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PROJECT 'PURCHASE OF STUDIES FOR THE PREPARATION OF A DESIGNATED SPATIAL PLAN AND THE ASSESSMENT OF IMPACT'

PROJECT PART: STUDIES FOR THE ELIMINATION OF THE REACTOR COMPARTMENTS

ACTIVITIES 4-5. Studies for the elimination (decommissioning) of the reactor compartments. Comparison of reactor compartment decommissioning alternatives. Activities related to the most suitable reactor compartment decommissioning alternative

FINAL REPORT

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CONTENT

ABBREVIATIONS

CA	Control Area
DTM	Difficult-To-Measure
ECL	Exclusion and Clearance Level
ETM	Easy-To-Measure
EU	European Union
FPNC	Former Paldiski Nuclear Center
IAEA	International Atomic Energy Agency
MAUT	Multi-Attribute Utility Theory
MB	Main Building of FPNC
MB&IS	MB and Interim Storage
NFP	Nuclear Fuel Pool
NV	Nuclide Vector
PGC	Personnel Dosimetric Control
RC	Reactor Compartment
RRL	Reuse or Recycling Level
RW	Radioactive Waste
SAR	Safety Analysis Report
SNF	Spent Nuclear Fuel
SRW	Solid RW
TLD	Thermoluminescence Dosimeter

INTRODUCTION

This is the final report intended to summarise the results of the studies performed in Activity 4 “Studies for the elimination (decommissioning) of the reactor compartments”, including sub-activities 4.1 – 4.11, Activity 5, “Comparison of reactor compartment decommissioning alternatives. Activities related to the most suitable reactor compartment decommissioning alternative”, including sub-activities 5.1 – 5.3, and sub-activity 2.24 “Additional assessment of the quantity of radioactive waste and additional characterisation of waste”, which is a part of Activity 2 “Studies of the three repository locations”, which were envisaged in the Technical Specifications of the Project “’Purchase of studies for the preparation of a designated spatial plan and the assessment of impact”.

The sub-activities included the following studies:

Sub-activity 4.1: the study carried out to provide assessment concerning the construction of the MB of the FPNC, to determine the condition of the heating, ventilation, plumbing and electrical systems of the building under consideration and to assess its potential for future use as well as to assess the compliance of the building with the essential requirements for fire safety (see Appendix 1).

Sub-activity 4.2: the study carried out to provide assessment concerning the building materials and structure of the interim radioactive waste storage facility of FPNC (see Appendix 2).

Sub-activity 4.3: the radiological study of the MB of FPNC (see Appendix 3).

Sub-activity 4.4: the radiological study of RCs of FPNC (see Appendix 4).

Sub-activity 4.5: the study carried out to provide assessment concerning the construction of the reactor sarcophagi and RCs of FPNC, to determine the condition of the heating, ventilation, plumbing and electrical systems of the structures under consideration and to assess its potential for the future (see Appendix 5).

Sub-activity 4.6: the radiological study of the Paldiski site (see Appendix 6).

Sub-activity 4.7: the study of the development of 3D models of RCs of FPNC (see Appendix 7).

Sub-activity 4.8: the study of the safety assessment of the decommissioning of two naval training reactors that are installed at FPNC. Safety assessment presents the MB decommissioning technology, its internal structures, systems and components, including RCs, as well as the safety analysis of this technology (see Appendix 8).

Sub-activity 4.9: the study is intended for preparation of monitoring programmes for the decommissioning of the RCs. It includes monitoring programmes to ensure the radiation safety of personnel as well as the public and the environment during the MB decommissioning (see Appendix 9).

Sub-activity 4.10: the study of the risk analysis and assessment for decommissioning of RCs at FPNC was performed in order to: identify emergencies and assess their consequences for decommissioning of the RCs; identify preventive measures for diversification of risks (see Appendix 10).

Sub-activity 4.11: the study carried out to provide an assessment of the transboundary transport of radionuclides that may enter the atmosphere as a result of radiation accidents during the decommissioning of RCs that are installed at FPNC and aims to prove the radiation safety of the planned work (see Appendix 11).

Sub-activity 5.1: the study carried out to provide an analysis based on economic, radiation safety and other such aspects of the “zero alternative”, i.e. preservation of mothballed RCs and storage of the packaged RW in the interim storage at the FPNC, and give an overview of the possible drawbacks and advantages entailed in the implementation of the “zero alternative” compared to the decommissioning of the RCs (see Appendix 12).

Sub-activity 5.2: the study carried out to present an initial analysis based on economic, radiation safety and other such aspects of the RC decommissioning alternatives and provide an overview of the possible drawbacks and advantages entailed in the implementation of different decommissioning alternatives (see Appendix 13).

Sub-activity 5.3: the study carried out for preparation of a decommissioning plan and a safety case for the most suitable RC decommissioning alternative (see Appendix 14).

Sub-activity 2.24: the study carried out for the additional assessment of the quantity of RW and the additional characterisation of the waste generated from the operation of the reactors (steam generators and concreted sediments of liquid waste); the boxes containing radiation sources which were removed from the Tammiku RW storage; the concrete containers that contain the control rods of the reactors; and the area contaminated around a former special sewerage well located in the territory of the Paldiski site (see Appendix 15).

1. ENGINEERING STUDIES

1.1 Engineering Study of the Condition of the Main Building of the Paldiski Site

1.1.1 General information

The engineering study of the condition of the MB was done within sub-activity 4.1.

As it stands, the MB can be divided into two parts. The designed dimensions of the storage block of the MB are 140 × 20 m, plus an administrative-non-work block with designed dimensions of 90 × 12 m. The MB was part of a land-based training centre for the Soviet Navy's nuclear submarine crews, the construction work of which began in the early 1960s and which was equipped with educational stands simulating a nuclear submarine and a working nuclear reactor. In April 1968, an educational stand of the first-generation Soviet nuclear submarine simulator with a working nuclear reactor was launched (Echo II Class, Project 658). In 1983, a stand with the second-generation reactor was launched (Delta I-IV class, project 667).

The building was partially demolished and the perimeter structures were rebuilt. The MB currently houses 2 sarcophagi built around the reactor sections and an interim storage facility for RW.

The building was constructed in several stages, and in its current form two stages can be clearly distinguished: the old part of the building (also referred to as Building No. 301), which was designed in 1963 and completed a few years later, and the new part of the MB (also referred to as Building No. 302), which was designed in 1974 and completed a few years later (See Fig. 1.1.1, Fig. 1.1.2). In its current state, the use of the building is intended to ensure that the sarcophagi and the interim storage facility are protected and isolated from the weather impact. In addition, there are various rooms for staff and for processing radioactive waste in the building. The MB is used all year-round, although there is no indoor climate control in the MB.

Fig. 1.1.1 General view of the MB

Fig. 1.1.2 Scheme of the main building (new (302) and old (301) parts)

1.1.2 Geological and hydrogeological conditions

The subsoil on the Pakri Peninsula is limestone. Limestone's uniaxial strength is measured to be 56.8 MPa, which makes it comparable to concrete. Therefore, this limestone makes an excellent base for foundations. Geotechnical surveys from the locations nearby suggest that the limestone is positioned quite high, lying 1.00 ÷ 2.30 m below ground level. Ground water horizon, on the other hand, is measured 2 ÷ 12 m below ground level, the highest value being measured at springtime, when the groundwater is naturally higher.

1.1.3 Previously made renovations and reconstructions

The building has been partially demolished over time and the perimeter structures have been rebuilt. Mostly the surrounding parts of the building have been demolished, the storage block (MB) has been left intact. The MB in the southeastern part of the building has also been shortened by eight axle spans, i.e. by about 48 m. In this way, the optimal size for the intended use of the building has been achieved. The administrative-non-work block, which has remained at the moment, was originally also higher, and its height has also been reduced to an optimal size. The perimeter structures of the building were rebuilt. The remaining walls have been insulated and finished with sheet metal cladding, and a pitched sheet metal roof with external drainage has been built on the original roof with internal drainage. Since some parts of the building have been demolished, only the walls common to the remaining part of the building have been preserved, special metal elements have been installed to increase the stability of the brick walls.

1.1.4 Building structures

Overall structural solution

The load-bearing frame of the building is made up of frames with a pitch of 6 m, consisting of roof girders and double stem reinforced concrete posts, most likely installed in the socket-type footing. The roof girders are roof beams 1 Б8-18-2 (or analogue) with an 18 m span, supporting various roof ceiling panels ПНС-1/3 x6 and ПНС-10/1.5 x6 (or analogues). The geometrical design of the frame of the industrial building is based on the principle that all elements of the load-bearing structure – posts, beams and roof panels – remain within a rectangle bordered by vertical modular planes. As a result, the posts of the building in question are positioned in two different ways in relation to the axes: along the length of the building, the outermost posts are positioned with one edge on the axis (the so-called “zero binding”) and the posts on the short side of the building are offset by 50 cm from the axis. This design makes it possible to freely join the building blocks and thus to design buildings of unlimited span, the load-bearing structure of which is divided into several temperature blocks.

The transverse stiffness of the building is provided by rigid frames formed by beams and posts, the stiffness being achieved by rigid and moment-resistant fastening of the posts to the foundation. Also, the section modulus of the post in the transverse direction is significantly higher than the section modulus in the longitudinal direction, due to the stems placed at a distance in relation to one another. The overall longitudinal rigidity of the building, i.e. in the transversal direction of the roof beams, is ensured by diagonal terrace ties and reinforced concrete ties along the length of the building. There are records that it is an industrial building with a strengthened frame, designed to withstand earthquakes of up to 6 on the Richter scale.

Types and sizes of effective loads

The loads on a building are divided into permanent loads and variable loads according to their duration.

The permanent loads on the walls and the original roof ceiling have been slightly increased due to the weatherproofing of the building, which involved the construction of a pitched roof over the flat roof and the covering of the walls with insulation material and sheet metal. However, the additional dead loads are relatively small. Variable loads are mainly applied to the floors and, through the crane, also to the crane tracks and posts. The lifting capacity of the cranes is currently limited to 30 tonnes, but it would be possible to restore it to 50 tonnes as it was initially. In terms of snow load, the Pakri Peninsula is located in an area where the normative snow load on the ground is 1.5 kN/m^2. However, as the crane is extremely slow moving, it would be more likely to be classified as a static load.

According to the information from the operator of the building, in practice there is no snow on the roof in most cases, because the combination of an open space and a high building means that the wind carries the snow away from the roof. However, the combination of an open space and a high building means that the building is located in terrain type 0, and with a building height of about 20 m, the wind velocity at 21 m/s at base velocity is about 1 kN/m^2.

Building materials used

The building was constructed using a variety of materials available at the time, but mainly reinforced concrete, in the first place reinforced concrete elements produced in the factory. It can be speculated that, since the building was built for military purposes from the beginning, it was made of stronger concrete than used to manufacture standard catalogue elements. In the building also cast-in-place concrete has been used, for example, to build floors and the necessary foundations for the reactor sections. As external walls made of lightweight concrete panels made of slag concrete have been used. Steel has been used for the overall rigidity of the building, for the ties, walkways and stairs ladders, as well as for the load-bearing structure of the new pitched roof.

The most common structural elements of the walls are (see Fig. 1.1.3):

- Double stem post;
- External wall panel;
- Intermediate floor panel/tie.

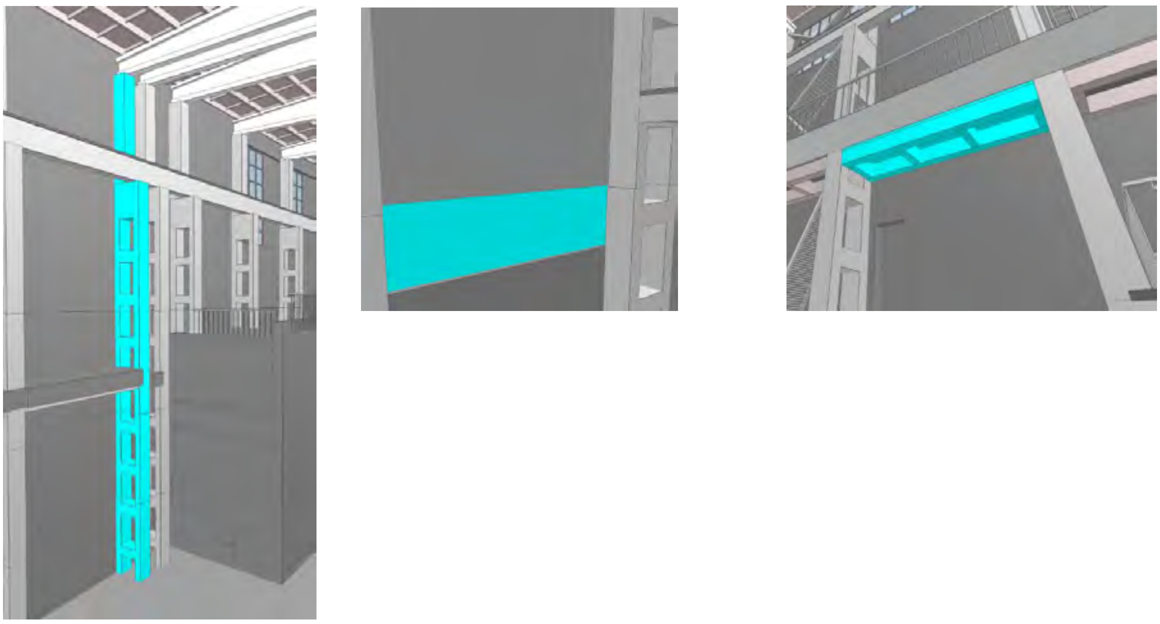


Fig. 1.1.3 Construction elements of the MB walls

The most common structural elements of the roof are (see Fig. 1.1.4):

- Roof ceiling beam (18 m span roof beams 1 Б8-18-2 or analogue);
- Roof ceiling panel (ПНС-10/1.5 x6 or their analogues).

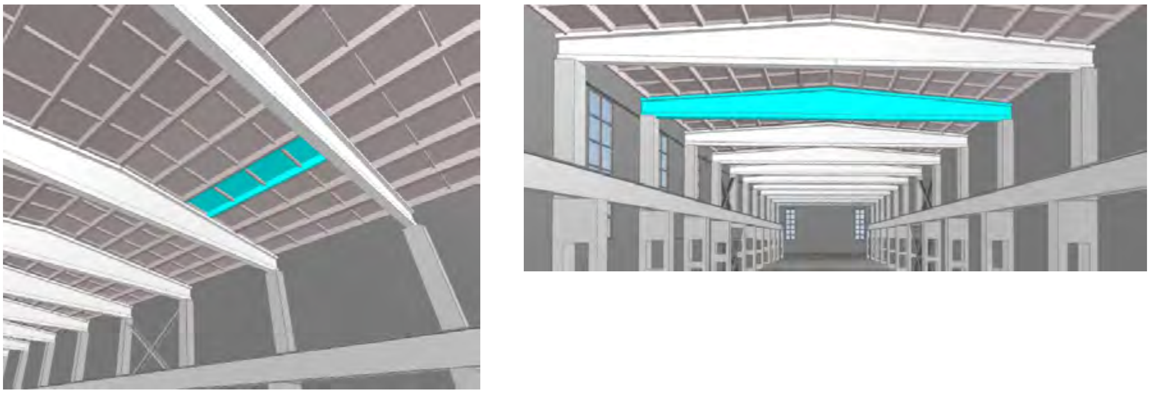


Fig. 1.1.4 Construction elements of the MB roof

Assessing of the structures

All structures were observable from the inside, but could not be observed from the outside. Assessing of the structures was done by visual inspection, the non-destructive testing on site and destructive testing in the laboratory. The following equipment was used for testing on site:

- Photo camera;
- Hilti PS 1000 X-Scan concrete scanner;
- Silver Schmidt OS8200;
- Phenolphtaleine 1% solution;
- ASR Detect I-AS-3000;
- Proceq Profoscope+
- Proceq Resipod;
- Proceq Pundit Lab Ultrasonic Instrument;
- Proceq Equotip Piccolo 2.

The following equipment was used for testing in the laboratory:

- Hydraulic press.

Visual inspection combined with non-destructive testing on site and destructive testing in the laboratory provide a good assessment reliability, but the fact that not all sides of the structure were accessible, and therefore visible, will not decrease the overall reliability, since this situation mainly affected the external wall elements and masonry structures, while the main load-bearing structures were accessible and visible.

Construction-technical condition of the structures

The posts and roof beams of the building are in good condition. There is no significant damage. Roof ceiling panels are in a slightly worse condition, and external wall panels are in the worst condition, the visual inspection of the panels revealed some areas where the reinforcement is corroding.

Hooks made of steel profiles for attaching the two-stem post of external wall panels to the anchor plates were covered with a layer of rust on top, but there was no penetrating rust. However, plaster tell-tales were installed in three places to make sure that the external loads would not yield in the joint.

Wall panels are the largest source of problems. The concrete is of very uneven quality there, and the surface is cracked in places. These wall elements do not play a very important role in terms of the load-bearing capacity of the building as a whole, but the crumbling of the protective layer can primarily be dangerous for the occupants of the buildings. Due to existing damage, the residual life of several panels may not exceed 10 years. However, it must be taken into account that it is very likely that this damage has largely occurred already in the period that preceded the last major repair, when extensive leaks occurred.

Also, some roof-ceiling panels have visible damage. They are not extensive and large. They originate from the time before the last leaks occurred. For example, around the in-building drainage pipes and roof outlets. At the same time, the load condition of the roof ceiling panels has become significantly more favourable, due to the fact that they are no longer affected by snow load, since the base structure of the pitched roof allows the load to concentrate on the roof ceiling beams.

There is salt and moisture damage in the masonry, which also dates back to the time before repairs and demolition works, when the building still had adjacent parts. These joining parts (to the building described in this audit) were likely the places of leaks. Plaster has fallen off from the walls and the compressive strength of the masonry mixture has decreased. The situation has been remedied by installing additional anchors to the frame posts.

There were no signs visible of foundation sinking, but given that the foundations are resting on limestone, this is rather expected to find, since consolidation of soil cannot occur.

Building envelope and energy efficiency

Since a new roof has been installed on the entire building, and the outer walls are covered with new materials, it can be said that the building has a new building envelope, which is now more than 15 years old. These renovations have significantly reduced and even eliminated the impact of certain weather effects and loads on the original building envelope:

- Better load conditions of the roof ceiling panels due to the fact that they are no longer affected by snow load;
- No rain effect on the original flat roof;
- Contemporary windows and doors;

The building does not meet modern energy efficiency requirements, but is suitable for fulfilling the goals set for it.

Construction-technical condition of the building envelope

Although the exterior of the building is covered with recently manufactured modern materials, there are still certain problems. For example, there is a hole in the roof that needs to be patched. Only the top layer of the roof screws is rusted, the remaining part of the screws are not rusted. There is a problem with air leaks around the opening fillings and from the corner of the floor and wall. There are also places in the structure where there are defects in the insulation. In a few places in the administrative block, care must be taken to avoid mould forming on the inner surfaces of the railings. Mould harmful effects present danger for people who are working in those rooms, but it doesn’t affect concrete properties.

1.1.5 Heating, ventilation, water and sewerage systems

Heating and ventilation systems

The heating of this part of the building is designed with electric radiators. There are 11 ventilation systems, two of which are air intake systems for the non-work rooms, two are exhaust systems for the non-work rooms and the rest are various local exhaust systems.

According to the design, a ventilation system with a plate heat exchanger was foreseen for some of the non-work rooms, but a separate inlet fan with an electric calorifier and a separate exhaust fan have been installed, with no heat recovery from the exhaust air to the inlet air.

The ventilation systems in the non-work area are equipped with speed regulators.

The pipework is constructed of galvanised spiral rolled steel pipes. Perforated steel tape is used as hangers. During a visual inspection, it was detected that the fastenings of the exhaust piping on the facade of the building are heavily corroded.

Air ducts of the wall penetrations in the ventilation chamber are mostly fitted with fire dampers, however, in some pipes they are missing. The fireproofing of ventilation ducts and the sealing of penetrations is inadequate and does not comply with current requirements.

Piping in the ventilation chambers and the cold attic is insulated with glass wool insulation covered with foil. Insulation joints are mostly taped with grey building tape unsuitable for this purpose, the adhesive of which has come loose in many places, causing the insulation to come loose around the pipes or fall off altogether. Poor insulation quality and loose insulation result in considerable heat losses. In the current situation, the insulation of the ventilation ducts does not comply with current requirements.

According to the information received from the personnel representative, the rest of the ventilation systems are in use and functioning.

According to the assessment, the maintenance of the ventilation systems has been carried out by the on-site personnel for a maximum of twice per year in the past and once a year thereafter. No relevant documentation has been prepared on maintenance work.

The ventilation systems are generally in good condition and functioning. The systems do not need to be upgraded, and their expected lifetime under current conditions corresponds to the 50 years normally required from the date of the installation, i.e. approximately until 2050. In order to ensure the lifetime of the systems up to 2050, it is necessary to replace the fastenings of the extraction pipes on the facade with fastenings with a suitable coating and to regularly assess the condition of the equipment and the electrical system components and the pipe fastenings with perforated tape and replace them if necessary. As the fans contain moving parts, they may need to be replaced in the following period in systems S1, V1, S2, V2 and V8.

Water and sewerage system of the administrative and non-work block

In this part of the building, there is a hot and cold water supply and sanitation system for service personnel. The systems were installed when the building was renovated in 1997.

Overall, the water supply and sewerage system is in a satisfactory condition. The perspective for the systems to run until 2040 is satisfactory. Equipment in the systems, such as hot water boilers, tanks or pipe sections, may need to be replaced during this period.

1.1.6 Electrical installation inspection

Following inspection of the electrical installation has been performed:

- Electrical main circuits of the building complex;
- Energy distribution system inside the building;
- Internal wiring of the high current part;
- Weak current cabling;
- Lighting and power equipment;
- Lightning protection system;

Summary of the electrical installation inspection:

1. The internal electrical installation of the object is generally in accordance with the Electrical Safety Act in force at the time of construction (adopted on 24.01.2007, valid until 01.07.2015), and there are no significant inconsistencies with the Equipment Safety Act in force since 01.07.2015 and its implementing provisions, as well as with the requirements of the standards for the construction of electrical installations.
   The equipotential bonding system and the marking of electrical equipment need further inspection/adjustment.
2. The electrical installation has been audited by AS KH Energia-Konsult on 19.06.2020. According to the audit report No. KH-20-00081 of 19.06.2020, the electrical installation complies with the established safety requirements and can be used for its intended purpose.
3. There is some controversy about the lightning protection system:
   - According to page 2 (10) of the inspection report, the roofing is made of stone-coated bituminous roll material. *In reality, it is a metal roof;*
   - According to page 2 (10) of the inspection report, a hot-dip galvanised round steel lightning protection grid with a mesh size of 15 × 15 m has been built on the roof, there are 22 downs and a distance between downs is 15 m. *In reality, the lightning protection grid is not visible on the roof. Earthing downs are fixed to the roof*

sheeting. In accordance with EVS-EN 62305-3:2011 clause 5.2.5 tab.3, it needs to be clarified whether in this case the use of a roof covering made of sheet material with a thickness of less than 1 mm, which may get punctures in the event of lightning, is a suitable solution;

- On page 2 (10) of the inspection report, the protection class of the lightning protection system is II. - *According to EVS-EN 62305-3: 2011 clause 5.2.2 tab.2 the lightning protection grid with mesh size 15x15m belongs to class III (third), not class II Besides, the trap grid is not visible on the roof.*

1.1.7 Fire safety

The safety of the building is assessed on the basis of current fire safety requirements and the building is considered to be an industrial building. The building fails to comply with the essential fire safety requirements in the following points:

1. The wall between the boundary area structure (storage room and office block) does not provide the required fire resistance EI90 for penetrations/openings in the structure that are not insulated as fire resistant.
2. Boundary area structure (ceiling between the storage and attic) does not provide the required fire resistance EI90, in terms of penetrations/openings the insulation of which is not fire-resistant, including the absence of a fire-resistant attic hatch.
3. The ceiling between the office block and the attic does not partly provide the required fire resistance EI30, the structure is not insulated so that it is fire-resistant with respect to the absence of fire-resistant ventilation piping, etc./openings, including fire doors.
4. The wall between the boundary area structure (storage room and switchboard room) does not provide the required fire resistance EI90 for the penetrations/openings of the structure, which have no fire-resistant insulation.
5. Not all rooms of the building requiring smoke detectors are equipped with one.
6. Not all emergency exits are marked with safety signs.
7. Not all the rooms of the building requiring antipanic lighting are equipped with one.
8. The fire water tank on the property has capacity of 700 ÷ 1000 m^3 and it currently holds approximately 50 m^3 of water, which does not meet the requirements.
9. The water point is not marked according to Regulation No. 10 of the Minister of the Interior of 18 February 2021 “Requirements, conditions and procedure for the construction, testing, use, maintenance, marking and information exchange of a water point”.

