08	2.81E+08
$_{53}I^{129}$	1.6+7 years	2.31E+07	2.31E+07	2.31E+07	2.31E+07	2.31E+07
$_{55}Cs^{135}$	2.3E+6 years	1.40E+09	1.40E+09	1.40E+09	1.40E+09	1.40E+09
$_{55}Cs^{137}$	30 years	4.14E+13	3.97E+13	3.53E+13	2.81E+13	1.77E+13

Nuclide	T ½	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
$_{56}Ba^{137m}$	2.552 min	3.91E+13	3.72E+13	3.34E+13	2.65E+13	1.68E+13
$_{61}Pm^{147}$	2.623 years	5.56E+09	3.85E+09	8.95E+08	6.40E+07	3.28E+05
$_{62}Sm^{151}$	90 years	1.67E+12	1.64E+12	1.59E+12	1.46E+12	1.24E+12
$_{63}Eu^{152}$	13.2 years	4.69E+05	4.01E+05	3.08E+05	1.83E+05	6.45E+04
$_{63}Eu^{155}$	4.68 years	7.30E+10	5.45E+10	2.60E+10	5.85E+09	2.99E+08
TOTAL, Bq		1.56E+14	1.49E+14	1.38E+14	1.05E+14	6.60E+14
TOTAL, kCi		*4.22E+00*	*4.03E+00*	*3.73E+00*	*2.84E+00*	*1.78E+00*

Table A2. Activity of long-lived fission products in the starboard reactor, Bq.

Nuclide	T ½	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
$_{34}Se^{79}$	6.5E+4 years	3.45E+08	3.45E+08	3.45E+08	3.45E+08	3.45E+08
$_{36}Kr^{85}$	10.72 years	1.06E+12	9.35E+11	6.74E+11	3.54E+11	9.74E+10
$_{37}Rb^{87}$	5.56E+10 years	2.32E+04	2.32E+04	2.32E+04	2.32E+04	2.32E+04
$_{38}Sr^{90}$	29.2 years	3.90E+13	3.71E+13	3.30E+13	2.60E+13	1.63E+13
$_{39}Y^{90}$	64.26 hours	3.90E+13	3.71E+13	3.30E+13	2.60E+13	1.63E+13
$_{40}Zr^{93}$	1.5E+6 years	2.24E+09	2.24E+09	2.24E+09	2.24E+09	2.24E+09
$_{41}Nb^{93m}$	13.6 years	1.87E+09	1.90E+09	2.16E+09	2.63E+09	2.84E+09
$_{46}Pd^{107}$	6.5E+6 years	1.18E+07	1.18E+07	1.18E+07	1.18E+07	1.18E+07
$_{50}Sn^{121}$	27.05 hours	6.74E+09	6.52E+09	5.42E+09	5.23E+09	3.99E+09
$_{50}Sn^{121m}$	52 years	9.09E+09	8.40E+09	7.81E+09	6.73E+09	5.13E+09
$_{50}Sn^{126}$	1.0E+5 years	3.01E+08	3.01E+08	3.01E+08	3.01E+08	3.01E+08
$_{51}Sb^{126}$	12.4 days	4.18E+07	4.18E+07	4.18E+07	4.18E+07	4.18E+07
$_{51}Sb^{126m1}$	19 min	3.01E+08	3.01E+08	3.01E+08	3.01E+08	3.01E+08
$_{53}I^{129}$	1.6+7 years	2.47E+07	2.47E+07	2.47E+07	2.47E+07	2.47E+07

Nuclide	T ½	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
$_{55}Cs^{135}$	2.3E+6 years	1.50E+09	1.50E+09	1.50E+09	1.50E+09	1.50E+09
$_{55}Cs^{137}$	30 years	4.43E+13	4.25E+13	3.78E+13	3.01E+13	1.89E+13
$_{56}Ba^{137m}$	2.552 min	4.18E+13	3.98E+13	3.57E+13	2.83E+13	1.80E+13
$_{61}Pm^{147}$	2.623 years	5.95E+09	4.12E+09	9.57E+08	6.84E+07	3.51E+05
$_{62}Sm^{151}$	90 years	1.79E+12	1.75E+12	1.70E+12	1.56E+12	1.33E+12
$_{63}Eu^{152}$	13.2 years	5.02E+05	4.29E+05	3.29E+05	1.96E+05	6.90E+04
$_{63}Eu^{155}$	4.68 years	7.81E+10	5.83E+10	2.78E+10	6.26E+09	3.20E+08
TOTAL, Bq		1.67E+14	1.59E+14	1.48E+14	1.12E+14	7.06E+14
TOTAL, kCi		*4.51E+00*	*4.30E+00*	*3.99E+00*	*3.03E+00*	*1.91E+00*

Table A3. Activity of actinides in the portside reactor, Bq

Nuclide	T ½ ,years	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
U^{234}	2.48E+11	8.23E+09	8.23E+09	8.23E+09	8.23E+09	8.23E+09
U^{236}	2.34E+07	6.93E+08	6.93E+08	6.93E+08	6.93E+08	6.93E+08
Np^{237}	4.14E+06	6.08E+08	6.08E+08	6.08E+08	6.08E+08	6.08E+08
Pu^{238}	87.75	9.35E+11	9.31E+11	8.62E+11	8.17E+11	6.98E+11
Pu^{239}	24380	8.50E+10	8.50E+10	8.50E+10	8.50E+10	8.49E+10
Pu^{240}	6537	2.04E+10	2.04E+10	2.04E+10	2.04E+10	2.03E+10
Pu^{241}	14.54	3.95E+11	3.59E+11	2.84E+11	1.76E+11	6.75E+10
Am^{241}	432.8	1.36E+11	1.36E+11	1.36E+11	1.35E+11	1.34E+11
Am^{242m}	152	4.23E+08	4.19E+08	4.09E+08	3.89E+08	3.58E+08
TOTAL, Bq		1.58E+12	1.54E+12	1.40E+12	1.24E+12	1.02E+12
TOTAL, kCi		*4.27E-02*	*4.17E-02*	*3.78E-02*	*3.36E-02*	*2.74E-02*

Table A4. Activity of actinides in the starboard reactor, Bq.

Nuclide	$T_{½}$,years	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
U^{234}	2.48E+11	8.80E+09	8.80E+09	8.80E+09	8.80E+09	8.80E+09
U^{236}	2.34E+07	7.41E+08	7.41E+08	7.41E+08	7.41E+08	7.41E+08
Np^{237}	4.14E+06	6.50E+08	6.50E+08	6.50E+08	6.50E+08	6.50E+08
Pu^{238}	87.75	9.99E+11	9.95E+11	9.22E+11	8.74E+11	7.49E+11
Pu^{239}	24380	9.09E+10	9.09E+10	9.09E+10	9.09E+10	9.08E+10
Pu^{240}	6537	2.18E+10	2.18E+10	2.18E+10	2.18E+10	2.17E+10
Pu^{241}	14.54	4.22E+11	3.84E+11	3.04E+11	1.88E+11	7.22E+10
Am^{241}	432.8	1.45E+11	1.45E+11	1.45E+11	1.44E+11	1.43E+11
Am^{242m}	152	4.52E+08	4.48E+08	4.37E+08	4.16E+08	3.83E+08
TOTAL, Bq		1.69E+12	1.65E+12	1.50E+12	1.33E+12	1.09E+12
TOTAL, kCi		*4.57E-02*	*4.45E-02*	*4.05E-02*	*3.58E-02*	*2.95E-02*

Table A5. Activity of tritium in the portside reactor, Bq.