1.1.8 Overhead cranes

Two cranes with crane tracks are used in the storage block of the building. Crane beams form the base of the crane track and carry both vertical and horizontal loads of the cranes. Crane beams are designed to be relatively high so that when receiving loads, there are no excessive deflections that could interfere with the operation of the crane.

The cranes used in the building have an electric drive, and are designed to be used for medium and heavy-duty work. Two bridge cranes move on the crane beams. The first of them was installed in 1964. It has a lifting height of 17.5 m and the lifting capacity of the main telpher is 50 tonnes; the lifting capacity of the auxiliary telpher is 10 tonnes.

The second crane was installed in 1976. It has a lifting height of 17.5 m and the lifting capacity of the main winch is 50 tonnes; the lifting capacity of the auxiliary winch is 10 tonnes.

However, at present, the allowed lifting capacity of the main telpher of both cranes is limited to 30 tonnes, since there were no loads heavy enough to test on when tests were carried out.

Cranes are subject to regular inspection and testing by an independent technical inspection. Also, cranes are subjected to regular maintenance, such as changing cables.

1.2 Engineering-Technical Study of the Building Materials and Structure of the Interim Storage Facility of Radioactive Waste

1.2.1 General information

The engineering-technical study of the interim storage facility was done within sub-activity 4.2.

In 1997, an interim RW storage facility was built in the MB. The interim storage is located near the northwestern end of the MB, roughly between the axes К …Ж/104 …106. Interim storage facility location inside the MB is shown in Fig. 1.2.1, marked with yellow. With 139 m^2 net surface and 250 mm thick walls, its planar dimensions are 13.2 m × 11.7 m. The total height of the structure is 11.0 m from which 1.60 m is located below the MB floor and 8.4 m above the MB floor. It is divided in half with the partition wall, both compartments with capacity to hold 360 waste containers measuring 1.2 × 1.2 × 1.2 m. The interim storage facility was planned taking into account that it is possible to accommodate all the RW from decommissioning nuclear facilities of FPNC, excluding RCs.

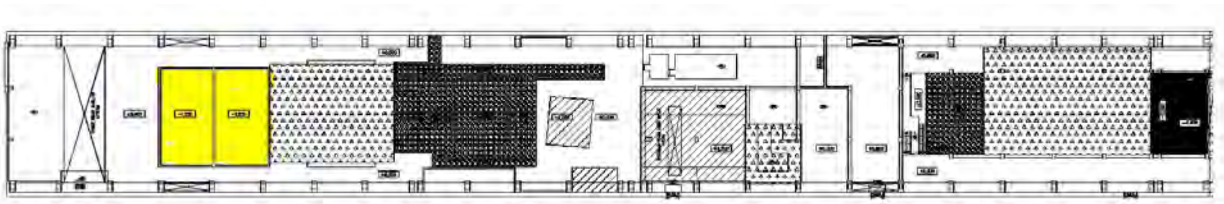


Fig. 1.2.1. Interim storage facility location inside the MB yellow marked

1.2.2 Moisture and temperature regimes

Since the interim storage facility is located in the MB, its outside moisture and temperature regimes are to a large extent the same as the MB. There are never minus temperatures and the air humidity is within a reasonable range.

1.2.3 Building structures

Structural solution, loads and materials

The interim storage facility is made of in-situ cast concrete, class B15. Its walls are 250 mm thick and 11 m high, its floor is 300 mm thick, with bottom surface 154 m^2. The walls and bottom floor is reinforced with 12 mm rebars, which are placed bidirectional, creating a mesh with 200 × 200 openings. The mesh is located in the outer and inner surface of the wall as well in the upper and lower surface of the bottom slab.

The effective loads acting on the structure are dead loads, storage loads and possible loads from the shock due to relocation of containers. The floor is designed for loads up to 320 kN/m^2 and the walls are designed to withstand the shock due to relocating of the containers.

Assessing and inspection of the structures

The structure was visible from three sides and from the top. The fourth side is cast against the sarcophagus No. 2 and was therefore not visible. Since the structure is considered a contaminated area, and the access inside the structure was not possible.

Assessing the structures was done by visual inspection, the non-destructive testing on site and destructive testing in the laboratory. The same equipment as for the audit of the MB was used for testing.

Construction-technical condition of the structures

The overall visible quality of the structures was good. The structure has been erected using modern formwork and modern technology. The carbonation depth of the concrete is approximately 1 cm, which is acceptable, as the concrete protective layer of the steel reinforcement is between 2 and 4 cm and thus the carbonisation has not reached the reinforcement. The compressive strength class of concrete determined with a hammer is between C40/50 and C50/60, which can be considered as a very good indicator. The compressive strength measured on laboratory was 45 MPa, which is also a very good, since the design value for concrete was B25. Measurement of electrical resistance showed no significant corrosion of the reinforcement, and that information is also supported by a reinforcement sample. Tensile strength of the tested reinforcement sample was also tested and the average value was 369 MPa, which is slightly lower than the design value of 390 MPa.

Damages to materials and structures

A considerable vertical crack, that is not repaired, with a depth of at least 20 cm, measured with an ultrasonic device, runs in the wall (see Fig. 1.2.2). Other cracks follow the lines of work joints.

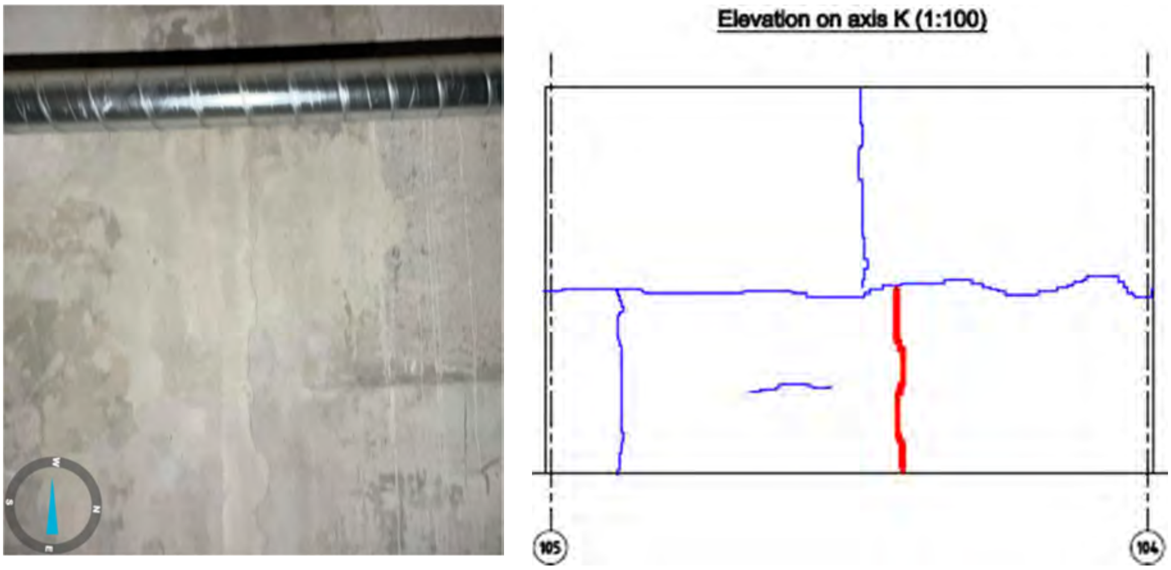


Fig. 1.2.2 Cracks on the wall of Interim storage (existing crack red marked, repaired cracks blue marked)

Heating and ventilation

Both compartments are equipped with its own dehumidifying system with an additional duct fan. DT 400 dehumidifiers are from DehuTech AB. The dehumidifying systems are in good working order.

1.3 Study of the Structure of the Reactor Sarcophagi and the Reactor Compartments

1.3.1 General information

The engineering study of the reactor sarcophagi and the RCs was done within sub-activity 4.5.

In their current form, the purpose of the sarcophagi is to separate the reactor sections from the rest of the building. It is basically a two-stage insulation, with the radioactive material inside a steel section and the section itself in a reinforced concrete sarcophagus. Locations of the sarcophagi in the MB are shown in Fig. 1.3.1.

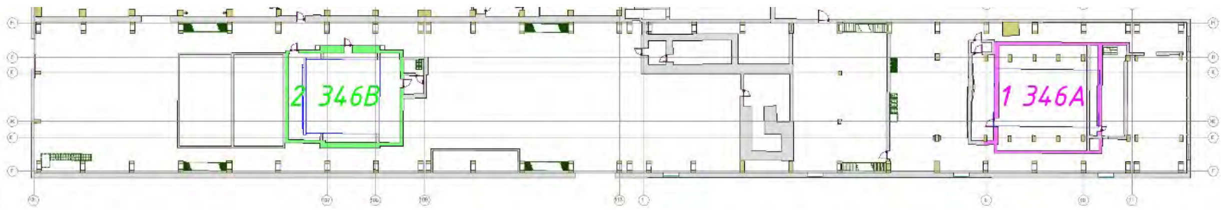


Fig. 1.3.1 Location of the sarcophagi in the building

1.3.2 Assessing of the structures

It was possible to see all the structures from the inside. As a concrete overlay had been installed above, which is unlikely to be part of the load-bearing structure, it can be said that it was not possible to view the load-bearing structure from above. Also, the north-west wall of sarcophagus No. 2 has been built together with the interim storage facility and therefore it was not possible to see it from the outside. The same equipment as for the audit of the MB was used for testing.

Visual inspection combined with non-destructive testing on site provide a fairly good assessment reliability, but the fact that not all the parts of the structure were accessible, and therefore not visible, will lower the overall reliability slightly.

1.3.3 Building structures of reactor sarcophagus No. 1 and RC No. 1

Overall structural solution and loads of reactor sarcophagus No. 1

The sarcophagus No. 1 is partly constructed of precast reinforced concrete elements and partly in-situ cast concrete. The vertical load-bearing elements are reinforced concrete poles with cantilevers. Monolithic concrete wall sections have been cast between the reinforced concrete poles. The main beams rest on the reinforced concrete poles and the supporting beams between them. It is not possible to more accurately identify whether the main and supporting beams are cast in situ or prefabricated. The ceiling of the sarcophagus is partly constructed of precast intermediate panels and largely of monolithic concrete. An additional concrete layer with a thickness of 20 cm has been poured on top of the load-bearing structure of the ceiling. This is probably intended to fill possible sealing issues in the load-bearing ceiling and to increase the concrete layer. At the moment there are only permanent loads due to dead weight, as there are no loads due to other external impacts. The first group includes, for example, the dead load of structures, permanent technological installations and the weight of surface finishing.

Sarcophagus No. 1 is located outside the CA, which means that to access it there is no need to wear equipment that limits the possible spread of contamination. The access to sarcophagus No. 1 is through doors located at both ends of the sarcophagus.

Construction-technical condition of the structures

The load-bearing structures of the sarcophagus are in good condition. The internal surface is finished with paint.

Carbonation of concrete on the internal surfaces is between 2 ÷ 13 mm and has not reached the reinforcing bar (protective layer: 28 ÷ 67 mm). The depth of carbonation of the outer surface of the ceiling of the sarcophagus was measured to be 38 mm, but as the concrete protective layer exceeds 100 mm, this is far from the dangerous limit value.

The compressive strength class of concrete measured on the internal surfaces determined by an impact hammer is C20/25 (on walls and beams), which can be considered as an average value. The strength class, measured from the top surface of the ceiling of the sarcophagus, was clearly different, being C16/20.

Measurement of the electrical resistance showed that reinforcing bars were not corroded.

Cracks can be seen in the ceiling of the sarcophagus, which are 20 cm deep when measured from above using ultrasound.

The tensile strength class of reinforcement steel determined by portable metal hardness tester is A-II (A300).

Damages to materials and structures

No major damage was visible on inspection. A small piece of concrete has been detached from one of the main beams and the reinforcing bar has been exposed. This is covered with a paint layer. On the outside of the upper part of the southeast wall, poor quality concrete work is visible. Gaps can be seen there.

1.3.4 Building structures of reactor sarcophagus No. 2 and RC No. 2

Overall structural solution and loads of reactor sarcophagus No. 2

Sarcophagus No. 2 is predominantly constructed of in-situ cast concrete, with its ceiling constructed of precast elements.

The vertical load-bearing structure is monolithic reinforced concrete walls. The monolithic walls support two combined steel beams, each 6 m from the end wall and spaced approximately 1.5 m apart. Ribbed panels and ceiling panels, partially monolithic ceiling parts also rest on the combined beams and walls. An additional concrete layer with a thickness of 20 cm has been poured on top of the load-bearing structure of the ceiling. This is probably intended to fill possible sealing issues in the load-bearing ceiling and to increase the concrete layer.

At the moment there are only permanent loads due to dead weight, as there are no loads due to other external impacts.

Construction-technical condition of the structures

The load-bearing structures of the sarcophagus are in good condition. The internal surface is finished with paint. Carbonation of concrete is between 3 ÷ 10 mm and has not reached the reinforcing bar (protective layer: 23 ÷ 88 mm).

The compressive strength class of concrete determined by the impact hammer is C30/37 from C40/50, which can be considered to be very good value. Once again, the strength class, measured from the top surface of the ceiling of the sarcophagus, was again different, being C20/25.

Measurement of the electrical resistance showed that reinforcing bars were not corroded. The tensile strength class of reinforcement steel determined by portable metal hardness tester is A-II (A300).

Damages to materials and structures

No major damage was visible on inspection. A piece has come out of one of the ribs of the ribbed panel, but the fittings are not exposed. One of the ribbed panels is likely to have been locally overloaded so that a thin layer of concrete has been broken. This has probably been repaired with a concrete overlay and no further damage was visible on the upper surface.

Problems of reactor sarcophagi and compartments construction works

The main problems are related to the upper surfaces of the ceilings of the sarcophagi. Compressive strengths there are relatively low and, in addition, cracks can be seen on the upper surface of the ceiling of sarcophagus 1 in the old part of the building. This seems to be a levelling layer with an apparent shrinkage crack, so this damage should not affect the bearing capacity. However, it is advisable to fill the cracks and monitor their possible future development.

1.3.5 RCs

Specimens from both compartments (No. 1 and No. 2) were taken and tested: Test specimen 1, RC No. 1 and Test specimen 2, RC No. 2.

Of the steels tested, test piece 1 has a higher tensile strength Rm (774 MPa vs 568 MPa) and a hardness HBW (540 vs 424) than test piece 2, but lower plasticity A (17% vs 24%). The flow diagram of the test piece 1 also shows the flow platform (although not very long). The absence of a flow platform on the test piece 2 indicates thermomechanical rolling in the production of the steel sheet, which increases the strength. The difference between the test pieces is in the chemical composition of the steel, the most important of which are test piece 1 and test piece 2, respectively C, (0.16 vs 0.09%), Ni (2.87% vs 0.50%), Cu (0.17% vs 0.46%), Si (0.36% vs 0.69%). The latter element was also used as a steel reducer and therefore both steels are Si-reduced mild steels. Based on the tensile strengths and according to СНиП II-B.3-72 the steel from RC No. 1 belongs to the class C 70/60 and the steel from RC No. 2 belongs to the class C 52/40.

1.3.6 Doors and hatches

Entering to sarcophagi is via modern steel doors that are equipped with biometric access control. Both sarcophagi have two consecutive doors. Doors are equipped with surface-mounted door closers. Sarcophagus No. 1 also has another door leading to the other end of the sarcophagus. Entering to sarcophagi is also possible through the hatches mounted on top of both sarcophagi. There is also a possibility to access the room under the compartment. Since this is elevated dose rate area, the doors must be radiation proof and made of thick metal. The modern steel doors are in good condition and working properly. Thick metal doors are difficult to open, but since their default position is closed position, then this cannot be considered as a deficiency. Still it can be pointed out that the latches of the door located in the southeastern part of sarcophagus No. 1 are cut and therefore it is difficult for the door to stay closed. Hatches are in good condition. Based on the tensile strengths and according to СНиП II-В.3-72 the steel from sarcophagus No. 1 doors belongs to the class C 52/40 and the steel from sarcophagus No. 2 door belongs to the class C 60/45.

1.3.7 Heating, ventilation, water and sewerage systems

Two dehumidifying systems and two exhaust systems are installed in both sarcophagi, one for the sarcophagus room and the other for the lower support section of the submarine compartment and the chamber it forms. The dehumidifying system operates uninterrupted, the exhaust system operates in emergency situations.

Piping is constructed of spiral-rolled galvanised sheet steel ventilation ducts. The piping in the sarcophagi is irregularly fixed with perforated steel tape. The piping in the storage room is correctly and properly fixed. The pipe fixings do not comply with the current requirements.

The systems are equipped with HEPA filters. DT 450 dehumidifiers are from DehuTech AB. The dehumidifying systems are in good working order. The exhaust systems are in working order, according to the personnel. The ventilation systems are generally in good condition and functioning. It is advisable to have better and correct support for the pipework in the sarcophagus room.

Otherwise, the systems do not need to be upgraded, and their expected lifetime under current conditions corresponds to the 50 years normally required from the date of the installation, i.e. approximately until 2055. As dehumidifiers in continuous use contain moving and electrical parts, they may need to be replaced in the following period. It is necessary to assess the technical condition of the dehumidifiers and fans on an ongoing and regular basis.

1.3.8 Electricity

Intra-sarcophagus lighting and switchgear equipment are conforming to the requirements of the standard series EVS-HD 60364 „Selection and erection of electrical equipment“. On-surface installation method has been used, cables of the strong current part are PPJ and MMJ type cables. Sockets are powered through circuit breakers located in the switchboard. Switchboard protection class IP44 is suitable for the surrounding environment in the interior of the sarcophagus.

1.4 Preparation of a 3D Model of the Reactor Compartments

To support the development of the plan for dismantling the RCs, the 3D CAD models of RCs No. 1 and No. 2 were created using the computer program SolidWorks version 2013 (https://www.edrawingsviewer.com). The source of the model is all 2D drawings of equipment and components provided by AS A.L.A.R.A. This work was done within sub-activity 4.7.

1.4.1 3D model of RC No. 1

The general view of the reactor No. 1 is presented in Fig. 1.4.1. The model contains the RC shell, internal components and poured concrete inside the compartment, compartment access staircases and platforms, concrete shielding walls, and etc. The internal components of the RC are shown in Fig. 1.4.2. In this picture some parts of the model, such as shielding walls, structural elements, bulkheads, poured concrete, and others, are hidden for a better representation of reactor main cooling circuit components. The list of internal components of the model contains:

1. Reactor,
2. Steam generators (8 units),
3. Pressurisers (6 units),
4. Main circulation pump,
5. Auxiliary circulation pump,
6. Activity filters (2 units),
7. Refrigerators (2 units),
8. Heat exchangers (2 units),
9. Circulation pumps (4 units),
10. Current converter.

Besides the mentioned components 3D model also includes some piping to represent the sequence of connection of main components, compartment support structures, bulkheads, reactor shield tanks, poured concrete over radioactive components, etc.

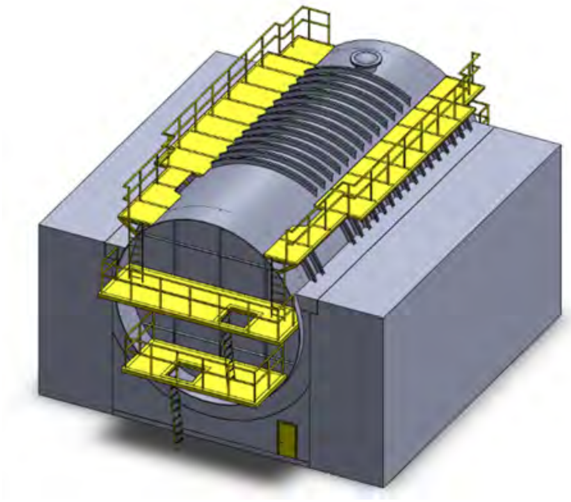


Fig. 1.4.1 General view of reactor No. 1 3D model

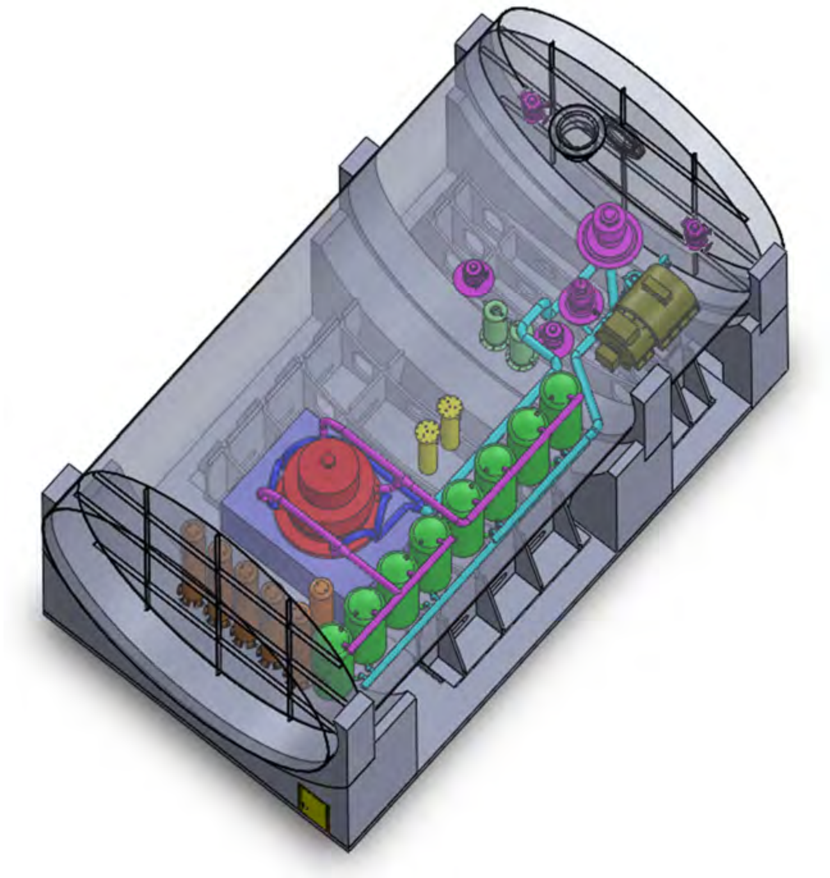


Fig. 1.4.2 View of reactor No. 1 3D model's internal components

1.4.2 3D model of RC No. 2

The general view of the reactor No. 2 is presented in Fig. 1.4.3. The model contains the RC shell, internal components and poured concrete inside the compartment, compartment access staircases and platforms, concrete shielding walls, etc. The internal components of the RC are shown in Fig. 1.4.4. In this picture some parts of the model, such as shielding walls, structural elements, bulkheads, poured concrete, biological shield, and others, are hidden for a better representation of reactor main cooling circuit components. The list of internal components of the model is as follows:

1. Reactor,
2. Steam generators – primary circuit pumps (5 units),
3. Pressurisers (3 units),
4. Primary circuit filter,
5. Filter cooler.

Besides the mentioned components 3D model also includes compartment support structures, bulkheads, reactor shield tanks, poured concrete over radioactive components, biological shielding, etc.

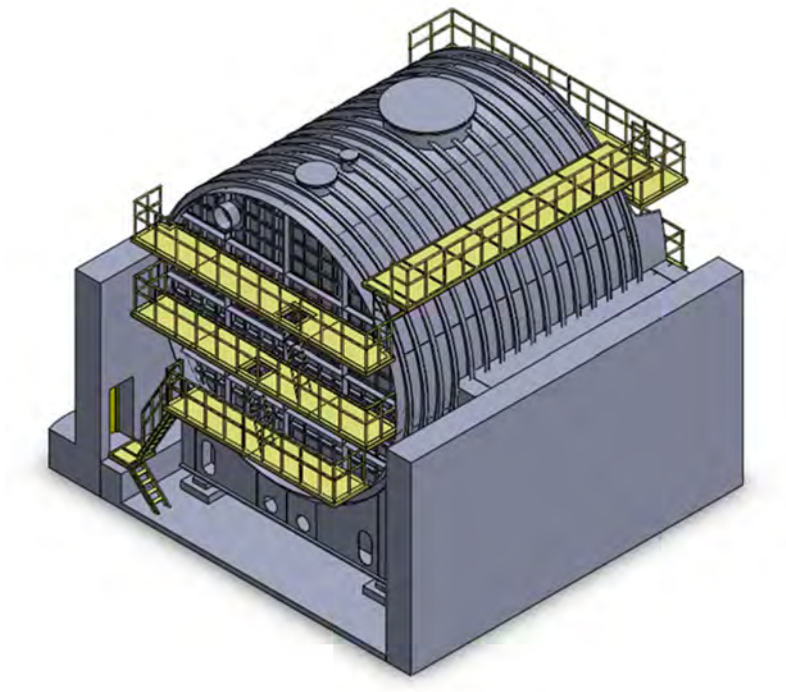


Fig. 1.4.3 General view of reactor No. 2 3D model

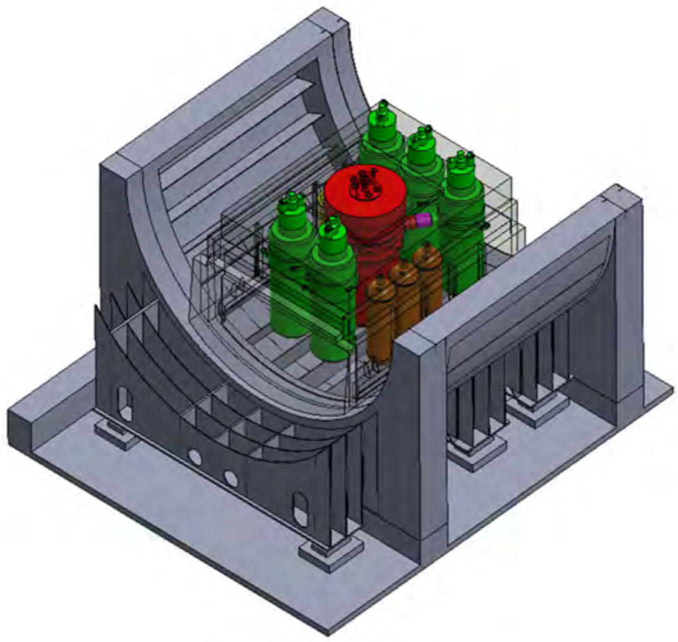


Fig. 1.4.4 View of reactor No. 2 3D model's internal components

2. RADIOLOGICAL STUDIES

2.1 Radiological Study of the Main Building of the Paldiski Site and the Interim Storage Facility

2.1.1 Implementation of the radiological survey

Radiological study of the MB and the interim storage was done within sub-activity 4.3.

Radiological study of the MB of FPNC and the interim storage facility (MB&IS) included:

- Analysis of the effect of ongoing waste management activity on the radiological situation in MB&IS,
- Analysis of available measurement results on the gamma-ray dose rate and radioactive contamination in MB&IS,
- Measurement of density and natural radionuclide inventory of the main materials used in the constructions of MB&IS,
- Preliminary categorisation of the area of MB&IS according to available data as possibly affected by contamination or not affected based on historical data and results of the analysis of the ongoing waste management activity,
- Preparation of the programme of radiological study of MB&IS.

The methodology of the radiological survey is presented in the Programme of radiological study of the MB of the Paldiski site and the interim storage facility. The radiological survey has been implemented in two campaigns. The first campaign provided preliminary data on types of radioactive contamination. The second campaign was targeted to confirmation of preliminary results and total detailed characterisation of the contaminated area.

2.1.2 Determination of contaminated areas

Survey of gamma-ray dose rates in the CA

The first campaign was implemented in two steps. As the first step total scanning by measuring the gamma-ray dose rate to obtain a dose rate higher than 0.2 µSv/h with portable hand-held instruments (see Programme of radiological study of the MB of the Paldiski site and the interim storage facility for the details about the devices and measurement method) on all easily accessible without ladder surfaces have been performed. The findings of the first survey confirmed the results of the preliminary categorisation of the area of MB&IS and were used to optimise the next surveys. The only area of an increased dose rate due to contamination during the management of spent nuclear fuel was defined. During the second step, more detailed measurements of a gamma-ray dose rate have been done. In the areas of increased gamma-ray dose rate, where measurements in the first step showed a gamma-ray dose rate lower than 0.2 µSv/h, for better determining areas and the places, mapping the area with a 2 m × 2 m grid and measuring around every point where lines of a 2 m × 2 m grid intersect.

In the third step, which was implemented during the second campaign, measurements of a gamma-ray dose rate in the areas, where a gamma-ray dose rate higher than 0.2 µSv/h was found, have been performed using collimated with lead shielding instruments Radiameter

СРП-68–01 and a gamma scanner with a $CeBr_3$ detector to reduce at least 10 times the background, which can be affected by nearby located more active radiation sources, and to determine the boundaries of the areas with higher contamination. After that during the second campaign, the gamma-ray dose rate has been measured on difficult to access highly located surfaces using a ladder, a 24-meter high mobile lift, and platforms of the cranes installed in the MB.

The scheme of the CA of the MB is shown in Fig. 2.1.1. For display convenience of the survey results, three zones (F-1, F-2, and F-3), walls (A, B, C, D) and pillars (1 ÷ 28) are marked in the scheme.

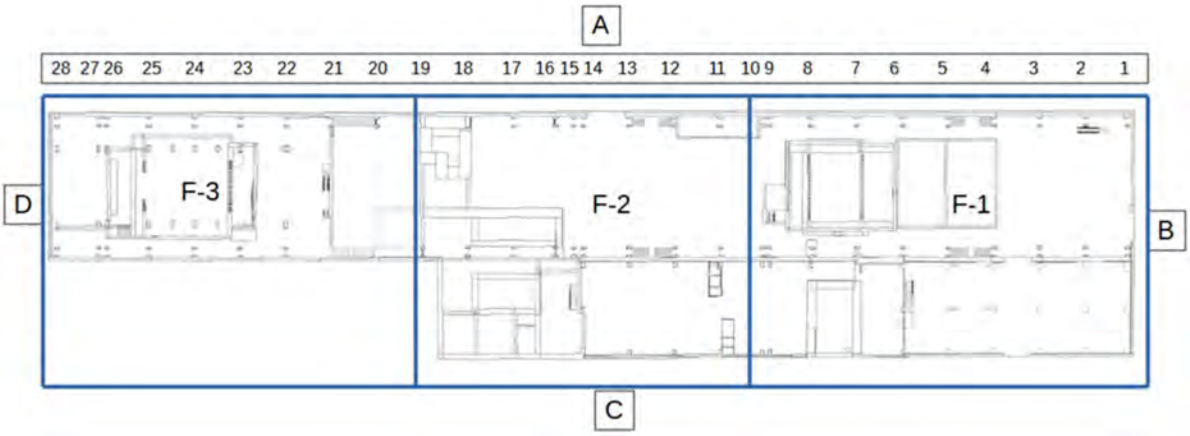


Fig. 2.1.1 Scheme of the CA

The data of averaged gamma-ray dose rates (µSv/h) around shown point results of a survey in the CA with unshielded devices was collected during the measurement campaigns. The uncertainty of measurements is lower than 20 percent.

The dose rate on the walls inside the MB was measured at various fixed heights above the floor (typically, at 1 m, 2 m and 4 m from the floor). The values varied between 0.08 µSv/h and 0.13 µSv/h on all measured surfaces of the outer walls of sarcophagus No. 1. Slightly higher dose rates were detected on the side of the sarcophagus No. 1 facing to the northern direction (i.e. facing to the location where the chimney is installed). There were no sharp increases in the dose rate detected around sarcophagus No. 1. The dose rates measured every 5 m at various heights from the floor on the walls of sarcophagus No. 1 are provided in the sub-activity 4.3 report.

The variation of the gamma-ray dose rate measured on the outer walls around sarcophagus No. 2 was more prominent as compared to the case of sarcophagus No. 1. The dose rate at the walls of sarcophagus No. 2 was at some investigated points as high as 0.22 – 0.25 µSv/h.

There is also an increased gamma-ray dose rate near the interim storage caused by the waste inside the storage. These conclusions have been confirmed by gamma-ray dose rate measurements by collimated with lead shielding measuring instruments. The gamma-ray dose rate was below 0.2 µSv/h when shielded devices were used in all places, where gamma-ray dose rates higher than 0.2 µSv/h were obtained with unshielded devices. The increase in the gamma-ray dose rate in some places is caused by containers with RW from current waste management activity. The measurements with shielded instruments confirmed the presence of the contaminated area. The contaminated area (AREA 1) is located on the wall of the NFP near RC No. 1. The location and dimensions of the contaminated area are shown in Fig. 2.1.2 and Fig. 2.1.3. The biggest value of gamma-ray dose rate in this contaminated area is equal to 0.28 µSv/h. The location and dimensions of the NFP (Nuclear Fuel Pool) are shown in Fig. 2.1.4. The gamma-ray dose rate measurements inside the NFP resulted in gamma-ray dose values up to 1.8 µSv/h near the bottom (see Figs. 2.1.5, 2.1.6).

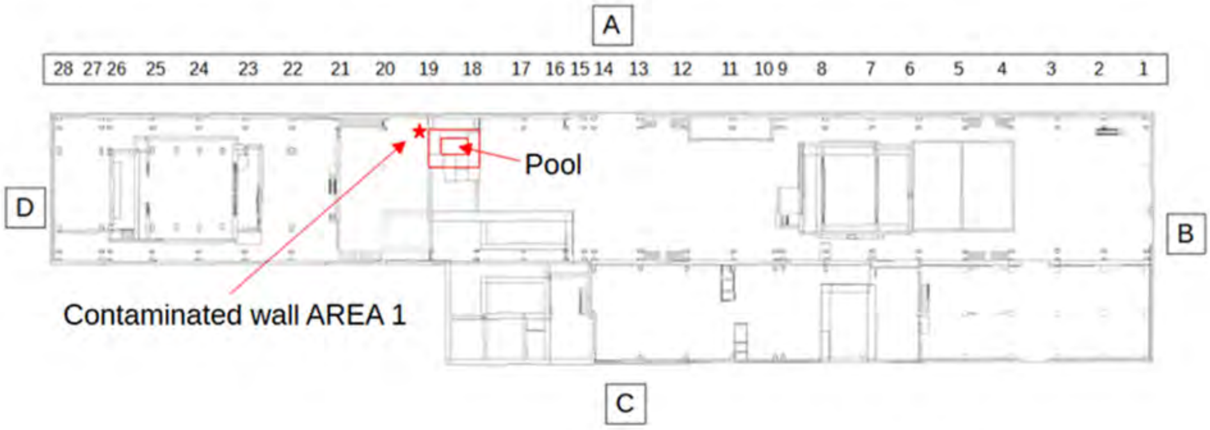


Fig. 2.1.2 Contaminated area (AREA 1) in the MB

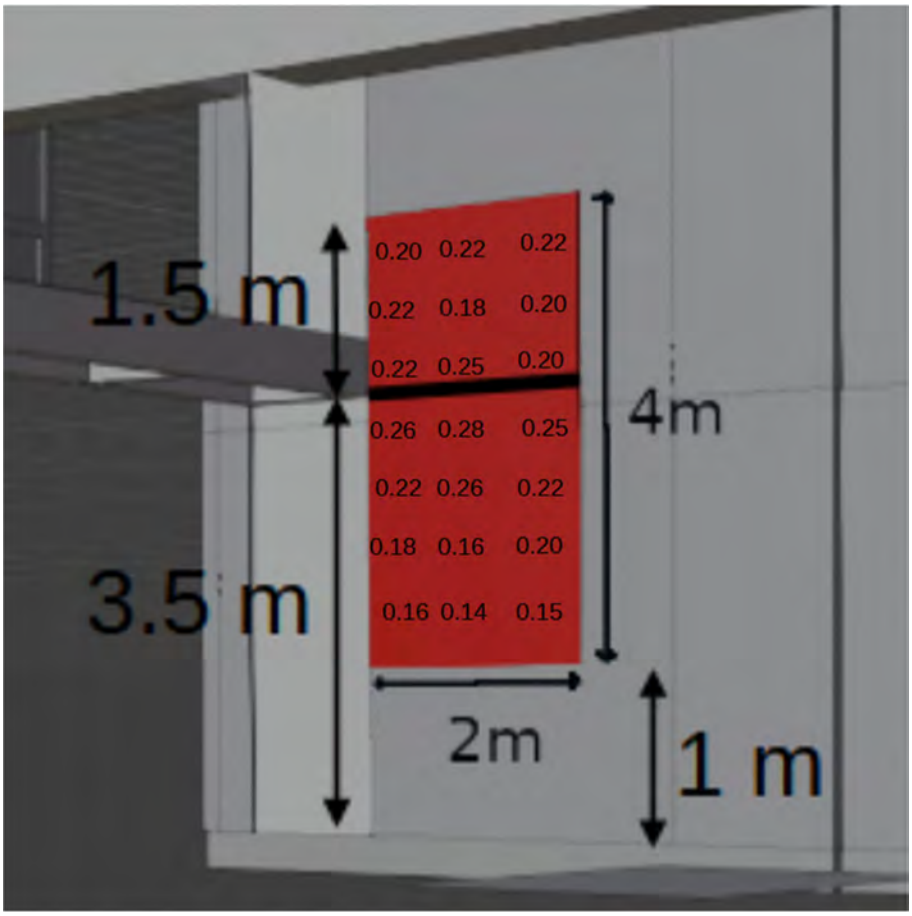
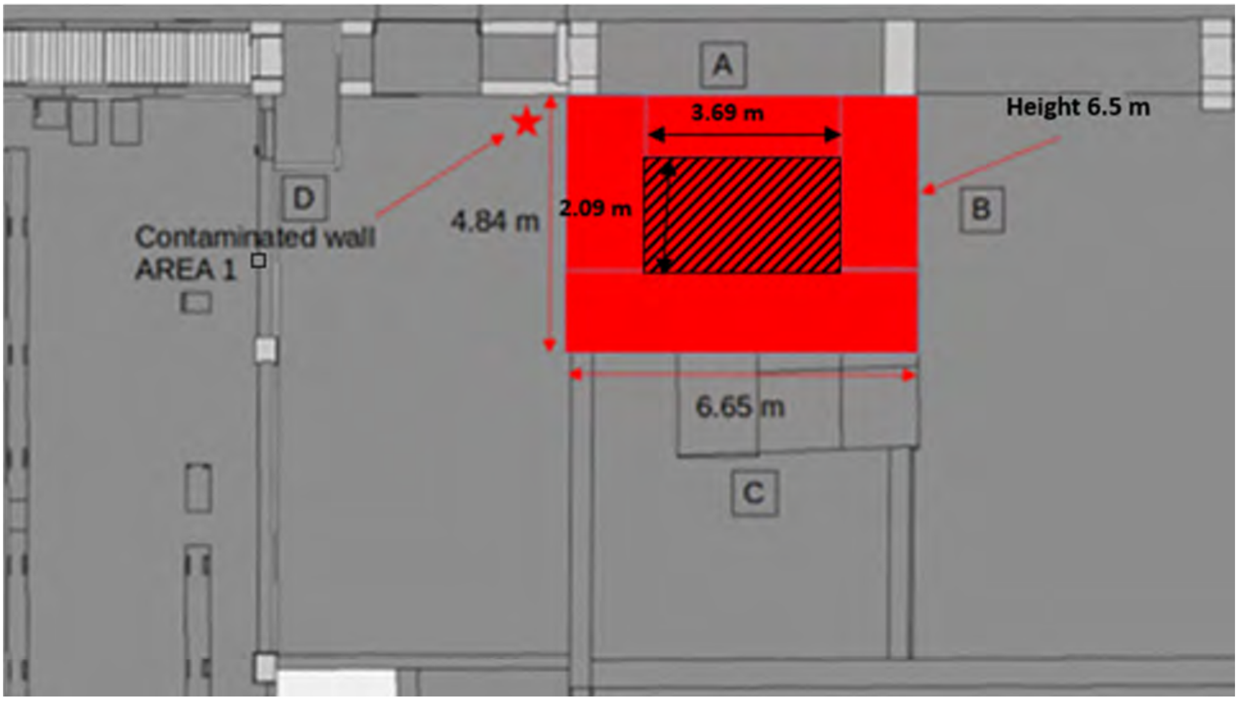


Fig.2.1.3 Gamma-ray dose rate (μSv/h) in the contaminated area (AREA 1) of the MB



- contaminated walls with surrounding - contaminated floor

Fig. 2.1.4 Location and dimensions of the contaminated NFP

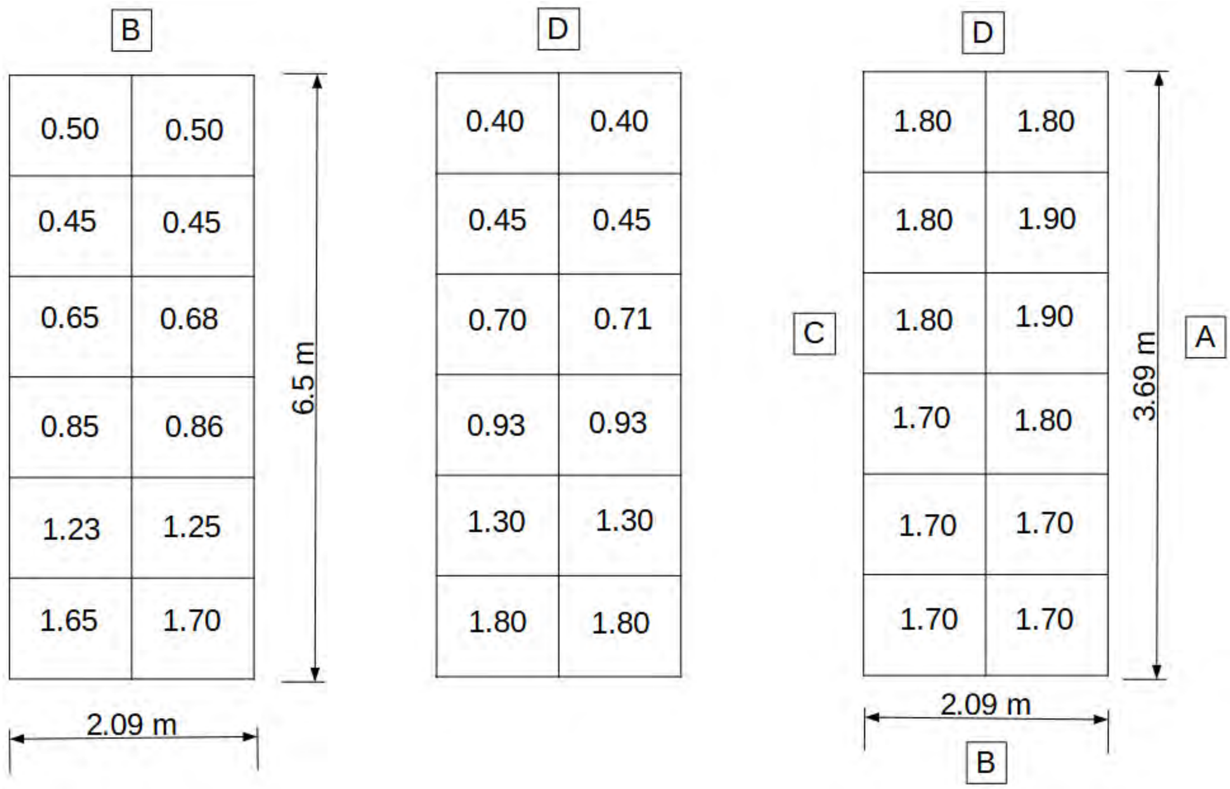


Fig. 2.1.5 Gamma-ray dose rate (µSv/h) measured on the floor and B, D walls in the NFP

A

0.45	0.45	0.50	0.50
0.45	0.45	0.48	0.47
0.70	0.68	0.68	0.66
0.92	0.91	0.90	0.91
1.29	1.28	1.25	1.25
1.70	1.70	1.70	1.70

6.5 m

3.69 m

C

0.40	0.40	0.40	0.40
0.45	0.45	0.45	0.45
0.64	0.65	0.70	0.68
0.90	0.90	0.85	0.85
1.25	1.28	1.20	1.20
1.75	1.80	1.70	1.70

Fig. 2.1.6 Gamma-ray dose rate (µSv/h) measured on C, A Wall in the NFP

Determination of gamma emitters' activity *in situ*

The gamma-ray spectrometric in situ measurements have been done to get additional information on the gamma emitting radionuclides in MB&IC. The spectrometric in situ measurements have been performed by using a gamma spectrometer with $CeBr_3$ detector. 27 spectrometric measurements have been done in the area where no contamination was found and 1 measurement in the contaminated area (AREA 1).

In-situ gamma survey of the MB&IS showed only naturally occurring radionuclides (mostly K-40) present, the measured areas could be classified as clean (Fig. 2.1.7, a graph on left, example graph for MB ceiling). The exceptional point is S-12 on the outside wall of sarcophagus No. 2. As the samples taken from this area do not show any contamination, the measured Cs-137 radiation is caused by waste containers present in the adjacent area. As a result, this area should be designated as not contaminated.