Components	Activity as of the end of the respective year				
	2013	2015	2020	2030	2050
Core	3.61E+12	3.22E+12	2.44E+12	1.39E+12	4.51E+11
Side deflector	5.50E+11	4.90E+11	3.71E+11	2.12E+11	6.87E+10
Upper butt reflector*	1.08E+11	9.63E+10	7.29E+10	4.16E+10	1.35E+10
Lower butt reflector*	1.73E+11	1.54E+11	1.17E+11	6.66E+10	2.16E+10
CPS rods	5.02E+10	4.48E+10	3.39E+10	1.93E+10	6.27E+09
TOTAL, Bq	4.49E+12	4.01E+12	3.03E+12	1.73E+12	5.61E+11
TOTAL, kCi	*1.22E-01*	*1.08E-01*	*8.20E-02*	*4.70E-02*	*1.52E-02*

Note: Upper and lower butt reflectors are the parts of fuel rods*

Table A6. Activity of tritium in the starboard reactor, Bq.

Components	Activity as of the end of the respective year				
	2013	2015	2020	2030	2050
Core	3.86E+12	3.45E+12	2.61E+12	1.49E+12	4.82E+11
Side deflector	5.88E+11	5.25E+11	3.97E+11	2.26E+11	7.35E+10
Upper butt reflector*	1.16E+11	1.03E+11	7.79E+10	4.45E+10	1.44E+10
Lower butt reflector*	1.86E+11	1.65E+11	1.25E+11	7.12E+10	2.30E+10
CPS rods	5.37E+10	4.80E+10	3.63E+10	2.07E+10	6.71E+09
TOTAL, Bq	4.81E+12	4.28E+12	3.25E+12	1.85E+12	6.01E+11
TOTAL, kCi	*1.30E-01*	*1.16E-01*	*8.90E-02*	*5.00E-02*	*1.60E-02*

Note: Upper and lower butt reflectors are the parts of fuel rods*

Table A7. Activity of europium in CPS rods of the portside reactor, Bq.

Nuclide	T ½,years	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
Eu^{152}	13.53	1.33E+13	1.20E+13	9.29E+12	5.57E+12	1.99E+12
Eu^{154}	8.59	1.47E+12	1.16E+12	7.78E+11	3.49E+11	6.94E+10
TOTAL, Bq		1.48E+13	1.32E+13	1.01E+12	5.92E+12	2.06E+12
TOTAL, kCi		*4.01E-01*	*3.57E-01*	*2.73E-01*	*1.60E-01*	*5.60E-02*

Table A8. Activity of europium in CPS rods of the starboard reactor, Bq.

Nuclide	T ½,years	Activity as of the end of the respective year				
		2013	2015	2020	2030	2050
Eu^{152}	13.53	1.42E+13	1.28E+13	9.93E+12	5.96E+12	2.13E+12
Eu^{154}	8.59	1.57E+12	1.24E+12	8.32E+11	3.73E+11	7.42E+10
TOTAL, Bq		1.58E+13	1.41E+13	1.07E+12	6.33E+12	2.20E+12
TOTAL, kCi		*4.27E-01*	*3.81E-01*	*2.89E-01*	*1.71E-01*	*5.90E-02*

Table A9. Activity of steel constructions of the portside reactor, Bq.