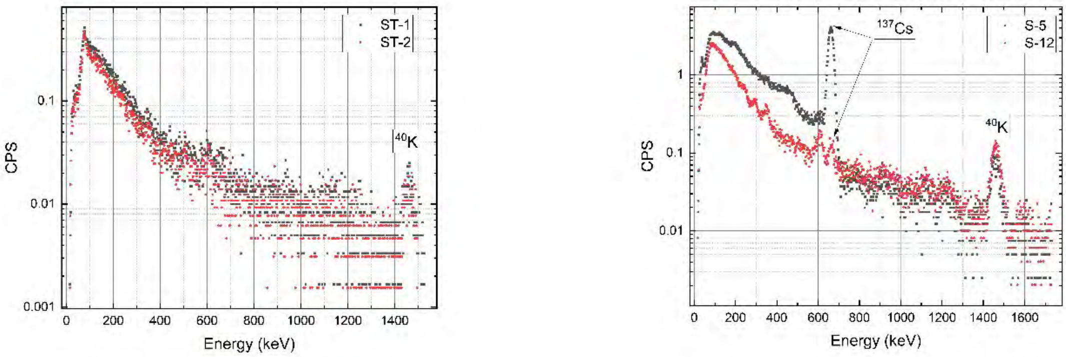


Fig. 2.1.7 Spectra of in situ measurements: clean ceiling (left) and contaminated areas of walls (right)

Two areas have been detected with increased activity of Cs-137 (Fig. 2.1.7, graph on right). The activity of the contaminated area S-5 (AREA 1) is determined by the Cs-137 gamma activity and is caused by the radioactive substance spill, also confirmed by the sample measurements in the laboratory. The area near the sarcophagus No. 2 (S-12) shows some enhanced presence of Cs-137. The samples taken from this area show some radionuclide activity; however, it is below clearance levels.

Determination of gamma, beta and alpha surface contamination

Measurement of the total α and β/γ surface contamination has been performed with the portable device Thermo Scientific™ FHT 111 CONTAMAT Contamination Monitor (details on the measurement method are provided in Programme of radiological study of the MB of the Paldiski site and the interim storage facility) on floors, walls, ceilings. Additional analysis, which included a procedure consisting of in situ measurement combined with a smear sample taking, was carried out at a few randomly selected points to evaluate an easily removable part of the contamination. The procedure included several steps: 1) in situ measurement of the surface contamination by the Contamat FHT 111M; 2) removing the dust from a surface by taking a smear from a marked spot; 3) in situ measuring the surface contamination by the Contamat FHT 111M at the same spot after taking a smear; 4) measuring a smear in the laboratory to determine the gamma-ray emitting radionuclides content and their surface activity concentration. The gamma-ray spectrometry confirmed the presence of Cs-137, Ra-226, and Am-241 in measured smears.

From smear measurement results, it has been concluded that activity levels at wall surfaces are by one-two orders lower than at other particular places (point S-12). The typical removable Cs-137 fraction is from 0.4-0.5% to 2%, but can be as high as 60% for the area with very low activity. It is important to note that all detected levels are far below exclusion levels and can be treated as not contaminated. Smear measurements inside the NFP also have not shown the presence of contamination on the surfaces.

Determination of radionuclide specific activity in samples

Sampling

For the determination of representative sampling places in the MB, the results of gamma-ray dose and spectroscopic measurements have been used. 87 smear samples and 51 volume samples from various places in the MB have been taken on every wall for determination of difficult-to-measure radionuclide concentrations in the laboratory. The exact location and description of the samples are presented in the sub-activity 4.3 report.

Specific activity of radionuclides in the samples

Specific pretreatment procedures have been applied for smear and volume samples. Solid volume samples were homogenised. Later on the samples were prepared according to dedicated procedures for alpha and beta spectrometry measurements. For the measurement of gamma-, beta- and alpha- activity, nuclear spectroscopy (gamma-ray spectrometers, liquid scintillation counter, alpha spectrometer and ICPMS mass spectrometer) methods have been used. Activity concentrations of all gamma-ray emitters in all samples were measured. Activity concentrations of relevant to long-term radiation safety difficult-to-measure nuclides were determined.

The results of destructive analysis performed in samples from the contaminated area (AREA 1) and NFP have been used for determination of a NV (Nuclide Vector), as the ratios of nuclide activities shown the reactor origin of contamination. The rest of the CA can be classified as non-contaminated as being below exclusion and release levels of activity concentrations for all taken samples.

The characterisation of contamination of the MB by radionuclides has been based on the measurement at various places of the MB as described in the sections above. The average, maximal and minimal values of specific activity of obtained nuclides have been evaluated for the smear and for the volume samples as presented in Table 2.1.1 and Table 2.1.2, respectively. All measured surface activities (Bq/m^2) and specific activities (Bq/kg) values, with the exception of the contaminated area, as can be observed in Tables 2.1.1, 2.1.2 are far below levels of release of activity concentrations of radionuclides for reuse or recycling of a building in all samples except values of a dust sample (easy removable from surface contamination). The dust sample, according to Grubs test, was excluded as an outlier as loose contamination related to current waste management activity in the MB. The mean values as well as maximal and minimal detected specific activity values have been estimated also in the contaminated area volume samples as presented in Table 2.1.3.

Table 2.1.1 Surface activities (Bq/m^2) of nuclides in smear samples from the CA

Nuclide	RRL (Reuse or Recycling Level)	Average	Max	Min
Co-60	1.0E+04	3±0.5	8±1	0.8±0.1
Cs-137	1.0E+04	62±9	580±90	0.6±0.1
Pu-238	1.0E+04	3±0.5	6±1	0.5±0.1
Pu-239/240	1.0E+03	8±1.2	16±2	2.8±0.4
Am-241	1.0E+04	22±3	450±70	0.7±0.2
Ni-63	1.0E+08	22±3	27±4	16±2
Sr-90	1.0E+06	725±109	2000±300	100±15
Pb-210	1.0E+04	61±9	660±99	5.3±0.8
Ra-226	1.0E+04	95±14	580±87	9±1

Table 2.1.2 Specific activities (Bq/kg) in volume samples from the CA

Activity	Cs-137	Am-241	Ra-226
ECL	1.0E+02	1.0E+02	1.0E+03
RRL	1.0E+03	1.0E+02	1.0E+03
average	40±6	19±3	72±11
max	280±40	40±6	340±50
min	0.3±0.1	11±2	3±0.5
Sample No. 971 (Dust) – removable contamination sample	**210±30**	**1950±300**	260±40

Table 2.1.3 Specific activities (Bq/kg) of nuclides in volume samples from contaminated areas

Nuclide	ECL	RRL	Average	Max	Min
Co-60	1.0E+02	1.0E+02	70±10	140±20	0.6±0,1
Cs-137	1.0E+02	1.0E+03	**(1.5±0.2) E+04**	**(8±1) E+04***	430±50
C-14	1.0E+03	1.0E+04	**(1.8±0.3) E+04***		
Pu-239/240	1.0E+02	1.0E+02	0.20±0.03	0.40±0.06	0.04±0.01
Pu-238	1.0E+02	1.0E+02	0.30±0.04	0.66±0.10	0.02±0.004
Am-241	1.0E+02	1.0E+02	0.50±0.07	1±0.2	0.01±0.003
Ni-63	1.0E+05	1.0E+06	7±1	15±3	2.3±0.5
Sr-90	1.0E+03	1.0E+03	33±5	87±15	5.4±1
Ra-226	1.0E+03	1.0E+03	90±10	260±30	15±2
Ka-40	1.0E+04	1.0E+04	150±25	160±30	130±20

* There is only one result of C-14 measurement in contaminated areas of MB. The other measurements show values below detection level.

2.1.3 Determination of the NV

In contaminated area, where increased dose rate values are observed, comprehensive research was performed including gamma spectrum measurement at different sides of the wall and measurement of samples in the laboratory using alpha, beta and gamma spectrometry. For detailed characterisation of radionuclide inventory, including DTM (Difficult-to-measure) nuclides, in the contaminated area a NV approach was applied. First of all, the comparison of the results of the specific activity in the samples from the contaminated area in terms of Pu isotopic ratios with the same ratios in RC zones and contaminated spot in the territory of FPNC (AREA 2) have been done. It was found that samples in the contaminated area (AREA 1) of the MB can be attributed to contamination related to the reactor origin. This conclusion enabled to develop one NV for declared nuclides of all contaminated areas including the data from the simulation of nuclide generation in FPNC reactors. Cs-137 has been determined as a key nuclide suitable for the characterisation of all contamination (contamination from the reactors and waste management) in all contaminated areas (including RCs, radioactive spots in the MB and in the territory of FPNC). Methodology of determination of the NV is presented in subsection 2.2.

2.1.4 Characterisation of MB&IS according to contamination level

The contaminated area (AREA 1) and NFP have been determined in MB&IS. The gamma-ray dose rate is equal to about 0.24 µSv/h ± 0.03 µSv/h in AREA 1 and can reach up to 1.8 µSv/h in the NFP. The values of a gamma-ray dose rate in all other areas of the CA when measurements are done with collimated shielded devices and the influence of radiation sources stored in MB&IS is excluded are below 0.2 µSv/h. Average and conservative (upper limit) activity concentrations for contaminated area (AREA 1) for year 2041 are presented in Table 2.1. 4.

Table 2.1.4 Activity concentrations (Bq/kg) for the contaminated area (AREA 1)

Nuclide	Average	Upper limit	ECL	RRL
C-14	8.4E+02	**1.4E+03**	1.0E+03	1.0E+04
Ni-59	4.6E-01	1.1E+02	1.0E+05	1.0E+06
Co-60	7.3E-02	1.0E-01	1.0E+02	1.0E+02
Ni-63	3.6E+01	8.7E+03	1.0E+05	1.0E+06
Sr-90	7.3E+01	2.1E+02	1.0E+03	1.0E+03
Nb-94	4.3E-02	1.0E+01	1.0E+02	1.0E+02
Cs-137	**9.8E+03**	**1.3E+04**	1.0E+02	1.0E+04
Eu-152	4.4E+00	**1.0E+03**	1.0E+02	1.0E+02
Eu-154	5.8E-01	**1.4E+02**	1.0E+02	1.0E+02
Pu-238	3.4E-01	5.5E-01	1.0E+02	1.0E+02
Pu-239	3.4E-01	1.2E+00	1.0E+02	1.0E+02
Pu-240	9.7E-02	3.4E-01	1.0E+02	1.0E+02

Am-241	5.0E-01	6.9E-01	1.0E+02	1.0E+02

Provided in Table 2.1.4 average activity concentration values, which are derived from the measured Cs-137 activity concentration, for all nuclides, except Cs-137 itself, are lower than the ECL. However, conservative activity concentration values (upper limit) of nuclides C-14, Eu-152, E-154 are also higher than the ECL. Upper limits of nuclides Cs-137, Eu-152 and E-154 are higher than release of activity concentrations of radionuclides for reuse or recycling of a building.

The internal surface of the NFP was decontaminated. The biggest part of the surface is clean according to smear tests. Most samples have activity concentrations lower than ECL. Activity concentrations in the other near-surface samples are not much higher than ECL. In general, near-surface samples, including samples from open cracks are not representative for the assessment of bulk contamination, which was not affected by decontamination. However, the high gamma-ray dose rates inside the NFP, point out that there is bulk contamination. One can see in Fig. 2.1.5 the gamma-ray dose variation is not big, especially if a geometric factor (there is no contamination source in the upper part) and the NFP could not be maximum filled with water, is taken into account. These facts enable to use of simulation for conservative estimation of a gamma emitter concentration. The main gamma emitter is Cs-137 as one can see from measurement results of samples from AREA 1 and AREA 2 (see subsection 2.3), which have not been decontaminated. For the simulation of the gamma-ray field inside the NFP, Microshield computer program was used. It is supposed conservatively that the Cs-137 creates the maximum measured gamma-ray dose rate. The calculated Cs-137 activity concentration together with activity concentrations of the other nuclides for the year 2041, which are calculated using the NV, is presented in Table 2.1.5.

Table 2.1.5 Activity concentrations (Bq/kg) for contamination of the structures of the NFP

Nuclide	**Average**	**Upper limit**	**ECL**	**RRL**
C-14	3.8E+02	6.6E+02	1.0E+03	1.0E+04
Ni-59	2.1E-01	5.0E+01	1.0E+05	1.0E+06
Co-60	3.3E-02	4.7E-02	1.0E+02	1.0E+02
Ni-63	1.7E+01	4.0E+03	1.0E+05	1.0E+06
Sr-90	3.4E+01	9.5E+01	1.0E+03	1.0E+03
Nb-94	2.0E-02	4.7E+00	1.0E+02	1.0E+02
Cs-137	**4.5E+03**	**6.0E+03**	1.0E+02	1.0E+04
Eu-152	2.0E+00	**4.8E+02**	1.0E+02	1.0E+02
Eu-154	2.7E-01	6.3E+01	1.0E+02	1.0E+02
Pu-238	1.5E-01	2.5E-01	1.0E+02	1.0E+02
Pu-239	1.5E-01	5.5E-01	1.0E+02	1.0E+02
Pu-240	4.4E-02	1.6E-01	1.0E+02	1.0E+02

Am-241	2.3E-01	3.1E-01	1.0E+02	1.0E+02

Provided in Table 2.1.5 average activity concentration values, which are derived from the calculated for all long-term radiation safety relevant radionuclides, except Cs-137 itself, are lower than the ECL. However, conservative activity concentration values (upper limit) of radionuclides Cs-137 and Eu-152 are higher than the ECL. The upper limit of Eu-152 is higher than the release of activity concentrations of radionuclides for reuse or recycling of a building.

All other areas in the MB after removal of radioactive sources related to current waste management activity can be classified as non-contaminated in the year 2041.

2.1.5 Waste streams from decommissioning of MB&IS

In the year 2041, there will be two waste streams from demolishing walls in the contaminated area of AREA 1 below NFP and NFP (includes the upper part above the bold line of AREA 1 in Fig. 2.1.3. It was determined that there is no easy removable contamination on the surface of the contaminated areas.

One can suppose that nuclear fuel pool has never been filled with water up to the top. Nuclear pools are usually filled with water about 0.5 m below the top. Therefore, it is decided to diminish the height of contaminated walls by 0.5 m. The gamma-ray dose rates at the top of the NFP are a few times lower than the ones at the floor (see Fig. 2.1.5). The increased measured dose rates at the top of the NFP can be caused by contaminated parts of the NFP that are below the top of the NFP. This supports the suggestion that the 0.5 m of walls at the top of the NFP is not contaminated. Calculation of volume of contaminated structures is presented in Table 2.1.6.

Table 2.1.6 Volume of contaminated areas

Height*, m	Length, m	Contamination depth, m	Volume, m^3	Structure	Location in MB (see Fig. 2.1.4)
7.50	6.69	1.50	75.27	NFP wall	A
7.50	6.69	1.50	75.27	NFP wall	C
7.50	2.09	1.50	23.52	NFP wall	B
7.50	2.09	1.50	23.52	NFP wall	D
2.09**	3.69	1.50	11.57	NFP floor	Height of 3.5 m from the floor
Total volume of NFP structures			209.15		
2.5	2.0	1.5	7.5	AREA 1 part below NFP	AREA 1 up to level 3.5 m
Total volume of NFP and AREA 1 part below NFP			**216.65**		

* Height = contaminated NFP height 6.00 + contaminated concrete below the NFP 1.50 m.

** NFP width

A conservative estimation of the volume of the affected areas supposes that all the bulk of the structure, under the contaminated surface, is contaminated as much as the surface. Therefore, the conservative approach results in about 217 m^3 volume of contaminated material. The density of the contaminated piece of the concrete wall after measurement of samples of the wall material (several samples) is estimated to be of (2400±200) kg/m^3. So, the mass of contaminated concrete structure can be estimated as about 521,000 kg. The results of *in situ* and laboratory measurements, simulations of radionuclide generation in the reactors show that contamination is of the reactor origin. Therefore, the calculated NV (see Table 2.1.4) can be used to characterise the amount of waste from demolishing AREA 1 part below the bottom of the NFP. The mass of AREA 1 part below the bottom of the NFP is 18,000 kg. The nuclide activities for the 2041 year based on average and conservative (upper limit) values of NV in the waste stream from demolishing a wall in the contaminated area (AREA 1 below the bottom of the NFP) are presented in Table 2.1.7.

Table 2.1.7 Nuclide activities (Bq) in the waste stream from demolishing a contaminated wall

Nuclide	Activity, average	Activity, upper limit
C-14	1.51E+07	2.52E+07
Ni-59	8.28E+03	1.98E+06
Co-60	1.31E+03	1.80E+03
Ni-63	6.48E+05	1.57E+08
Sr-90	1.31E+06	3.78E+06
Nb-94	7.74E+02	1.80E+05
Cs-137	1.76E+08	2.34E+08
Eu-152	7.92E+04	1.80E+07
Eu-154	1.04E+04	2.52E+06
Pu-238	6.12E+03	9.90E+03
Pu-239	6.12E+03	2.16E+04
Pu-240	1.75E+03	6.12E+03
Am-241	9.00E+03	1.24E+04

The calculated NV (see Table 2.1.5) is used to characterise the amount of the NFP waste. The nuclide activities for the 2041 year based on average and conservative (upper limit) values of NV in the waste stream from demolishing the NFP are presented in Table 2.1.8.

Table 2.1.8 Nuclide activities (Bq) in demolished structures of the NFP

Nuclide	Activity, average	Activity, upper limit
C-14	1.66E+08	2.88E+08
Ni-59	9.18E+04	2.19E+07
Co-60	1.44E+04	2.05E+04
Ni-63	7.43E+06	1.75E+09
Sr-90	1.49E+07	4.15E+07
Nb-94	8.74E+03	2.05E+06
Cs-137	1.97E+09	2.62E+09
Eu-152	8.74E+05	2.10E+08
Eu-154	1.18E+05	2.75E+07
Pu-238	6.56E+04	1.09E+05
Pu-239	6.56E+04	2.40E+05
Pu-240	1.92E+04	6.99E+04
Am-241	1.01E+05	1.35E+05

2.2 Radiological Study of the Reactor Compartments

Radiological study of the RCs was performed within sub-activity 4.4.

The methodology of the radiological survey of the RCs is presented in the Programme of Radiological Study of the Reactor Compartments. For determination of contaminated areas in sarcophagi, following methods were applied:

- Gamma-ray dose rate measurements in all the rooms in sarcophagi to determine the dose rate distribution on all the surfaces of the rooms.
- Gamma spectrometric measurements in the rooms under the reactor.
- Total gamma and beta, total alpha in situ measurements on all the surfaces of the rooms.
- Determination of specific activity of nuclides in the samples from representative places.

A radiological survey of the RCs has been implemented during 3 campaigns. The first campaign was a preliminary survey of the sarcophagi area and provided data on types of possible radioactive contamination. The campaign included scanning of the whole RCs area (accessible by measurement technique to reach without any additional installations) by examining every constructions part of both RCs. The survey of the RC No. 1 and RC No. 2, without entering into rooms under the reactor, was performed with portable hand-held dose rate measuring instruments (details about the measurement instruments and method are presented in Programme of radiological study of the RCs) to determine the areas with the gamma-ray dose rate higher than 0.20 μSv/h (possibly contaminated areas). At higher dose rate points easily removable construction material samples (for instance, 7 samples of paint) and smear samples (17 samples) from the surfaces have been taken for nuclide composition determination.

During the second campaign the detailed gamma spectrometric measurements were performed at the higher dose rate areas of RC No. 1 and RC No. 2, the additional construction material samples (25 samples) with dedicated instruments have been taken for nuclide composition determination as well as additional smear samples (20 samples).

During the third campaign additional smear samples (7 samples) have been taken from the walls inside of RC No. 1 and RC No. 2 and rooms under the reactors. Detailed gamma spectrometric measurements using collimators if needed to avoid the background from other radioactive sources have been done.

2.2.1 Survey of gamma-ray dose rates

The dose rates have been measured on walls, floor, ceiling of RC No. 1 and RC No. 2 as well as on the surface of the metal shell of the compartments. No sharp increases of a dose rate were determined on all the surfaces inside both sarcophagi (except the rooms below reactors, where higher dose rates were measured). This fact allows one to conclude that there are no radioactive contamination areas on the inner walls of sarcophagi. The maps of gamma-ray dose rates on the top of reactor No. 1 and reactor No. 2 constructions are presented in Fig.

2.2.1 and Fig. 2.2.2, respectively. Fig. 2.2.3, Fig. 2.2.4 show gamma-ray dose rates measurements in the rooms under reactor No. 1 and reactor No. 2, respectively.

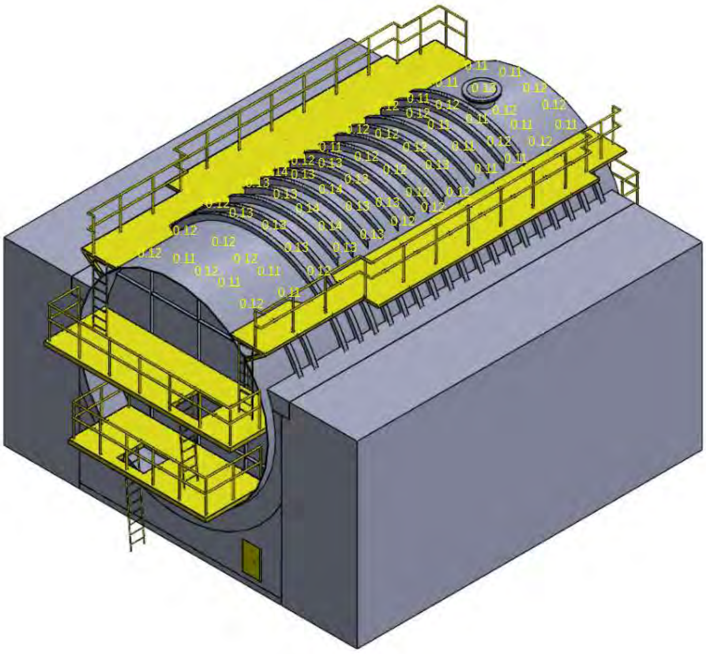


Fig 2.2.1 Gamma-ray dose rates (μSv/h) on top of reactor No. 1 structures

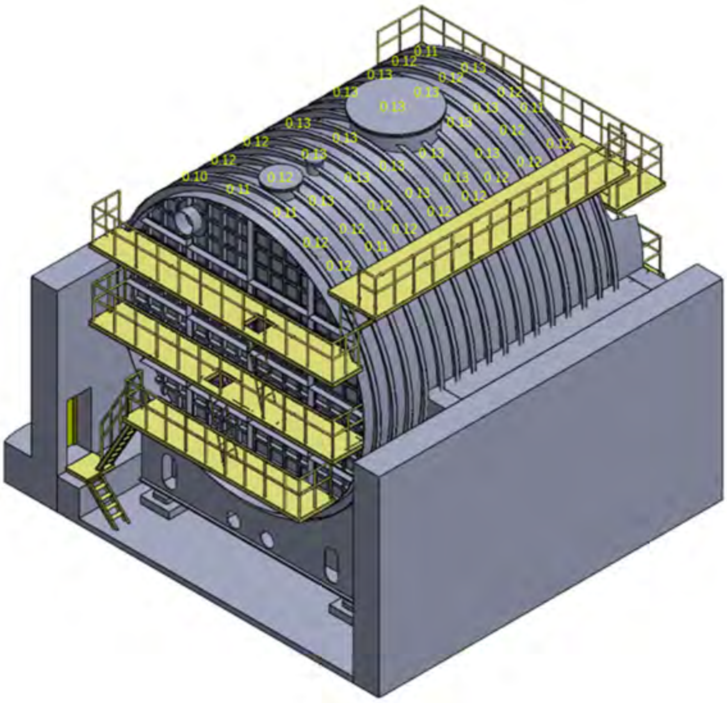


Fig 2.2.2 Gamma-ray dose rates (μSv/h) on top of reactor No. 2 structures

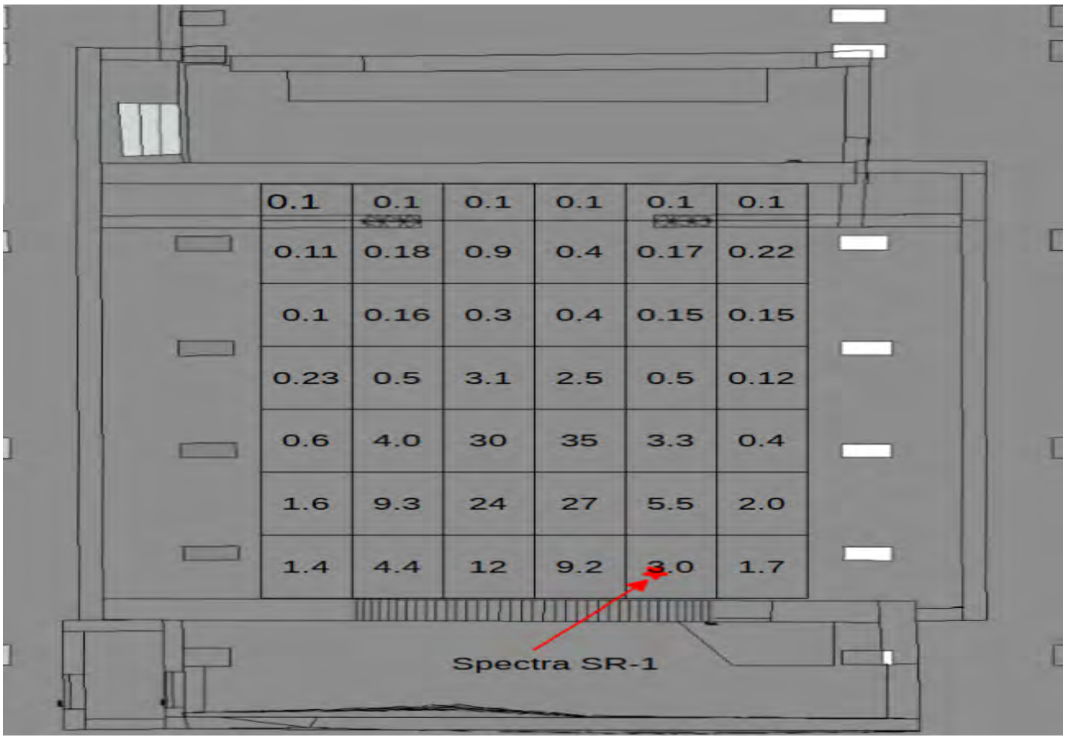


Fig. 2.2.3 Gamma-ray dose rates (μSv/h) in the room under the reactor No. 1 1 m above the floor

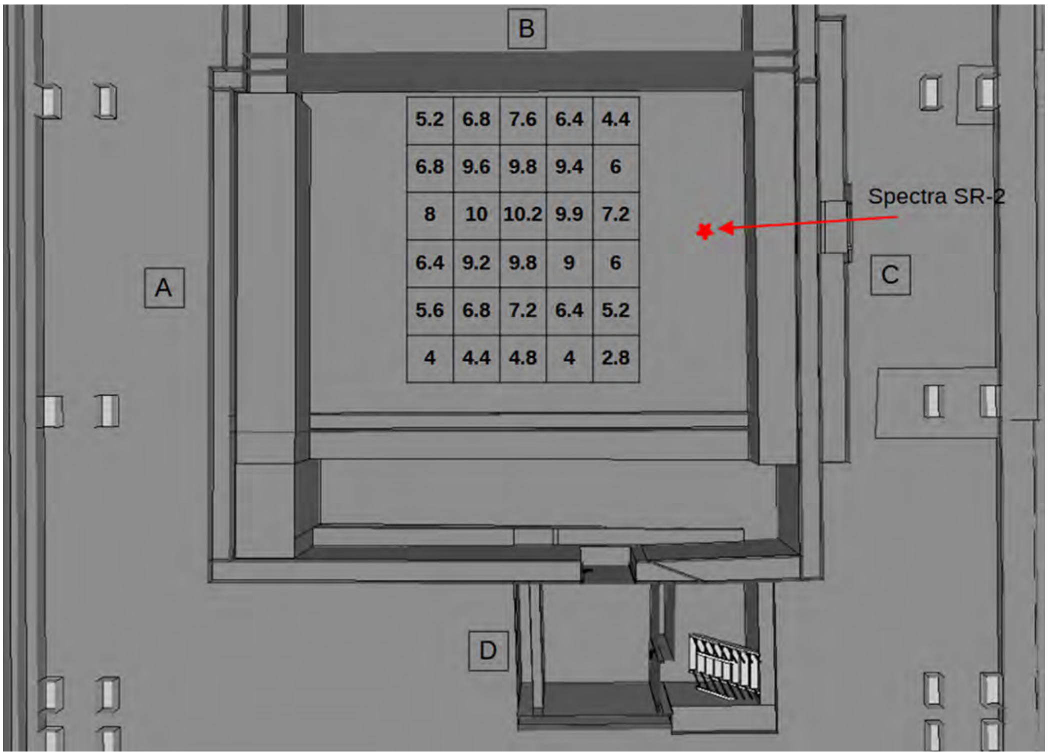
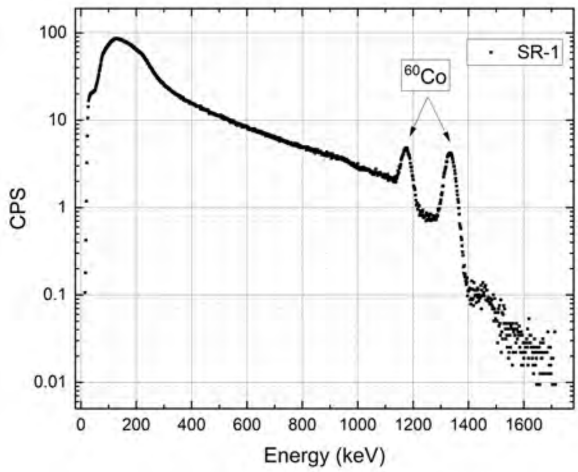


Fig. 2.2.4 Gamma-ray dose rates (μSv/h) in the room under the reactor No. 2 1 m above the floor

The average dose rate value at the ceilings of both sarcophagi was 0.11±0.02 μSv/h. Dosimetry measurements do not say anything about the pollution of the RCs (external) surfaces - the "background" average (0.13±0.02) μSv/h is typical for those rooms everywhere as well as for the metal shell of the compartments.

Higher levels of a gamma-ray dose rate were measured in the rooms below reactors. The maximum value of a gamma-ray dose rate in the room below reactor 2 was 10.5 μSv/h on 16 May 2022. The highest gamma-ray dose rate was in the room below reactor 1, it was 0.4 mSv/h on 16 May 2022. In order to assess the contamination levels in the rooms under the reactors, the gamma spectra have been measured by using the $CeBr_3$ spectrometer to determine the gamma-ray emitting radionuclides content (Fig. 2.2.5). The results show that the increase of a gamma-ray dose rate in rooms is caused by Co-60 radiation and measurable activity of radionuclide Cs-137 is not detected. The absence of Cs-137 activity which is found in the areas, where is contamination of the reactor origin, one could assume that contamination is absent. This conclusion was confirmed by measurement of smear samples in the laboratory.



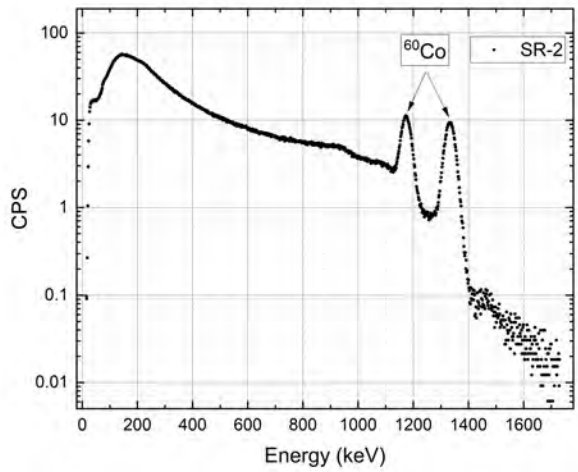


Fig. 2.2.5 Gamma spectra in rooms under the reactors: left - RC No. 1; right - RC No. 2

2.2.2 Results of measurements of alpha and beta surface contamination

In sarcophagi, surface contamination by alpha- and beta particles was assessed by direct measurement with the portable device Thermo Scientific™ FHT 111 CONTAMAT Contamination Monitor (details on the measurement method are provided in Programme of Radiological Study of the Reactor Compartments) on the walls of RCs. Measurements were performed at randomly selected points close to those where the gamma-ray dose rate was measured. An additional experiment which included a procedure consisting of a direct measurement combined with a smear sample taking was carried out at a few randomly selected points to evaluate an easily removable part of the contamination. It can be concluded that activity level of the contamination at wall surfaces varied from 0 Bq to 0.2 Bq and from 0 Bq to 1 Bq, for alpha- and beta radiation, respectively. This confirms efficient shielding of the containment by both sarcophagi in terms of preventing the spread of radioactive materials from radioactive waste management activities in the MB.

2.2.3 Determination of radionuclide specific activity in samples

For the determination of representative sampling places in the RCs, the results of the gamma-ray dose and spectroscopic measurements have been used. 42 smear samples and 32 construction material samples from various places in the RCs have been taken on every wall for determination of difficult-to-measure radionuclide concentrations in the laboratory. Methods of determination of radionuclide specific activity in samples at laboratories are described in Programme of Radiological Study of the Reactor Compartments.

The sources of long-term radiological contamination in accessible parts of sarcophagi are the same as in MB&IS. Therefore, it is anticipated the same contamination of the surfaces of RCs by C-14, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Cs-137, Eu-152, Eu-154, Ra-226, Th-232, Pu-238, Pu-239, Am-241. There are additional long-term radiological sources inside the submarine metal shield of RCs, typical representatives of natural radioactivity are Pb-210 and Ra-226 (represents the U-238 decay chain).

2.2.4 Determination of natural activity levels in the construction materials

To assess background activity levels of the construction materials by artificial radionuclides, the samples of concrete have been taken by using drilling equipment with drills of a special shape and hardness. This approach allowed taking bulk samples from concrete-based materials in the MB: walls, floor and bricks. Drills of another type have been applied to separate necessary and, at the same time, sufficient for the analysis amount of material from metal construction of the RC. It can be concluded that Co-60 can be measured with the activity concentration in metallic constructions as high as 140 Bq/kg. The metallic constructions with measurable levels of activation product Co-60 may be found in areas located in the lowest part of RC, just beneath former reactors.

2.2.5 Simulation of radionuclide composition

The scaling factor methodology is applied for evaluation of the radioactive inventory of DTM nuclides for reactor RC No. 1 and RC No. 2. The SNF modelling was used as an added information for RW samples analysis in order to confirm the nature of contamination in the samples.

Calculation of radionuclide composition in VM-A (70 MW) and VM-4 (90 MW) reactors irradiated fuel for the assessment of radioactive contamination of RCs

The purpose of the calculation was to obtain the radionuclide inventory of irradiated nuclear fuel of 20% initial enrichment U-235 of two reactors VM-A and VM-4 by using SCALE6.2.3 code. Calculated radionuclide inventory was used to assess the nuclide ratios in fuel and neutron activated metallic constructions of the dedicated reactors, which were operated in FPNC since 1968 until 1989. The procedure of calculation was comprehensively explained in Annex A of Programme of Radiological Study of the Reactor Compartments. As the actual reactor power history of FPNC reactors was not available, the sensitivity analysis due to different operating regimes was performed in different cases (VM-A and VM-4) to obtain possible values of isotopic ratios in the measured samples taken from FPNC.

Sensitivity analysis due to different power regimes of operation was performed in VM-A (70 MW) and VM-4 (90 MW) cases:

- Averaged reactor power for operation period (I case: 9.16 MWth – 12 years, II case: 6.2 MWth – 9 years and III case 9 MWth – 6 years);
- Nominal reactor power short period and decay (Ia case: 70 MWth – 1 month every year for 12 years, IIa case: 70 MWth – 1 month every year for 9 years, and IIIa case 90 MWth – 1 month every year for 6 years also additionally 2D calculation IIIa case);
- Nominal reactor power (Ib case: 70 MWth – 574.21 d, IIb case: 70 MWth – 293.33 d, and IIIb case 90MWth –222.22 d).

Averaged reactor power case calculation results

Depleted fuel radionuclide composition in mass (m, g) and activity (A, Bq) for VM-A and VM-4 operating at averaged reactor power for operation period calculated at year 2022 for sensitivity analysis of trace nuclide isotopic ratios evaluation is presented for following cases:

- I case: 9.16 MWth – 12 years,
- II case: 6.2 MWth – 9 years,
- III case 9 MWth – 6 years.

Nominal reactor power for short operation period every year and decay calculation results

Depleted fuel radionuclide composition in mass (m, g) and activity (A, Bq) for VM-A and VM-4 operating at nominal reactor power for a short operation period every year and decay reactor power for operation period calculated at year 2022 for sensitivity analysis of trace nuclide isotopic ratios evaluation is presented for following cases:

- Ia case: 9.16 MWth – power 48 d and 317 d of decay every year for 12 years, decay 15647 d after,
- IIa case: 6.2 MWth – power 33 d and 332 d of decay every year for 9 years, decay 12377 d after,
- IIIa case: 9 MWth – power 37 d and 328 d of decay every year for 6 years, decay 12338 d after and additionally the same for IIIa 2D case.

Nominal reactor power case calculation

Depleted fuel radionuclide composition in mass (m, g) and activity (A, Bq) for VM-A and VM-4 working at nominal reactor power calculated at year 2022 for sensitivity analysis of trace nuclide isotopic ratios evaluation is presented for following cases:

- Ib case: VM-A 70 MWth – 574.21 d,
- IIb case: VM-A 70 MWth – 293.33 d,
- IIIb case: VM-4 90 MWth – 222.22 d.

Sensitivity analysis of trace nuclide isotopic ratios correspondingly for depleted fuel radionuclide composition in mass (m, g) and activity (A, Bq) for VM-A and VM-4 reactors different operation regimes is presented for following cases:

- I average case for VM-A 70 MWth (574.21 d of operation in different nominal power, average power and nominal reactor power for a short operation period every year and decay regimes, respectively),
- II average case for VM-A 70 MWth (293.33 d of operation in different nominal power, average power and nominal reactor power for a short operation period every year and decay regimes, respectively),
- III average case: VM-4 90 MWth (222.22 d of operation in different nominal power, average power and nominal reactor power for a short operation period every year and decay regimes, respectively).

As can be observed from measurement results in Table 2.2.1 of mass and activity ratios for different samples taken at FPNC in 2022, the almost all Pu-240/Pu-239 mass ratios (with the exception of sample 104) and Pu-238/Pu-239+240 activity ratios perfectly fit in the calculated mass and activity ratios ranges if different reactor/operation regimes are taken into account. Contrary, there is no agreement for Am-241/Pu-239+240 activity ratios – which most probably is influenced by Am-241 additional source, the same tendency is also observed for Am-241/Cs-137, where 3-17 times higher values are obtained comparing with maximal modelling value (except for sample 1039, it seems that here reactor contamination could be identified according to all measured ratios), Sr-90/Cs-137 ratio is also too large compared with modelled ratio, most probably contamination accrued from additional Sr-90 source.

*Table 2.2.1. Experimentally obtained mass and activity ratios for different samples FPNC**

Sample label	965	1077	1093	1096	109	104	average
Mass ratio							
Pu-240/Pu-239	0.14	0.13	0.09	0.10	0.14	0.57	0.19
Activity ratio							
Pu-238/Pu-239+240	0.6	0.5	0.4	0.4	0.2	1.4	0.57
Am-241/Pu-239+240	100	173	72	195	119	164	137.28
Am-241/Cs-137	0.16	0.65		0.80	0.31	0.14	0.35
Sr-90/Cs-137	1.85	50	1.4	3.1	<0.42	<0.4	9.53

*Data from the MB (AREA 1) and contaminated spot in the territory of FPNC (AREA 2) are of the same reactor origin.

2.2.6 The C-14 estimation in metallic construction

There is insufficient data for the full reactor vessel and other metallic hardware activation simulations to achieve accuracy better than presented in previous reports. In order to estimate the amount of C-14 the ratio of formation C-14 compared with formation of Ni-63 was calculated. In the simplified approach of metallic construction activation of VM-A and VM-4 the SCALE 6.2.3 simulation of 2D configuration with non-standard geometry of the fuel cells geometry with SS304 cladding material activation was used to estimate the relative C-14/Ni-63 activation. The impurity of material was taken from considering the same impurities as for reactor vessel material (Steel 1.6310), the impurity N makes 0.013 (wt%). According to typical steel compositions for FPNC reactors, the C-14/Ni-63 activity ratio is of (0.5÷5) ´10^{-4}. According to available in literature data, the activity of Ni-63 in the reactor vessel is on the order of 10^{13} Bq. Then the activity of C-14 is about 10^{9} Bq, and the specific activity is about 3 ´10^{4} Bq/kg, this is less than the C-14 exemption level of 10^{5} Bq/kg.

Also according to the assumptions made in the IAEA-TECDOC-938, in the overall balance of the induced activity of structural materials, radiocarbon makes up no more than 0.01% ÷ 0.001% of the total induced activity. If we convert these data into average specific activity, we get the amount of C-14 in the metal of the reactor vessel: $3.7 \times 10^{4} \div 9.3 \times 10^{5}$ Bq/kg (averaged data of 10 ship reactors). In FPNC reactors case, the power generation was relatively low, so the accumulation of C-14 is lower and estimation of specific activity of 3 ´10^{4} Bq/kg is quite conservative.

2.2.7 Simulation of VM-A and VM-4 metallic construction activation

In order to assess measurements of traces of neutron activation materials measured in the samples of FPNC, additionally the MCNP6 geometry of VM-4 (taking into account OK-150 reactor's geometry) was created for reaction rates calculation of the main activation products (mainly Co-60, Ni-63, Fe-55). The geometry of a small fragment of active core described by the MCNP6 model is presented in Fig. 2.2.6. It was used to obtain the fluxes and the reaction rates in fuel, Eu_2O_3 control rod (absorber) and cladding (SS304) materials.

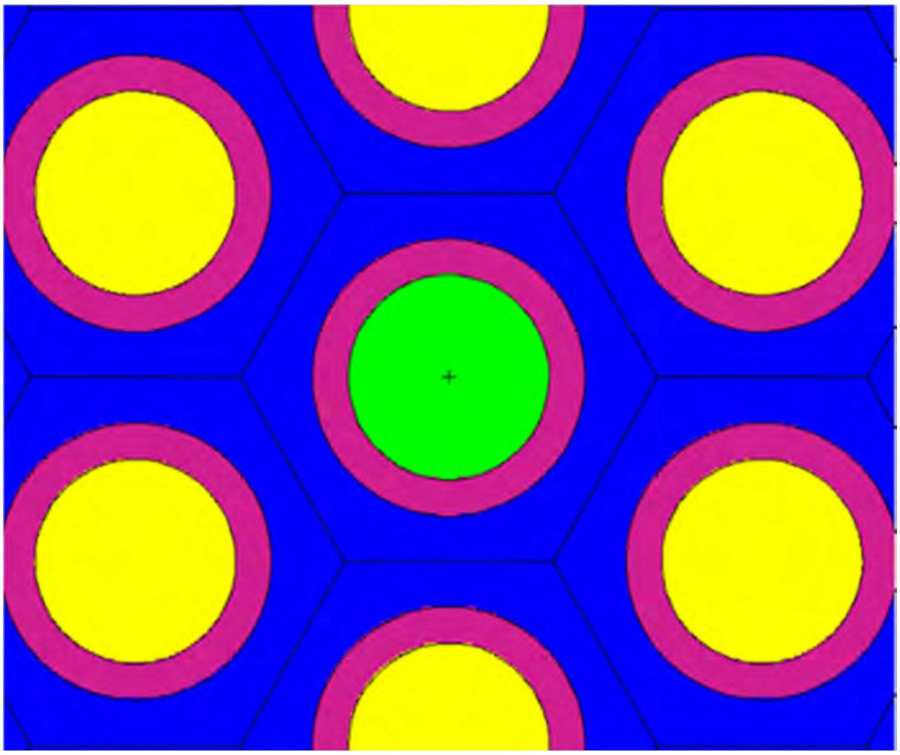


Fig. 2.2.6 Fragment of OK-150 based VM-4 geometry for main activation products reaction rates calculations in fuel (yellow), Eu_2O_3 absorber (green) and cladding materials (SS304 red)

2.2.8 Simulation by SCALE6.2.3 code package of ratios of transuranic isotopes

Taking into account the calculation results of nuclear fuel depletion in the reactors the modelling and neutron activation in the construction materials calculations the scaling factors were obtained as presented in Table 2.2.2, the ratios between experimentally detectable nuclides and Pu-239+240 are presented in Table 2.2.3.

Table 2.2.2 Scaling factors determined by simulation of nuclear fuel depletion in the reactors

Nuclide ratio	Modelling ratio	Calculation code
Co-60/Cs-137	1.86E-04	MCNP
C-14/ Cs-137	0.03	MCNP
Pu-239+240 /Cs-137	0.014	SCALE6.2.3
Pu-238/Cs-137	0.011	SCALE6.2.3
Am-241/Cs-137	0.02	SCALE6.2.3
Sr-90/7Cs-137	0.89	SCALE6.2.3
Ni-63/Cs-137	0.53	SCALE6.2.3
Eu-154/Eu-152	0.23	MCNP
Ni-63/Co-137	6.21	According to IAEA-TECDOC-938

Ni-59/Ni-63	0.01	MCNP
Eu-152/Ni-63	0.27	According to IAEA-TECDOC-938

Table 2.2.3 Ratios between experimentally detectable nuclides and Pu-239+240

Nuclide	Modelling ratio with Pu-239+240
Ni-63	39
Sr-90	67
Cs-137	75
Eu-152	0.01
Eu-154	0.15
Pu-238	0.76
Pu-239	0.78
Pu-240	0.22
Pu-241	7.0
Am-241	1.5

2.2.9 Determination of NV

The methodology of determination of a NV is provided in IAEA Nuclear Energy Series No. NW-T-1.18. It was found that Cs-137 is an appropriate key nuclide suitable for the characterisation of the total contamination (contamination from the reactors and waste management) in the territory of FPNC in all contaminated areas. For all declared radionuclides, the activities of which can be measured, a correlation analysis was performed using the values of the concentrations determined during the measurements. If a sufficient correlation ($R > 0.5$) was found between DTM and ETM nuclide-specific activities, a linear regression of logarithms was performed and a scaling factor of the nuclide concentrations was determined. In the case when an insufficiently strong correlation ($R < 0.5$) was found between the concentrations of DTM and ETM nuclide-specific activities, the weighted arithmetic average of the ratios of the activities of DTM and ETM nuclide-specific activities was determined. Activity values and scaling factors for other radionuclides whose activities could not be measured were determined by theoretically estimated ratios with intermediate key nuclides. In the Table 2.2.4, scaling factors between nuclides and Cs-137 specific activities derived from results of specific activity measurements in contaminated samples are presented for the year 2041.

Table 2.2.4 Scaling factor (K) based on Cs-137 specific activity calculated for the year 2041

Nuclide	*K*	*K*, upper limit
C-14	8.5E-02	1.5E-01

Ni-59	4.7E-05	1.1E-02
Co-60	7.4E-06	1.1E-05
Ni-63	3.7E-03	8.8E-01
Sr-90	7.4E-03	2.1E-02
Nb-94	4.4E-06	1.0E-03
Cs-137	1.0E+00	-
Eu-152	4.4E-04	1.1E-01
Eu-154	5.9E-05	1.4E-02
Pu-238	3.4E-05	5.6E-05
Pu-239	3.4E-05	1.2E-04
Pu-240	9.8E-06	3.5E-05
Am-241	5.1E-05	7.0E-05

These scaling factors are valid for characterisation of RCs and other “reactor origin” waste including increased dose rates/activity spots in the MB and environment due to leakage of radionuclides during the waste treatment procedures if the measured radionuclide ratio proves the reactor origin of the spots.