Components	Nuclide	$T_{1/2}$, years	Activity as of the end of the respective year				
			2013	2015	2020	2030	2050
Fuel cladding (340 kg)	Ni^{59}	7.5E+04	1.26E+09	1.26E+09	1.26E+09	1.26E+09	1.26E+09
	Ni^{63}	100.1	8.26E+10	8.19E+10	7.89E+10	7.29E+10	6.40E+10
	Co^{60}	5.27	6.60E+09	5.08E+09	2.63E+09	7.06E+08	5.08E+07
	Fe^{55}	2.72	3.46E+07	2.08E+07	5.81E+06	4.56E+05	2.77E+03
	TOTAL, Bq		9.04E+10	8.82E+10	8.27E+10	7.49E+10	6.53E+10
	TOTAL, kCi		*2.45E-03*	*2.38E-03*	*2.23E-03*	*2.02E-03*	*1.76E-03*
Fuel assembly grids (450 kg)	Ni^{59}	7.5E+04	8.34E+08	8.34E+08	8.34E+08	8.34E+08	8.34E+08
	Ni^{63}	100.1	5.45E+10	5.40E+10	5.21E+10	4.81E+10	4.22E+10
	Co^{60}	5.27	4.36E+09	3.36E+09	1.73E+09	4.66E+08	3.36E+07
	Fe^{55}	2.72	2.29E+07	1.38E+07	3.85E+06	3.02E+05	1.83E+03
	TOTAL, Bq		5.97E+10	5.82E+10	5.47E+10	4.94E+10	4.31E+10
	TOTAL, kCi		*1.61E-03*	*1.57E-03*	*1.48E-03*	*1.34E-03*	*1.16E-03*
Ferrule of removable core (250 kg) (from the bottom line to outlet fitting)	Ni^{59}	7.5E+04	4.62E+08	4.62E+08	4.62E+08	4.62E+08	4.62E+08
	Ni^{63}	100.1	3.03E+10	3.00E+10	2.89E+10	2.67E+10	2.22E+10
	Co^{60}	5.27	2.42E+09	1.86E+09	9.64E+08	2.59E+08	1.86E+07
	Fe^{55}	2.72	1.07E+07	6.43E+06	1.79E+06	1.41E+05	8.57E+02
	TOTAL, Bq		3.32E+10	3.23E+10	3.03E+10	2.74E+10	2.27E+10
	TOTAL, kCi		*8.96E-04*	*8.74E-04*	*8.19E-04*	*7.41E-04*	*6.13E-04*

Components	Nuclide	$T_{1/2}$, years	Activity as of the end of the respective year				
			2013	2015	2020	2030	2050
Steel side reflector (2000 kg)	Ni^{59}	7.5E+04	1.53E+11	1.53E+11	1.53E+11	1.53E+11	1.53E+11
	Ni^{63}	100.1	1.03E+13	1.02E+13	9.84E+12	9.09E+12	7.98E+12
	Co^{60}	5.27	8.02E+11	6.17E+11	3.19E+11	8.58E+10	6.17E+09
	Fe^{55}	2.72	4.40E+10	2.64E+10	7.39E+09	5.80E+08	3.52E+06
	TOTAL, Bq		1.13E+13	1.10E+13	1.03E+13	9.33E+12	8.14E+12
	TOTAL, kCi		*3.05E-01*	*2.97E-01*	*2.79E-01*	*2.52E-01*	*2.20E-01*
Reactor shell (2100 kg) (bottom and side part from the bottom to outlet fitting)	Ni^{59}	7.5E+04	3.21E+10	3.21E+10	3.21E+10	3.21E+10	3.21E+10
	Ni^{63}	100.1	2.10E+12	2.08E+12	2.01E+12	1.85E+12	1.63E+12
	Co^{60}	5.27	1.68E+11	1.29E+11	6.69E+10	1.80E+10	1.29E+09
	Fe^{55}	2.72	8.82E+08	5.30E+08	1.48E+08	1.16E+07	7.06E+04
	TOTAL, Bq		2.31E+12	2.24E+12	2.11E+12	1.90E+12	1.66E+12
	TOTAL, kCi		*6.22E-02*	*6.06E-02*	*5.70E-02*	*5.14E-02*	*4.50E-02*
TOTAL, Bq			1.36E+13	1.34E+13	1.26E+13	1.13E+13	9.89E+12
TOTAL, kCi			*3.69E-01*	*3.63E-01*	*3.39E-01*	*3.05E-01*	*2.67E-01*

Table A10. Activity of steel constructions of the starboard reactor, Bq.