2.2.10 Characterisation of RCs according to contamination level

Although the results of the measurement of gamma-ray dose rate show a substantial increase in the room under the reactor in RC No. 1 and RC No. 2, in situ total gamma beta and alpha activity measurements on the surfaces of walls and floor of the room, measurement of specific activity in smear samples from the surfaces of the room did not show the presence of any activity higher than exclusion and release levels. The *in-situ* gamma spectrometry measurements provide a reasonable explanation that the increased gamma-ray dose rate is caused by radiation of radionuclide Co-60, whose presence is anticipated in activated reactor constructions. As a result, all areas inside sarcophagi accessible without the destruction of a submarine metal shell can be treated as non-contaminated.

2.2.11 Waste streams from decommissioning RCs

The results of the radiological study of RCs showed that no surface contamination is present in all area inside sarcophagi accessible without the destruction of a submarine metal shell. As a result, it is not anticipated any waste streams from decommissioning of structures of RCs, which are outside the submarine shell. All waste streams from decommissioning of RCs will be produced from dismantling the submarines.

There are 4 types of waste inside the submarine shell: separate waste pieces and bags embedded in concrete, boxes with spent sealed sources embedded in concrete, activated reactor structure materials, and materials contaminated by the coolant.

There is no data on the amount of radionuclides and the radionuclide content in individual waste and bags embedded in concrete. Analysis of available information on the history of waste contamination and gamma-ray dose rate on the surface shows that the waste embedded in concrete in both stands, except spent sealed sources belong to the category of short-lived very low-level waste now and will have specific activity below exemption and clearance level in 2039.

Vessels of the reactors will be disposed of entire, not fragmented. All other dismantled equipment will be fragmented into small pieces. There is a lack of radionuclide composition data for some equipment, total mass activity is only provided. Therefore, the conservative estimates were done.

The waste amount from dismantling of the primary circuit pipelines in RC No. 2 was set as big as for the primary circuit pipelines in RC No. 1. The specific activity of main corrosion products of the primary circuit filter was also assigned to primary circuit pipelines in RC No. 2. The ratio between surface and induced activity in reactor No. 1 was applied for reactor No. 2. The ratio between surface and induced activity in steam generators in RC No. 1 was applied to the equipment in RC No. 2. For composition of main corrosion nuclides on the surface of the equipment in RC No. 2, their composition in RC No. 1 was used.

Table 2.2.5 and Table 2.2.6 provide scaled characteristics of the metal equipment embedded in concrete in RC No. 1 and RC No. 2 for the year 2039, respectively.

Table 2.2.5 Characteristics of the equipment embedded in concrete in RC No. 1 for the year 2039

Equipment	Mass, kg	Total surface activity, Bq	Total induced activity, Bq	Total specific activity, Bq/kg				
				Co-60	Ni-59	Ni-63	Eu-152	Eu-154
<u>Steam generator</u>	21600	5.98E+09	7.70E+10	**1.02E+04**	6.55E+04	**3.76E+06**		
<u>Pressuriser</u>	7200	3.09E+08	1.00E+06*	**1.09E+02**	6.72E+02	4.23E+04		
Reactor coolant pump GCEN-146	4600	9.58E+07	1.00E+06*	5.31E+01	3.29E+02	2.07E+04		
<u>Auxiliary reactor coolant pump VCEN-147</u>	1800	7.66E+07	1.00E+06*	**1.09E+02**	6.73E+02	4.23E+04		
<u>Refrigerator HGCEN-601</u>	301	1.77E+08	1.00E+06*	**1.49E+03**	9.23E+03	**5.81E+05**		
<u>Refrigerator HGCEN-146M</u>	115	1.02E+08	1.00E+06*	**2.26E+03**	1.40E+04	**8.79E+05**		
<u>Refrigerator XVCEN-147M</u>	52	3.83E+07	1.00E+06*	**1.91E+03**	1.18E+04	**7.42E+05**		
<u>Iron-water shielding tank</u>	52000	-	7.99E+11	**7.02E+02**	4.35E+03	**2.74E+05**		
<u>Activity filter</u>	1130	1.56E+08	1.00E+06*	**1.14E+02**	7.08E+02	4.43E+04		
<u>Heat exchanger VP2-1-0</u>	450	1.93E+07	1.00E+06*	**3.56E+02**	2.21E+03	**1.39E+05**		
<u>Primary circuit pipelines</u>**	3000	4.23E+08	1.00E+06*	**1.09E+02**	6.72E+02	4.23E+04		
Hull beneath the reactor***	2700	-	2.84E+07	2.79E+01	1.80E+02	1.03E+04		
<u>Concrete blocks</u>***	66750	-	2.32E+08	3.40E+00			**3.43E+03**	4.59E+01

* - The maximal induced specific activity of the other equipment.

** - The total surface activity calculated from total specific activity.

*** - The total induced activity calculated from total specific activity.

<u>Bold and underlined</u> are considered as RW.

Table 2.2.6 Characteristics of the equipment embedded in concrete in RC No. 2 for the year 2039

Equipment	Mass, kg	Total induced activity, Bq	Total surface activity, Bq	Total specific activity, Bq/kg			
				Co-60	Ni-59	Ni-63	Nb-94
Steam generator block – primary circuit pump	71000	1.20E+09	9.32E+07	**2.00E+02**	2.32E+01	1.68E+04	
Primary circuit filter refrigerator	2780	7.97E+08	6.18E+07	**2.97E+03**	3.40E+03	**3.02E+05**	
Pressuriser	6000	1.94E+07	1.50E+06	2.56E+01	4.22E+01	3.41E+03	
Primary circuit filter	1980	7.96E+07	6.18E+06	**4.02E+02**	4.58E+02	4.25E+04	
Electric cool-down pump	750	8.27E+06	6.42E+05	**1.01E+02**	1.37+02	1.16E+04	
Primary circuit pipelines	3000	1.21E+08	4.02E+04	**3.94E+02**	4.09E+02	3.94E+04	
Shield tank	66180	3.10E+11	-	**9.00E+03**	6.20E+04	**4.50E+06**	4.99E+03
Concrete shield blocks (closest to reactor)	38100	1.20E+06	-	5.51E-01	3.94E-01	3.15E+01	

Bold and underlined are considered as RW.

As one can see the results provided in Tables 2.2.5, 2.2.6 show that the LLW-SL will be produced due to fragmentation of pressurisers, auxiliary reactor coolant pump VCEN 147, heat exchanger VP2-1-0, and concrete blocks in RC No. 1, steam generator block – primary circuit pump, primary circuit filter, electric cool-down pump, primary circuit pipelines in RC No. 2. The LILW-LL will be produced due to fragmentation of steam generators and connected pipelines, refrigerator HGCEN 601, refrigerator HGCEN-146M, refrigerator XVCEN 147M, activity filters, primary circuit pipelines, iron-water shielding tank in RC No. 1 and primary circuit filter refrigerator, shield tank in RC No. 2.

In the previous projects contamination of the equipment of reactors only by few corrosion radionuclides was estimated. For the assessment of the lacking characteristics of waste, a NV was used. Taking into account that the nickel amount in the reactor vessel material could be estimated within an order of magnitude, Ni-63 radionuclide was used as a key nuclide for the application of the recalculated NV. Table 2.2.7 summarises characteristics of waste streams produced after dismantling of all equipment inside submarine shells.

Table 2.2.7 Waste streams for the year 2039

Waste type	Volume, m^3	Mass, kg	Nuclide	Activity, Bq
LLW-SL	89.36	1.53E+05	Co-60	1.89E+07
			Ni-59	2.51E+07
			Ni-63	1.80E+09
			C-14	3.07E+08
			Sr-90	1.18E+07
			Nb-94	5.63E+05
			Cs-137	5.63E+08
			Eu-152	2.88E+08
			Eu-154	1.35E+07
			Pu-238	3.16E+04
			Pu-239	6.76E+04
			Pu-240	1.97E+04
			Am-241	3.94E+04
LILW-LL	147.13*	2.28E+05*	Co-60	4.26E+11
			Ni-59	1.34E+12
			Ni-63	7.98E+13
			C-14	1.16E+13
			Sr-90	1.76E+09
			Nb-94	4.14E+08
			Cs-137	8.38E+10
			Eu-152	1.51E+12
			Eu-154	4.81E+11
			Pu-238	4.69E+06
			Pu-239	1.01E+07
			Pu-240	2.93E+06
			Am-241	5.87E+06

* - Volume of fragmented equipment is equal to 108.25 m^3 (without reactors) and weight is equal to 1.47E+05 kg.

It is supposed at least 0.58 m^3 of waste can be filled in 1 m^3 concrete container with an outer volume of 1.728 m^3. For about 226 m^3 {28.0 (spent sealed sources in concrete + very low-level waste embedded in concrete) + 89.36 (LLW-SL) + 108.25 (LILW-LL)} of raw waste is needed 390 containers with an outer volume of about 673 m^3. For two reactors the volume of containers is 70 m^3 (2 × 35 m^3). The estimated disposal volume from decommissioning of RCs in previous studies (2014-2015) with reactors (2 × 35 m^3) was 987 m^3. Disposal volume estimation from current studies including reactors is 743 m^3 (673+70), i.e. 24.7 % less than from studies 2014-2015.

2.3 Analysis of Samples of Steam Generators and Concreted Sediments of Liquid Waste

The laboratory determination of the activity concentrations of gamma emitters Co-60, Nb-94, Ag-108^{m}, Cs-137, Am-241, beta emitters C-14, Ni-63, Sr-90 and alpha emitters Pu-238, Pu-239 and Pu-240 (Pu-239+240) in the samples taken from the steam generators and the concreted sediments of liquid waste has been performed within sub-activity 2.24. The results on the radionuclide activity concentration in the samples are presented in Tables 2.3.1 ÷ 2.3.2.

Table 2.3.1 Determination of gamma-ray emitters in concrete and metal samples

Sample No	Activity concentration, Bq/kg				
	Co-60	**Nb-94**	**Ag-108^{m}**	**Cs-137**	**Am-241**
1	460 ± 28	12.4 ± 0.9	< 30	317000 ± 20000	< 200
2	523 ± 32	17.5 ± 1.2	< 30	271000 ± 17000	< 200
3	193 ± 12	5.3 ± 0.8	< 30	103000 ± 7000	< 200
4	668 ± 40	22.4 ± 1.5	< 30	169000 ± 11000	< 200
5	763 ± 47	23.5 ± 1.5	< 30	206000 ± 13000	< 200
6	589 ± 37	20.6 ± 1.4	< 30	132000 ± 8000	< 200
7	445 ± 28	14.0 ± 1.2	< 30	133000 ± 8000	< 200
8	35.8 ± 4.7	< 2.0	< 30	65700 ± 4000	< 200
9	273 ± 18	9.4 ± 0.7	< 30	28000 ± 1700	< 200
10	97.3 ± 6.0	3.4 ± 0.6	< 30	15800 ± 950	< 200
11	66.2 ± 4.1	3.1 ± 0.6	< 30	29000 ± 1800	< 200
12	94.4 ± 5.8	3.0 ± 0.6	< 30	6370 ± 390	< 200
13	404 ± 25	12.3 ± 0.9	< 30	12200 ± 750	< 200
14	872 ± 53	25.0 ± 1.6	< 30	16400 ± 990	< 200
15	317 ± 20	9.1 ± 0.7	< 30	9800 ± 590	< 200
16	758 ± 47	22.3 ± 1.5	< 30	28200 ± 1700	< 200
17	24300 ± 1500	1730 ± 110	700 ± 50	8900 ± 540	400 ± 25
18	40.0 ± 2.5	< 2.0	< 30	134 ± 12	< 200
19	39300 ± 2400	216 ± 15	< 30	1000 ± 70	340 ± 21
20	246000 ± 15000	1120 ± 80	< 30	3070 ± 200	1270 ± 80

Table 2.3.2 Determination of beta-ray and alpha emitters in concrete and metal samples

Sample No	Activity concentration, Bq/kg				
	C-14	Ni-63	Sr-90	Pu-238	Pu-239+240
1	92600 ± 10800	8300 ± 1900	27700 ± 6500	12.6 ± 0.9	9.1 ± 0.6
2	66100 ± 7700	6300 ± 1400	13200 ± 2900	7.2 ± 0.5	7.0 ± 0.5
3	29600 ± 3600	2200 ± 500	735 ± 170	3.1 ± 0.2	2.7 ± 0.2
4	57600 ± 6700	9500 ± 2100	5000 ± 1100	4.4 ± 0.3	3.2 ± 0.2
5	63400 ± 8000	13000 ± 2900	12000 ± 2700	5.5 ± 0.4	4.0 ± 0.3
6	47600 ± 6000	9400 ± 2070	4100 ± 900	5.3 ± 0.4	3.8 ± 0.3
7	43400 ± 5400	7700 ± 1700	3600 ± 810	3.5 ± 0.2	2.5 ± 0.2
8	37500 ± 4700	89 ± 21	1850 ± 420	2.4 ± 0.2	2.0 ± 0.1
9	35100 ± 4400	2980 ± 680	348 ± 77	< 2	< 2
10	73500 ± 9500	1200 ± 280	270 ± 62	< 2	< 2
11	57300 ± 6700	530 ± 120	135 ± 30	< 2	< 2
12	121000 ± 14200	1300 ± 300	220 ± 49	< 2	< 2
13	55400 ± 6900	6500 ± 1500	293 ± 64	< 2	< 2
14	86400 ± 11000	13700 ± 3100	312 ± 70	< 2	< 2
15	126000 ± 15800	4200 ± 1000	206 ± 48	< 2	< 2
16	122000 ± 14400	11900 ± 2600	437 ± 98	< 2	< 2
17	175000 ± 22000	5500 ± 1300	510 ± 120	203 ± 14	246 ± 17
18	900 ± 200	< 30	< 20	< 2	< 2
19	142000 ± 17800	8600 ± 1900	166 ± 37	152 ± 11	206 ± 14
20	404000 ± 47000	49000 ± 11000	334 ± 78	530 ± 37	790 ± 55

2.4 Gamma-Ray Spectrometric Characterisation of Radiation Sources in Concrete Containers from the Tammiku Facility

Measurement of gamma-ray spectra from the concrete containers BE 251 and BE 252 with spent sealed sources removed from the Tammiku waste storage, which are stored at Interim Storage now, has been done by a portable spectrometer (Ortec instrument ISO-Cart-85). This work was done within sub-activity 2.24. The direct radiation measurements showed that there is some variation of both the ambient dose rate and the count rate in the full-energy peaks of gamma-ray spectra measured at the same distances from the container. This confirms the inhomogeneity of the activity distribution as well as the variation of shielding quality inside the containers leading to the different angular attenuation of the radiation. Therefore, simulation of the gamma spectra from containers has been performed by MCNP6.2 code package for various localisation of a radiation source inside the containers (see Fig. 2.4.1), the density of containers' wall concrete, the density of waste fill.

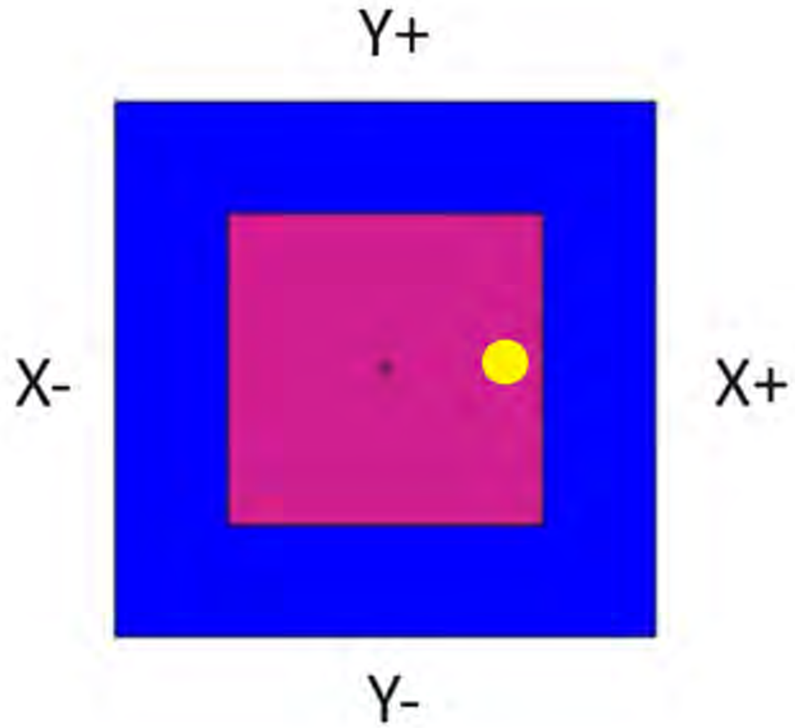


Fig. 2.4.1 Horizontal cross-section of concrete container with homogeneously distributed radioactive source (in yellow) placed closer to one wall

Comparing measurements of BE 251 container with simulated distributions, the closest configuration corresponds to 2.2 g/cm^3 density of wall concrete, 0.5 g/cm^3 density of waste fill and a Cs-137 source placed 15 cm sideways from the center of the container. This conservatively could correspond to 1000 GBq of Cs-137 source activity. For the BE 252 container, the similar source distribution gives 30 GBq of Co-60 source activity and 35,000 GBq of Cs-137 activity.

2.5 Gamma-Ray Spectrometric Characterisation of the Concrete Containers with the Control Rods of the Reactors

For purpose of determination of the activity of main gamma emitters of decommissioned control rods, the independent modelling of neutron spectrum using MCNP6.2 was performed to obtain the possible radionuclide inventory of the activated Eu_2O_3 control rods in the high enrichment nuclear fuel (20% initial enrichment by U-235) reactor VM-A , which were operated in Paldiski site since 1968 till 1989. This work was done within sub-activity 2.24.

Later on, the second MCNP6.2 modelling of containers with control rods material (Eu_2O_3) was performed in order to simulate the gamma detector response from calculated activated Eu source. The radionuclide inventory – especially Eu-152/Eu-154 ratio was used to assess the gamma spectrum properties outside 4 concrete containers of the dedicated waste. The calculated gamma spectrum lines were compared with experimentally obtained, the assumptions for determination of the activity of main gamma emitters of decommissioned control rods have been discussed and the reassessed activity values of Eu isotopes have been proposed.

The previously measured gamma spectra have been analysed by automatised ISO-CART procedure, and, most probably, the analysis was performed based on the highest gamma yield but low gamma energy peaks. If one takes into account the higher energy gamma lines, which escape from the container and are visible in the spectra, the Eu-152 activity is determined more accurately.

The new total activity of radionuclides Eu-152 and Eu-154 was determined with an uncertainty lower than the available one by simulation of radionuclide generation in the reactor and absorption of gamma rays in the containers. The new values of Eu isotopes activity **A** for the present time are provided in Table 2.5.1.

Table 2.5.1 New values of Eu isotopes activity for the 01.12.2023

Nuclide	Container			
	CRC-1	CRC-2	CRC-3	CRC-4
	A, TBq	A, TBq	A, TBq	A, GBq
Co-60	0.05 ± 0.02 TBq	0.2 ± 0.1 TBq	0.1 ± 0.05 TBq	0.14 ± 0.06 GBq
Cs-137	0.2 ± 0.1 TBq	5.1 ± 3.3 TBq	0.66 ± 0.44 TBq	43 ± 14 GBq
Eu-152	4.6 ± 1.6	31 ± 11	6.8 ± 2.8	0.2 ± 0.1
Eu-154	0.6 ± 0.2	3.1 ± 1.4	1.0 ± 0.4	< 0.05

2.6 Radiological Study of the Paldiski Site

Radiological study of the Paldiski site was done within sub-activity 4.6.

2.6.1 Natural radiation background on Pakri Peninsula

Preliminary measurements have shown that the soil, at least in some areas of the territory of FPNC, has even less specific activity of Cs-137 (5 Bq/kg) than the soil from the regular Estonian land approximately 50 km from Paldiski (11 Bq/kg).

The data from literature and maps have been analysed to find two uninhabited sites on Pakri Peninsula with similar geomorphological characteristics and biota. Two sites have been selected: one site (east reference site) is located at approx. 2 km east (with coordinates 59.349 N, 24.156 E) from the MB of FPNC (around sampling points with labels 181-184 on Fig. 2.6.1; separate water sampling point No. 184 was chosen as there is no surface water on the site) while the second site (north reference site) is located in the northern direction (59.373 N, 24.115 E) at about from 1 km distance MB (around sampling points with labels 185-188). Acceptability of the sites to be as reference sites for determination of the natural radiation background has been confirmed after visual inspection.

Fig. 2.6.1 Location of sampling places on the reference sites near FPNC

The gamma-ray dose rate measurements have been performed on both sites in several areas of about 100 m^2 each to ensure that no substantial variation is present. The typical ambient dose rate values ranged from 0.10 µSv/h to 0.12 µSv/h at both sites.

For the determination of representative sampling places on the reference sites, the results of the gamma-ray dose and spectroscopic measurements have been used. 5 soil samples from different depths, 2 grass, 2 water samples have been taken in every site for measurement of activity concentrations of gamma emitters (Be-7, K-40, Cs-137, Ra-226, Pb-210) and determination of difficult-to-measure radionuclide concentrations in the laboratory.

Analysis of the measurement results show that the biggest specific activity of gamma emitters on both reference sites differs less than 3 times and it is far below the clearance levels. The activity of other gamma emitters as well as activity of alpha and beta emitters in samples was below MDA (C-14 < 1.5 Bq/kg, Co-60 < 0.5 Bq/kg, Ni-63 < 1.8 Bq/kg, Sr-90 < 0.9 Bq/kg, Nb-94 < 0.5 Bq/kg, Am-241 (gamma) < 8 Bq/kg, Pu-238 < 0.005 Bq/kg, Pu-239+240 < 0.005 Bq/kg). Therefore, one can conclude that measurements on these sites give adequate representation of radiation background on Pakri Peninsula.

2.6.2 Radiation survey on the Paldiski site

Two survey units have been distinguished on the territory of FPNC after analysis of historical data of activity on the site. The area between the metal fence and the concrete fence, as shown in Fig. 2 (area between red lines), was preliminary classified as unaffected one and constituted the survey unit No. 1. The area inside the metal fence, as shown in Fig. 2.6.2 fence (inside red lines), was preliminary classified as possible affected one and constituted the survey unit No. 2.

Fig. 2.6.2 Unaffected area (shown left, 20.82 ha) and possibly affected area (shown right, 8.359 ha) areas of FPNC

A radiological survey of the territory of FPNC has been implemented by scanning the whole territory (accessible by walking part of survey unit No. 1, without entering into the young forest, and survey unit No. 2), which can be passed by slowly nonstop walking (velocity is about 0.5 m/s), with portable hand-held measuring instruments to determine the areas with the gamma-ray dose rate higher than 0.20 µSv/h (affected areas). The distance between parallel lines was set to 5 m.

During the scanning of the territory of survey unit No. 1 (tracks are shown on Fig. 2.6.3), the additional measurements of the gamma-ray dose rate have been fulfilled by stopping after every 50 m at some point on the line. As most parts of the young forest cannot be scanned, measurements have been done along its perimeter. The distance between measurement points have been equal to about 5 m in this case. The average ambient dose rate value in the survey unit No. 1 was determined as 0.13 ± 0.02 µSv/h. No area has ambient dose rate value bigger than 0.20 µSv.

The total number of parallel lines in the survey unit No. 2 (tracks are shown on Fig. 2.6.4) was 42. The length of the parallel lines varied from 90 m to 450 m depending on the location to be scanned. The territory of survey unit No. 2 has been scanned slowly walking in parallel lines and stopping after every 5 m to record the result of the gamma-ray dose rate measurement around the stopping point. The readings have been averaged at each parallel line, the standard deviation and the relative error were calculated and provided. The results of ambient dose rate measurement data are summarised in Table 2.6.1.

Fig. 2.6.3 Location of territory scanning tracks in unit No. 1

Fig. 2.6.4 Location of territory scanning tracks in unit No. 2

Table 2.6.1 Ambient dose rate at parallel lines in the survey unit No. 2

Line No.	Number of readings per line	Gamma-ray dose rate		
		Mean, µSv/h	Standard deviation, µSv/h	Standard uncertainty, % (k = 2)
1	19	0.132	0.004	8.7
2	19	0.132	0.005	9.8
3	19	0.128	0.003	8.2
4	19	0.132	0.004	9.0
5	19	0.130	0.004	9.1
6	19	0.129	0.004	8.8
7	76	0.130	0.004	8.8
8	76	0.132	0.005	9.3
9	76	0.130	0.004	8.8
10	76	0.130	0.004	8.8
11	76	0.129	0.004	8.5
12	76	0.139	0.010	15.2
13	76	0.141	0.010	15.0
14	47	0.132	0.005	9.0
15	47	0.132	0.005	9.3

16	47	0.132	0.004	8.7
17	47	0.132	0.005	9.2
18	47	0.132	0.005	9.0
19	58	0.132	0.004	8.5
20	58	0.132	0.005	9.3
21	76	0.131	0.005	9.4
22	76	0.132	0.010	16.1
23	69	0.135	0.024	36.7
24	69	0.132	0.004	9.3
25	83	0.132	0.004	9.0
26	90	0.132	0.005	9.4
27	90	0.129	0.005	10.5
28	90	0.129	0.004	8.8
29	90	0.131	0.004	8.5
30	90	0.131	0.004	8.4
31	90	0.130	0.003	7.8
32	88	0.132	0.004	8.8
33	88	0.132	0.004	8.7
34	90	0.132	0.005	9.3
35	90	0.132	0.005	9.3
36	90	0.132	0.005	9.3
37	90	0.131	0.005	9.5
38	90	0.132	0.005	9.4
39	90	0.132	0.005	9.4
40	90	0.132	0.005	9.4
41	90	0.131	0.004	8.9
42	90	0.132	0.005	9.4
Total	**2896**			

Some increase of an ambient dose rate of up to 0.15-0.16 µSv/h was determined in a few points near MB, which are caused, the most probably, by radiation from MB. A similar effect can be noticed for the lines, which cross the contaminated area (AREA 2). AREA 2 is the only contaminated area on the territory of FPNC. The result of scanning exceeded 0.20 µSv/h in AREA 2. AREA 2 is located at a distance of about 30 m from MB to the North direction (see Fig. 2.6.5). Contaminated area is marked in Fig. 2.6.5 by two colours: green and yellow. Yellow marking shows a place on the asphalt road, while the place with the detected higher dose rate on the grass is marked green. AREA 2 also includes a storm drain well (well 1 on Fig. 2.6.6). Additional measurements of the gamma-ray dose rate and in-situ gamma spectrometry measurements were done to define the affected area with better than 0.5 m uncertainty. The dose rate in AREA 2 on the grass surface varied from 0.11 µSv/h to 0.22 µSv/h (Fig. 2.6.6). This shows that the contamination on AREA 2 is non-homogeneous, there are smaller areas with no contamination inside a larger area with enhanced levels which on the grass surface can be evaluated as high as a doubled background radiation level. Furthermore, the highest dose rate measured at the spot located on the asphalt road was 0.24 µSv/h. The total contaminated area is up to 30 m^2.

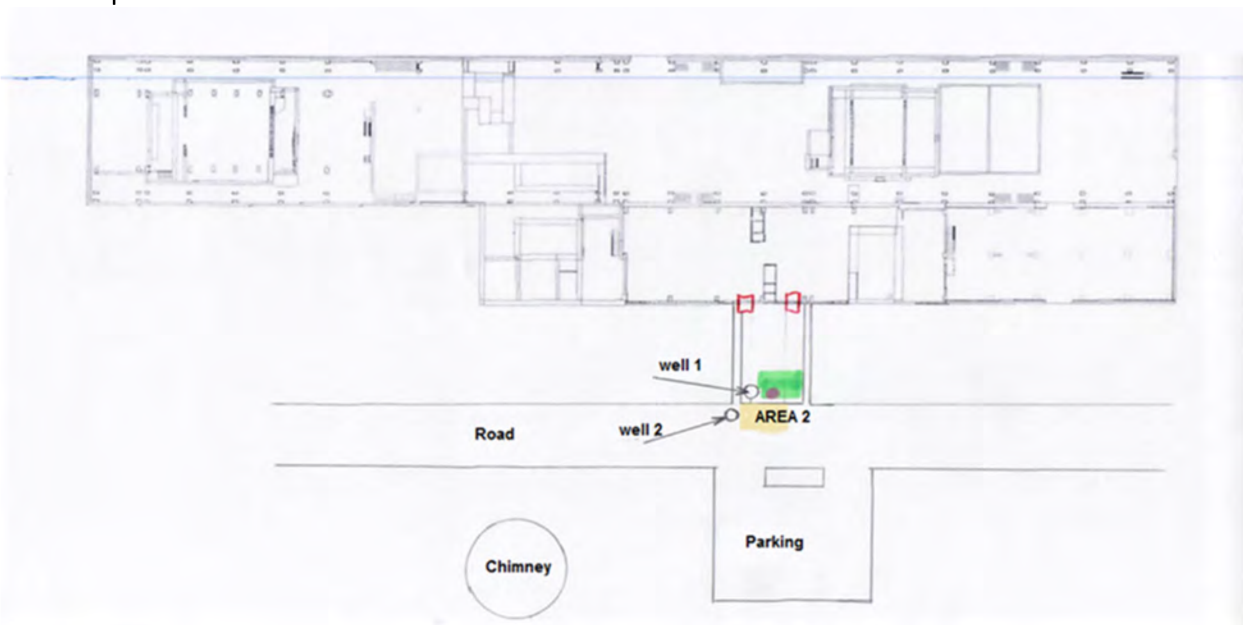


Fig. 2.6.5 Contaminated area (AREA 2) at the territory of FPNC

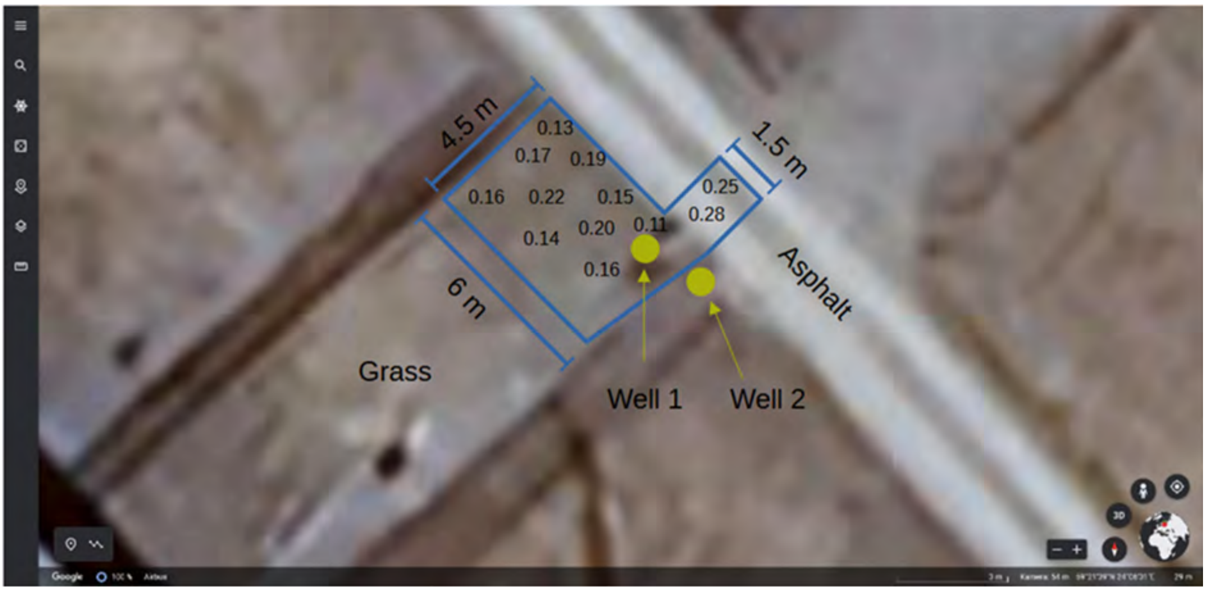
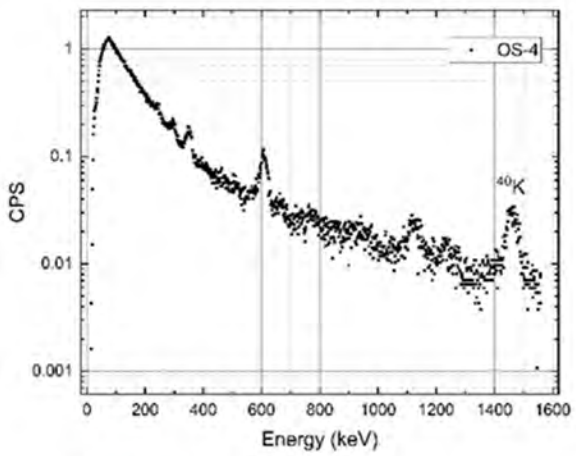


Fig. 2.6.6 Contaminated area (AREA 2) at the territory of FPNC, gamma-ray dose rate

The average ambient dose rate value in the survey unit No. 2 was determined as 0.13 ± 0.02 μSv/h. The higher variation of the observed dose rate was governed by elevated levels at a number of points where natural radioactivity from massive stones was determined. For instance, the dose rate value at the surface of some stones was measured as high as 0.6 – 0.7 μSv/h.

The second campaign has been targeted to confirmation of preliminary results and more detailed characterisation of the contaminated area. The gamma-ray spectrometric *in situ* measurements have been done to get additional information on the gamma emitting radionuclides by gamma spectrometer with a $CeBr_3$ detector. There have been spectrometric measurements done at various depths inside well 1. 4 spectrometric measurements have been done in the area where no contamination was found and 4 measurements in AREA 2. It is important to note that these measurements were qualitative, dedicated for gamma emitters – which primary determines the dose rate – identification and later on the quantitative measurements of samples were performed both using destructive and non-destructive nuclear spectrometry methods. The normalised in situ gamma spectra of natural background and of the contaminated AREA 2 the near well have been measured (Fig. 2.6.7).



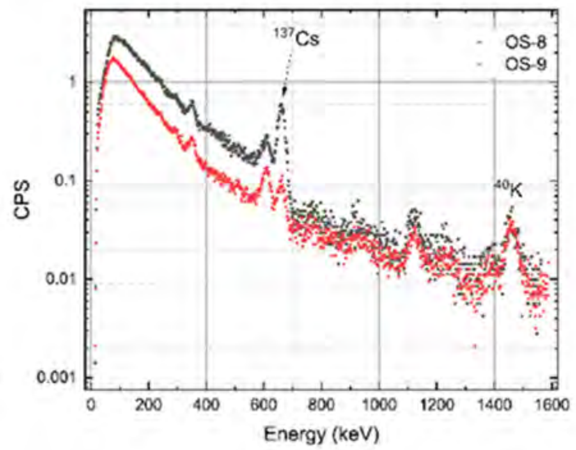


Fig.2.6.7 Normalised in situ gamma spectrum of natural background radiation (left) and contaminated area near the well (right)

2.6.3 Sampling

For the determination of representative sampling places, the results of the dose and spectroscopic measurements have been used. The plots with a similar gamma-ray dose rate and particular gamma-ray nuclide activity have been united in one sampling unit. Special sampling equipment, such as drills, has been used. 42 samples have been taken for radiological analysis. The soil has been sampled from the surface of the earth and from a depth of up to 30 cm (0-10 cm, 10-20 cm and 20-30 cm).

Three soil samples have been taken from the grass lawn where dose rate reached 0.2 µSv/h at the contaminated AREA 2 near the asphalt road. Results showed that the activity concentration of Cs-137 was getting higher with depth and the dose rate had increased correspondingly, at those mentioned layers: (427 ± 52) Bq/kg and (0.2-0.3) µSv/h, (2660 ± 320) Bq/kg and (0.6-0.7) µSv/h, (11450 ± 1380) Bq/kg and (1.4-1.6) µSv/h.

2.6.4 Determination of activity concentration of nuclides

Activity concentrations of all gamma-ray emitters in all samples have been determined in the laboratory. Results of the measurement of the specific activity of gamma-ray emitters in samples taken from unaffected areas by local contamination in the territory of FPNC show that the radiation situation in unaffected areas by local contamination in the territory of FPNC is typical for Pakri Peninsula.

The comparison of ratios of specific activity of various Pu isotopes with the results of the simulation of specific activity of Pu isotopes in the nuclear fuel (the results of the simulation are presented in subsection 2.2) shows that contamination of AREA 2 is of the reactor origin. This finding is supported by the fact of presence of the pipe below contaminated area, which was used for transportation of contaminated water from the reactors to the liquid water treatment facility. The depth profile of the contamination (an increase of specific activity of artificial radionuclides with an increase of the depth) also points out relation of contamination of AREA 2 with underlying pipe. The results of destructive analysis performed in the samples from the contaminated area have been used for the determination of a statistically proven NV.

2.6.5 Determination of the nuclide composition

A radiological survey of the territory of FPNC has shown that all the territory of survey unit No. 1 and almost all the territory of survey unit No. 2, except about 30 m^2 area (AREA 2), are unaffected by contamination. Radionuclide C-137 is the only gamma emitter, which is related to the fission of uranium and plutonium isotopes and was observed in areas unaffected by local contamination. Typical sources of Cs-137 in such areas are global fallouts (deposition of radioactive aerosols from the atmosphere after nuclear weapons tests or precipitation after accidents at nuclear power plants). Typical representatives of natural radioactivity are Pb-210, Ra-226 (represent the U-238 decay chain), As-228 (represents the Th-232 decay chain) and K-40. Activity measurement results show that values of concentration of nuclides in unaffected areas of FPNC are typical of background levels in the Pakri Peninsula and far below ECL.

As the contamination origin in AREA 2 is of reactors of FPNC, a common NV (presented in subsection 2.2) was determined for all contaminated areas. The key nuclide Cs-137 is suitable for the characterisation of the total contamination (contamination from the reactors and waste management) in the territory of FPNC in all other contaminated areas (including reactors compartments, contamination area in MB).

Activity concentrations of measured radionuclides (averaged, maximal and minimal detected values) in the soil samples from AREA 2 and for comparison exclusion and release activity concentrations as well as activity concentrations for reuse or recycling of a building are presented in Table 2.6.2. Provided in Table 2.6.3 average activity concentration values, which are derived from the measured Cs-137 activity concentration, for all long-term radiation safety relevant radionuclides, except Cs-137 itself, are lower than the ECL. However, conservative activity concentration values (upper limit) of radionuclides Eu-152 and E-154 are higher than the ECL. Upper limits of radionuclides Cs-137, Eu-152 and Eu-154 are also higher than release of activity concentrations of radionuclides for reuse or recycling of a building. Cs-137 activity concentration exceeds the levels of for reuse or recycling of a building by a factor of 5, also for Eu isotopes the upper limit is exceeding the levels of reuse or recycling by factor 2 (for Eu-154) and 8 (for Eu-152) if taking into account the conservative approach.

Average and conservative (upper limit) activity concentrations in the contaminated area of the territory of FPNC (AREA 2) based on the NV are presented in Table 2.6.3.

Table 2.6.2 Activity concentrations (Bq/kg) of radionuclides in the samples from AREA 2

Nuclide	Average	Max	Min	ECL
Co-60	0.6±0.04			1.0E+02
Cs-137	(4.9±0.8) ´10^3	(1.8±0.3) ´10^4	430±50	1.0E+02
C-14	40±9			1.0E+03
Pu-239/240	0.06±0.01	0.08±0.01	0.04±0.01	1.0E+02
Pu-238	0.06±0.01	0.12±0.01	0.02±0.004	1.0E+02
Am-241	0.04±0.01	0.08±0.02	0.01±0.003	1.0E+02
Ni-63	7±1	15±3	2.3±0.5	1.0E+05
Sr-90	33±5	87±15	5.4±1	1.0E+03
Ra-226	150±20	260±30	86±12	1.0E+03
Pb-210	5±1	7±0.4	3.3±0.2	1.0E+03

Table 2.6.3 Nuclide activity concentrations (Bq/kg) in the contaminated area based on NV

Nuclide	Average	Upper limit	ECL	RRL
C-14	2.7E+02	4.7E+02	1.0E+03	1.0E+04
Ni-59	1.5E-01	3.5E+01	1.0E+05	1.0E+06
Co-60	2.6E-01	3.7E-01	1.0E+02	1.0E+04

Ni-63	1.3E+01	3.1E+03	1.0E+05	1.0E+06
Sr-90	3.7E+01	1.1E+02	1.0E+03	1.0E+03
Nb-94	1.4E-02	3.3E+00	1.0E+02	1.0E+02
Cs-137	**4.9E+03**	**6.5E+03**	1.0E+02	1.0E+03
Eu-152	3.6E+00	**8.4E+02**	1.0E+02	1.0E+02
Eu-154	8.1E-01	**1.9E+02**	1.0E+02	1.0E+02
Pu-238	1.3E-01	2.0E-01	1.0E+02	1.0E+02
Pu-239	1.1E-01	3.9E-01	1.0E+02	1.0E+02
Pu-240	3.2E-02	1.1E-01	1.0E+02	1.0E+02
Am-241	2.0E+01	2.7E+01	1.0E+02	1.0E+02

2.6.6 Characterisation of the Paldiski site according to contamination level

The results of characterisation of the territory of FPNC according to contamination levels are shown in Fig. 6. For characterisation of the largest part of the territory, excluding the area of the young forest, have been used statistically reliable measurement results. The contamination of the young forest area has been assessed by extrapolation of results of measurements in its vicinity.

All areas between the metal fence and the concrete fence (survey unit No. 1), as shown in Fig. 2.6.8, as well as the area inside the metal fence (survey unit No. 2) are classified as unaffected except the contaminated AREA 2 marked by a red circle. AREA 2 is located around the point with latitude 59.360902 N and longitude 24.108692 E. The contaminated area is equal to about 30 m^2. The gamma-ray dose rate on the surface of the contaminated area varies from 0.11 μSv/h to 0.24 μSv/h showing non-homogeneous contamination character. The depth of the contaminated soil exceeds 4 m. The deeper layers are more contaminated than the surface layer.

Fig. 2.6.8 Characterisation of FPNC according to contamination levels

2.7 Radiological Study of the Area around a Former Special Sewerage Well

For detailed characterisation of the bulk of the contaminated area, the installation of new boreholes was carried out within sub-activity 2.24. The ground in the contaminated area is rather complicated and full with obstacles of various origins (stones, remains of previous buildings, etc.) preventing the standard drilling process and, as a consequence, the penetration of a drill to a necessary depth, normally up to 4-5 m, to make possible an efficient assessment of the contamination extent.

If the drilling was not possible below 1 m, then a different point for drilling has been chosen. In total, there were 11 attempts fulfilled to drill the ground in the area at selected points. Rock samples were taken almost in all cases.

A 113 mm PVC pipe (inner diameter of 103 mm) was inserted into every borehole. The gamma spectrum at various depths of a borehole was measured with a $CeBr_3$ gamma detector. It was found that Cs-137 is the gamma emitter whose activity overcomes all other artificial ones in all boreholes.

The results of gamma spectrometry with a SCIONIX $CeBr_3$ scintillation detector were interpreted using simulation of Cs-137 activity concentration in a rock inside the borehole by MCNP6 code package. The calculation model contains a $CeBr_3$ detector with protection, a plastic tube and a layer of 50 cm thickness rock. For plastic material was used PVC, for rock material – SiO_2 and for detector protection – Teflon and Aluminium. The simulation shows that the detector can register gamma quanta from rock layer approximately up to 50 cm thick.

At least two samples from every borehole were analysed in the laboratory to determine radionuclide concentrations. Totally 24 samples were investigated.

If gamma emitters are identified in a sample, the activity concentrations of gamma emitters, beta emitters C-14, Ni-63, Sr-90 and alpha emitters Pu-238, Pu-239+240 were determined in 12 samples. The only nuclide, which activity concentration in samples exceeds a value for exemption or clearance of materials is Cs-137.

Taking into account the information about activity concentration of rock samples, the localisation of the area to be excavated can be approximately specified. Location and preliminary dimensions of the excavation area based on conservative approach are shown in Fig. 2.7.1. Dimensions of the excavation area are: length (parallel to the asphalt road and MB) – 15.93 m; width (perpendicular to the asphalt road and MB) – 11.86 m.

The resulting volume to be excavated is: 15.93 m × 11.86 m × 5 m = 944.65 m^3.

Requirement No. 48 of the IAEA Safety Standards Series No. GSR Part 3 indicates “that the remedial and protective actions are justified and that protection and safety are optimized.” “Remedial or protective actions are expected to yield sufficient benefits to outweigh the detriments associated with taking them, including detriments in the form of radiation risks. The implementation of remedial actions (remediation) does not imply the elimination of all radioactivity or all traces of radioactive substances. The optimization process may lead to extensive remediation but not necessarily to the restoration of previous conditions.”

Therefore, it is recommended to assess the situation in the future and look for an appropriate and justified solution.

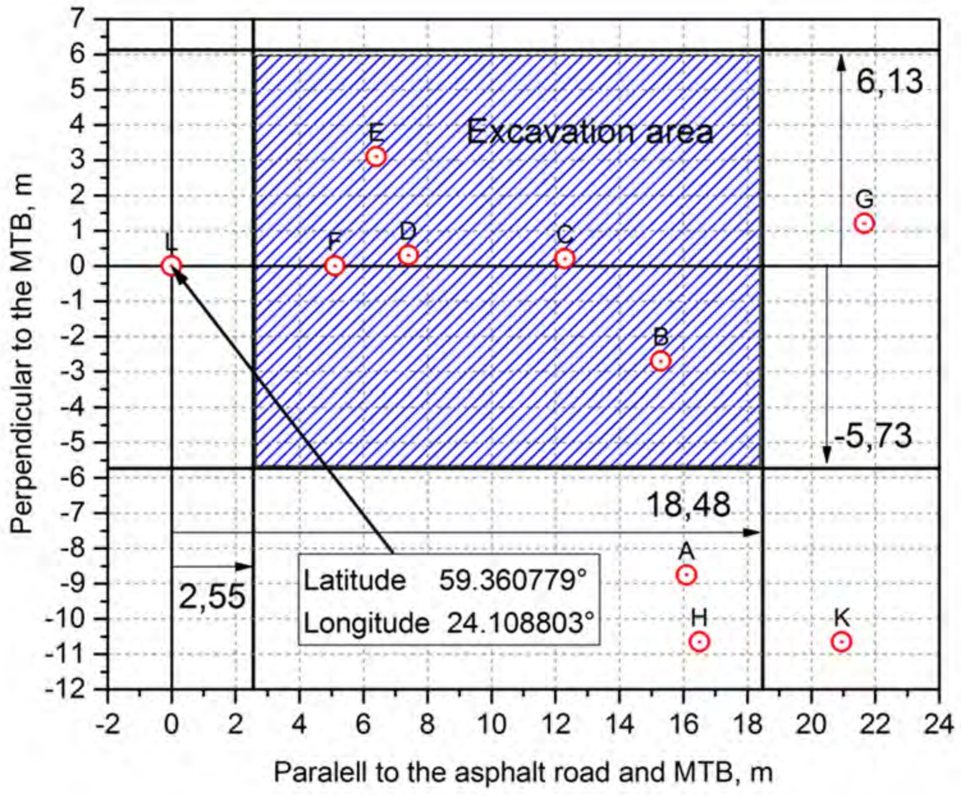


Fig. 2.7.1 Localisation of the excavation area of the contaminated rock

The average activity concentration of contaminated rock in the excavation area of the contaminated rock is equal to 1294 Bq/kg of Cs-137, the maximum measured activity concentration is 7300 Bq/kg of Cs-137. The activity calculation results of the rock to be excavated are summarised in Table 2.7.1.