Components	Nuclide	$T_{1/2}$, years	Activity as of the end of the respective year				
			2013	2015	2020	2030	2050
Fuel cladding (340 kg)	Ni^{59}	7.5E+04	1.35E+09	1.35E+09	1.35E+09	1.35E+09	1.35E+09
	Ni^{63}	100.1	8.83E+10	8.76E+10	8.44E+10	7.79E+10	6.84E+10
	Co^{60}	5.27	7.06E+09	5.43E+09	2.81E+09	7.55E+08	5.43E+07
	Fe^{55}	2.72	3.70E+07	2.22E+07	6.21E+06	4.88E+05	2.96E+03
	TOTAL, Bq		9.67E+10	9.43E+10	8.84E+10	8.01E+10	6.98E+10
	TOTAL, kCi		*2.62E-03*	*2.55E-03*	*2.38E-03*	*2.16E-03*	*1.88E-03*

Components	Nuclide	$T_{1/2}$, years	Activity as of the end of the respective year				
			2013	2015	2020	2030	2050
Fuel assembly grids (450 kg)	Ni^{59}	7.5E+04	8.92E+08	8.92E+08	8.92E+08	8.92E+08	8.92E+08
	Ni^{63}	100.1	5.83E+10	5.77E+10	5.57E+10	5.14E+10	4.51E+10
	Co^{60}	5.27	4.66E+09	3.59E+09	1.85E+09	4.98E+08	3.59E+07
	Fe^{55}	2.72	2.45E+07	1.48E+07	4.12E+06	3.23E+05	1.96E+03
	TOTAL, Bq		6.38E+10	6.22E+10	5.85E+10	5.28E+10	4.61E+10
	TOTAL, kCi		*1.72E-03*	*1.68E-03*	*1.58E-03*	*1.43E-03*	*1.24E-03*
Ferrule of removable core (250 kg) (from the bottom line to outlet fitting)	Ni^{59}	7.5E+04	4.94E+08	4.94E+08	4.94E+08	4.94E+08	4.94E+08
	Ni^{63}	100.1	3.24E+10	3.21E+10	3.09E+10	2.86E+10	2.37E+10
	Co^{60}	5.27	2.59E+09	1.99E+09	1.03E+09	2.77E+08	1.98E+07
	Fe^{55}	2.72	1.14E+07	6.88E+06	1.91E+06	1.51E+05	9.17E+02
	TOTAL, Bq		3.55E+10	3.45E+10	3.24E+10	2.93E+10	2.43E+10
	TOTAL, kCi		*9.58E-04*	*9.34E-04*	*8.76E-04*	*7.92E-04*	*6.55E-04*
Steel side reflector (2000 kg)	Ni^{59}	7.5E+04	1.57E+11	1.57E+11	1.57E+11	1.57E+11	1.57E+11
	Ni^{63}	100.1	1.06E+13	1.05E+13	1.01E+13	9.34E+12	8.19E+12
	Co^{60}	5.27	8.23E+11	6.33E+11	3.28E+11	8.81E+10	6.34E+09
	Fe^{55}	2.72	4.52E+10	2.71E+10	7.59E+09	5.96E+08	3.62E+06
	TOTAL, Bq		1.16E+13	1.13E+13	1.06E+13	9.58E+12	8.36E+12
	TOTAL, kCi		*3.14E-01*	*3.05E-01*	*2.86E-01*	*2.59E-01*	*2.26E-01*
Reactor shell (2100 kg) (bottom and side part from the bottom to outlet fitting)	Ni^{59}	7.5E+04	3.30E+10	3.30E+10	3.30E+10	3.30E+10	3.30E+10
	Ni^{63}	100.1	2.16E+12	2.14E+12	2.06E+12	1.89E+12	1.67E+12
	Co^{60}	5.27	1.72E+11	1.32E+11	6.87E+10	1.85E+10	1.32E+09
	Fe^{55}	2.72	9.06E+08	5.44E+08	1.52E+08	1.19E+07	7.25E+04
	TOTAL, Bq		2.37E+12	2.30E+12	2.17E+12	1.95E+12	1.70E+12
	TOTAL, kCi		*6.39E-02*	*6.22E-02*	*5.09E-02*	*5.29E-02*	*4.62E-02*
TOTAL, Bq			1.41E+13	1.38E+13	1.30E+13	1.17E+13	1.02E+13
TOTAL, kCi			*3.82E-01*	*3.74E-01*	*3.53E-01*	*3.15E-01*	*2.75E-01*

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Årsrapport 2014

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ISSN 1891-5191 (online)
ISSN 0804-4910 (print)