Table 2.7.1 Cs-137 activity of the rock to be excavated from contaminated area

Characteristics of the rock			Activity concentration, Bq/kg		Total activity, Bq	
Volume, m^3	Density, kg/m^3	Mass, kg	Average	Maximum	Average	Maximum
944.65	1600	1.51E+06	1294	7300	1.96E+09	1.10E+10

3. STUDY OF DECOMMISSIONING ALTERNATIVES

3.1 Zero Alternative

3.1.1 Review of "zero alternative" options

"Zero alternative", i.e. preservation of mothballed RCs and storage of the packaged RW in the Interim Storage at the FPNC was analysed within sub-activity 5.1.

This concept is considered as an alternative to decommissioning RCs. The choice of the "zero alternative" is made from the following possible options:

- Option 1: prolonging safe enclosure of the reactors for additional time period and postponing decommissioning works;
- Option 2: leaving the reactors not dismantled for ever, i.e. prolonged storage waiting until clearance levels are reached;
- Option 3: on site disposal (entombment).

3.1.2 Option 1. Prolonging safe enclosure of the reactors for additional time period and postponing decommissioning works

According to the international experience, the main reason for deferred dismantling is the decay of short-lived nuclides. It is well known that the main (or even the only one) advantage of the "safe enclosure" options is reduction of radioactivity. Decay of short-lived radionuclides results in:

- Significant reduction of doses during dismantling;
- Reduction of amount of the waste, as well as reduction of the disposal facility size;
- Simplification of dismantling process (no needs for robots or remotely managed tools as the dose rate is insignificant);
- Simplification of waste management process.

The main question is to what extent it is expedient to extend the exposure time of the RCs. At the same time, it is understood that the extension of the holding time of the RCs in no way affects a significant decrease in the activity of spent sealed sources. The main dose-forming nuclide for external exposure of personnel is Co-60. With a total activity of Co-60 in the RC No. 1 1.93 E+11 Bq (2039 year), the dose of external exposure of personnel during the dismantling work will decrease by more than 4000 times over 60 years, which will make it possible to perform most of the work without the use of remotely controlled equipment. Therefore, the holding period for RCs for option 1 is considered until 2100.

Throughout the entire period of operation (during the "zero alternative" period until 2100 year plus 10 years for dismantling works), the MB will perform the specified functions and maintain the necessary performance characteristics. The main requirement that determines the reliability of the construction of a facility is its suitability for its intended purpose and the ability to maintain the necessary operational qualities during the established period of operation. These include:

- Guarantee the safety of human health and life of people, property and the environment;
- Maintain the integrity of the facility and its main parts and fulfil other requirements that guarantee the possibility of using the facility for its intended purpose and the normal functioning of the technological process, including meeting the requirements for the reliability of building structures and foundations, their heat and sound insulation properties and tightness;
- Ensure the possible development of the facility and its adaptation to changing technical, economic or social conditions;
- Create the necessary level of convenience and comfort for users and operating personnel, including the requirements for maintaining the climatic regime in the premises (air exchange, temperature, humidity, illumination level, etc.), as well as accessibility for inspections and repairs, the possibility of replacing and upgrading individual elements;
- Limit the level of risk by fulfilling the requirements for fire protection, non-failure operation of protective devices, reliability of life support systems and networks, survivability of building structures.
- The building structures of the MB, the structures of sarcophagi and the Interim Storage act as barriers to the possible spread of radioactive substances and ensure the safe operation of the building as a whole.

Based on the experience of assessing the durability of building structures of such construction, the recommended value of the time from the moment of construction of structures to the moment of exhaustion of its resource for industrial buildings made in reinforced concrete structures is estimated at about 100 years.

The construction of the MB was carried out in the early 1960s. At the time of the 2022 survey, the age of the structures was estimated at about 60 years.

Taking into account the work on the complete reconstruction of the MB that was completed in the early 2000s and the possibility of extending the design life of the facility, it is conservatively assumed that the life of the MB under "zero alternative" will expire by 2100 (100 years from complete reconstruction date).

The prospect of extending the design life beyond 100 years should be considered on the basis of a comprehensive survey and assessment of the residual resource of building structures and the building as a whole at the end of the building's design life.

3.1.3 Option 2. Leaving the reactors as they are for ever, i.e. prolonged storage waiting until clearance levels are reached

Option 2 proposes maintaining the *status quo* for the time necessary to reduce the activity of radionuclides to levels of withdrawal from regulatory control.

In order to estimate the time required to achieve the levels of release from regulatory control, the main RW accumulated on the site are considered below:

- Equipment in RCs;
- Spent sealed sources, which were placed in the RC No. 1 during the conservation works;
- RW placed in the Interim Storage.

Almost all (99%) amount of activity of radionuclides in the equipment of RCs is determined by the radioactivity of the reactors and reactor internals. Since the mass characteristics of the equipment are known, and the equipment itself is considered as a whole, the exemption and clearance levels for radioactive substances established by the Estonian regulation can be taken as a criteria for the release of equipment from regulatory control.

As follows from the results of activity data analysis, the main RW accumulated at the Paldiski site cannot be released from regulatory control in the foreseeable future (tens of thousands of years that are required to achieve the release criteria), which excludes this option from the scope of the review.

3.1.4 Option 3. On site disposal (entombment)

Option 3 proposes the conversion of the existing building with RCs into a facility for RW disposal. The MB, which houses the storage facility of RW and RCs, is a surface structure. RW in the Interim Storage, equipment and materials inside RCs, and spent sealed sources contain long-lived radionuclides.

According to the IAEA classification of RW (GSG-1): «Intermediate level waste (ILW): Waste that, because of its content, particularly of long-lived radionuclides, requires a greater degree of containment and isolation than that provided by near-surface disposal. However, ILW needs no provision, or only limited provision, for heat dissipation during its storage and disposal. ILW may contain long-lived radionuclides, in particular, alpha-emitting radionuclides that will not decay to a level of activity concentration acceptable for near-surface disposal during the time for which institutional controls can be relied upon. Therefore, waste in this class requires disposal at greater depths, of the order of tens of metres to a few hundred metres».

Because disposal of long-lived waste in near-surface disposal facilities is considered unsafe, this option is excluded from consideration.

3.1.5 Analysis of factors influencing the choice of an option 1 “Prolonging safe enclosure of the reactors for additional time period and postponing decommissioning works”

Evaluation of economic factors

When assessing the financial costs for this option, it was assumed that no measures would be taken to create an infrastructure for RW management (a conditioning workshop and the creation of a RW disposal facility) and the refusal to carry out work to dismantle equipment and building structures after 2040. That is, this scenario assumes non-intervention in the situation and the continuation of the operation of the MB in its current state. It can be said that from a financial point of view, for the entire period of keeping RCs in a state of conservation, this scenario may look more attractive than the base one (an option that involves the dismantling of equipment and building structures and then management of the resulting waste). Since for the period at least until 2100, there will be a need to provide only operating costs for maintaining the building and supporting systems in working condition. These costs are comparable to the cost of operating the main technology building in any other scenario that assumes a holding period until 2040.

However, given that both building structures and technological systems cannot ensure the safe long-term storage of RW (until the conditions for release from regulatory control are met), it will be necessary to dismantle RCs and MB, eventually set up facilities for the management of accumulated RW and waste disposal in accordance with established norms.

The economic factors for the implementation of the "zero alternative" include the following components:

- Operating costs between 2040 and 2110 (during the "zero alternative" period until 2100 year plus 10 years for dismantling works);
- Refurbishment of the MB for safe storage of the RCs and RW.

The operating costs for the maintenance of the MB and engineering systems for 60 years are estimated to be up to 39 million euros (considering the current annual budget of about 650,000 euros). The costs include engineering surveys, regular repairs, replacements engineering systems, labour costs.

Additionally, 75 million euros will be needed for major repairs and refurbishment of the MB for safe storage of the RCs and RW.

Expenses for the dismantling of the RCs, the MB, and the removal of containers from the storage facility for disposal, the creation of infrastructure for the management of radioactive and non-radioactive waste should be assessed in the decommissioning project of the MB. For work carried out after 2100, inflation will be the determining factor.

This cost does not take into account the costs of RW disposal, which are produced during the dismantling process. With an expected amount of RW of 350 tons (100–150 m^3) produced during dismantling in 2040 and subject to disposal, the cost of disposal will be from 3.12 million euros. When implementing the zero option, the amount of waste to be disposed of may decrease by 2100 by no more than 20%, because RC equipment (reactor vessels with internals, shielding tanks, etc.) that has been subjected to neutron irradiation contains long-lived nuclides that are subject to disposal in geological formations.

Assessment of radiation factors

It is supposed that the radiation criteria for releases and discharges specified in Regulation No. 40 “Conditions for Exclusion and Release of Radioactive Substances Used or Generated in Radiation Operations and Requirements for Requests for Exclusion and Release” for additional time period will be met by applying a special work technology, maintaining safety barriers and controlled for compliance by a radiation monitoring system. Under normal operation conditions, radiation exposure of the population and the environment is not expected. But if the “prolonged zero alternative” is implemented, the waste inside the building will not be completely immobilised for a long time. It can cause environmental pollution and exceed the permissible levels of public exposure as a result of the failure of existing engineering barriers.

During normal operation, the most dangerous work, from the perspective of personnel exposure, will be dismantling of equipment of RCs and management of the resulting RW. Not exceeding the annual dose limit will be ensured by exposure planning, an exposure time limitation and the use of additional shieldings as well as dose rate control at workplaces and individual dosimetric control of personnel. Taking into account the significant decrease in the activity of Co-60 by 2100, if the “prolonged zero alternative” is implemented, it’s estimated the collective dose will not exceed 140 man-µSv. This will mainly occur due to the need to extract various types of RW stored in storage facilities, radiation sources from building structures, during the dismantling of reactor vessels.

There is a lot of safety-related uncertainties due to relatively limited knowledge about RCs. For example, over a long period of long-term storage, possible corrosion processes inside the RC can lead to the risk of leakage of radioactive substances into the environment. Elimination of the consequences of such an accident will lead to additional radiation exposure of personnel.

Assessment of non-radioactive factors

It can be noted that there is no negative impact of non-radiation factors on the environment, since the implementation of the “prolongation zero alternative” concept does not assume any additional activities associated with the generation of non-radioactive waste.

Under “prolonged zero alternative” the structures will be maintained in working condition, without any dismantling work for the entire period until 2100 and without any additional emissions and discharges during the normal operation.

Regulatory compliance

When considering the “prolonged zero alternative” option, various options for the decommissioning of nuclear installations laid down in the IAEA safety guides should be taken into consideration. In accordance to IAEA recommendations, disposal, during which the entire nuclear installation or part of it is encased in structurally robust materials, is not considered as a decommissioning strategy and is not an acceptable option in the event of a planned shutdown. It can only be considered as a possible solution in exceptional circumstances (e.g. after a severe accident).

The analysis of RW in the RCs and the Interim Storage shows that according to the classification currently used in Estonia, they hold a significant amount of low – and intermediate level long-lived RW. In accordance with the IAEA recommendations (GSG-1), medium-level waste containing large amounts of long-lived radionuclides is unacceptable for near-surface disposal and requires disposal at a depth of tens to several hundred metres. Consequently, without the dismantling of structures, removal and appropriate conditioning of all RW that do not meet the near-surface disposal requirements, such decommissioning option cannot be accepted.

The implementation of the zero option with prolonging safe enclosure of the reactors assumes that for a long time (more than 60 years) there is a radiation hazardous object on the territory of the FPNC. Safety barriers must be maintained at the facility in a safe condition.

Modern experience shows that the international situation in the world has changed dramatically over the past 20 years. The nuclear facilities can become targets of a terrorist act or military attacks of other countries. The risk of an impact on the environment, personnel and the public will be significantly higher than the option of completely dismantling the building and its contents within a shorter period.

Also it should be borne in mind that without solving the issue of safe waste disposal or postponing such a decision for a long time the "prolonged zero alternative" scenario directly contradicts the IAEA recommendations and the obligations assumed by Estonia under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management in part of not shifting the burden to future generations.

3.2 Comparison of Alternatives

3.2.1 Decommissioning options

The comparison of two decommissioning options for the reactor facilities of the FPNC in order to achieve the final state was done within sub-activity 5.2. These options are as follows:

- Option A: the dismantling of the RCs, including reactor equipment and structures; cutting the resulting components into small fragments. In this case, it is assumed that the reactor vessel will be fragmented into small parts and shipped/transported for disposal in standard containers;
- Option B: the dismantling of the RCs, including reactor equipment and structures; cutting the resulting components into small fragments. In this case, the reactor vessels will not be fragmented but sent intact for disposal in special containers.

3.2.2 Description of Decommissioning Options

Option A – Dismantling and fragmentation of reactor vessels

In accordance with option A, the decommissioning of the FPNC reactor installations will be carried out in several stages, including final shutdown; conservation; holding period and dismantling.

To date, the “final shutdown” and “conservation” work has been completed at the FPNC. RCs are in long-term storage (the stage of “holding period”).

During the holding period, the residual activity of the most radioactive equipment will be significantly reduced, which will allow the complete dismantling of the RCs without the risk of significant exposure of the personnel and without the need for expensive robots.

By the beginning of the “dismantling” stage, a RW treatment center must be established. By that time, the repositories for disposal of containers with conditioned RW that have been generated during the dismantling of RCs and temporarily stored in the MB should be built. Installations for RW decontamination, conditioning and packaging should be installed.

At the “dismantling” stage, the site must be cleaned from existing buildings and structures and the site will be brought to “greenfield site” condition. During dismantling, the following will be performed:

- The existing RW packages that are currently stored in the MB will be extracted and sent for disposal;
- The contaminated and uncontaminated equipment and structures, including the RCs, as well as auxiliary equipment located in existing buildings will be dismantled;
- RW will be managed and sent for disposal;
- Non-radioactive waste will be managed and sent for reuse or disposal;

- The existing buildings and structures: MB, ventilation chimney, entrance, water storage tank, biological wastewater treatment plant, engineering networks and site fencing will be demolished;
- Reclamation of the territory will be carried out.

Performance of dismantling works assumes initial dismantling of the RCs followed by cutting structures and equipment into small fragments (with fragmentation of reactor vessels) so that the final volume of waste is minimised and standard containers can be used. Subject to the results of radiation measurement, the waste from RCs will be sorted out and separated into radioactive and non-radioactive waste.

RW will be sent to RW management facilities for subsequent decontamination, fragmentation and packaging. RW will be segregated into Low-Level waste and Intermediate-Level waste, taking into account the established Waste Acceptance Criteria for near-surface disposal. Mixing of the different level wastes should be avoided. After characterisation, the conditioned (packaged) RW will be sent for disposal. Non-radioactive waste, after being released from regulatory control, will be sent for processing and reuse or disposal as industrial waste.

Option B – Dismantling without fragmentation of reactor vessels

Under option B, the RCs at the FPNC will be dismantled in the same consequential sequence as under option A. The work that needs to be done before the start of the "dismantling" phase, as well as the work performed during the dismantling, will also be similar to option A.

The dismantling of the RC No. 1 and RC No. 2 will start with the cutting of structures and equipment into small fragments (without fragmentation of the reactor vessels) in order to minimise the final volume of waste sent for disposal.

Opposite to option A, option B provides that, after removing the protective tank from the caisson, the reactor vessel will be loaded as one structure into a special shipping/disposal container.

Option B solutions for technological systems, water supply and sewage systems, ventilation, power supply, etc. are similar to option A.

3.2.3 Comparison of dismantling options

Main Technical and Economic Indicators by Options

The main technical and economic indicators for comparison by options are presented in Table 3.2.1.

Table 3.2.1 Technical and economic indicators of options

Indicators	Option A	Option B
The duration of the "dismantling" stage, years	8	7
Expected dose during dismantling of RCs	identical	
Expected dose due to fragmentation of reactor vessels, man-µSv	197,402.5	-
Expected dose of dismantling of the MB and infrastructure for RW management	identical	
Committed dose from the dismantling of the remaining buildings on the site, external networks, roads and fences	identical	
Labor costs, person-month	995.97	919.76
Waste from the dismantling of RCs, including:		
- non-radioactive waste, kg	2,870,000	
- RW, kg	300,000	
- hazardous non-radioactive waste, kg	1992	
The cost of dismantling RCs, thousand euros	25,544.0	24,138.8
The cost of containers for the removal of conditioned RW and reactor vessels, thousand euros	1,355.0	3,457.5

RW management for options A and B is identical, except for the need to fragment the reactor pressure vessels, and dose exposure to personnel for the options will differ only in terms of taking into account the work on fragmentation of the reactor pressure vessels.

When fragmenting reactor vessels, an increase in the total collective dose of personnel received during the dismantling of equipment and building structures of the RC is expected to increase by approximately 35%.

Analysis methodology

A formalised decision-making technique known as MAUT was used to compare both considered options. This methodology allows comparison of proposed decommissioning options both in terms of quantitative and qualitative parameters available at the time of the analysis.

In this document the methodology used to compare the proposed options is recommended by IAEA Nuclear Energy Series No. NP-T-1.1.10. This methodology is normally used to identify the strategy of and develop programs for the development of the nuclear energy industry, including, the cases where it is required to justify behaviour choices (in this case, decommissioning) of nuclear fuel cycle facilities.

In order to select the optimal option, groups of criteria (indicators) were identified at the MAUT analysis that characterises the “key elements” of the comparison between the options. In accordance with the selected methodology, the comparison is across the following groups of criteria that can characterise the actual decommissioning of a nuclear facility at the FPNC:

- Site considerations - K1;
- Safety - K2;
- Technical and other characteristics - K3;
- Radiation protection - K4;
- Environmental impact - K5;
- Site security - K6;
- Owner’s scope of supply - K7;
- Supplier issues - K8;
- Project schedule capability - K9;
- Economics - K10.

For each option the following is considered:

- The weight indicator of the fulfilment applied to the total value for each group of criteria (indicators), i.e. for the “key elements”;
- The total weight indicator that is a sum of criteria applied across the entire set of indicators;
- Priority is given to the option with the highest summation score.

Each indicator is then compared to others in the table, with an explanation of the position of the experts. According to the results of the MAUT analysis, the best options for the decommissioning plan for the nuclear facility is recommended.

Summarized results of the decommissioning options analysis

According to the results of MAUT analysis, the option without fragmentation of the reactor vessel scored the maximum number of points. The summarised results of the analysis for individual groups of the main criteria are given in Table 3.2.2.

Table 3.2.2 Results of the comparative analysis, according to the main criteria

Criteria number	Criteria	Option A Fragmentation	Option B The reactor vessel is not fragmented
K1	Site specific considerations	0.2025	0.2175
K2	Safety	1.0375	1.0725
K3	Technical and other characteristics	0.6405	0.6405
K4	Radiation protection	0.465	0.475
K5	Environmental impact	0.5	0.5
K6	Site security	0.1	0.1
K7	Owner’s scope of supply	0.1	0.1
K8	Supplier issues	0.094	0.094
K9	Project schedule capability	0.288	0.279
K10	Economics	0.742	0.886
TOTAL		**4.17**	**4.36**

Analysis of the individual groups of criteria shows that:

- **Option A “Dismantling and fragmentation of reactor vessels”** is the best option in terms of Project schedule capability. However, this option has average values or inferior to option B for all other comparison criteria.
- **Option B “Dismantling without fragmentation of reactor vessels”** is preferable to other options if the groups of criteria such as “Site features”, “Safety”, “Radiation protection” and “Economics” are considered; this advantage can be attributed to a relatively smaller number of traffic flows and a lower dose load on personnel and labour costs than in option A. Also, this option scored relatively low on the Project schedule capability criteria, since the implementation of this option depends on development and licensing of large-size waste containers. This option received the highest score.

Thus, according to the results of the integrated assessment of all groups of criteria for all the options, the option B “Dismantling without fragmentation of reactor vessels” is the preferred option.

4. PRELIMINARY DECOMMISSIONING PLANNING

4.1 Decommissioning Tasks, their Sequence and Interaction

The decommissioning process for the reactor facilities is carried out in several stages as follows: final shutdown; conservation; interim storage period; dismantling.

To date, the works performed at FPNC relate to the “final shutdown” and “preservation” stages. The RCs are being long-term stored (storage period). During the storage period, the residual activity of the most radioactive equipment will be significantly reduced, which will enable complete dismantling of the RCs without any risk of significant exposure of the personnel involved in the work, and without the need for expensive robotics.

By the beginning of work at the “dismantling” stage, certain activities shall be completed preparing the infrastructure necessary for handling and disposal of generated RW. Provisions shall be made for construction of repositories for disposal of containers with conditioned RW generated during dismantling of RCs, with facilities for decontamination, conditioning and packaging of RW being arranged.

At the “dismantling” stage, the work is to be performed to free the site from existing buildings and structures, followed by making the territory compliant with the requirements for the “green field”. During dismantling, the following is performed:

- Dismantling of contaminated and non-contaminated equipment and structures, including dismantling of the RCs of stand No. 346A (RC No. 1) and stand No. 346B (RC No. 2), as well as auxiliary equipment located in the MB;
- Management of radioactive waste with its subsequent transfer for disposal;
- Management of non-radioactive waste with subsequent disposal;
- Demolition of the existing MB;
- Reclamation of the territory (within the MB).

The main tasks at the “dismantling” stage are summarised in Fig. 4.1.1.

Dismantling of reactor compartment of stand No. 346B	Dismantling of reactor compartment of stand No. 346A	Demolition of buildings and structures on-site	Reclamation of the territory
Dismantling of sarcophagus No 2 (Stages 1, 9)	Dismantling of sarcophagus No 1 (Stages 1, 9)	Dismantling of equipment and utilities	Cleanup of the site
Dismantling of RC structures (Stages 2-5, 7-9)	Dismantling of RC structures (Stages 2, 3, 4, 8, 9)		Replacement of contaminated soil with fertile soil layer
Dismantling of RC equipment (Stages 6)	Dismantling of RC equipment (Stages 5-7)	Dismantling of structures, buildings and structural elements	
Management of radioactive waste (All stages)	Management of radioactive waste (All stages)	Management of radioactive waste	

Fig. 4.1.1 Main tasks at the dismantling stage

Based on previously completed work on preliminary studies for the decommissioning of the RCs of the FPNC and for the establishment of a RW repository (reports of Activity 3, sub-activity 4.8 and sub-activity 5.3) a preliminary schedule for decommissioning work is provided. The implementation schedule for the decommissioning of the RCs of the FPNC is shown in Fig. 4.1.2.

Name of works	Duration of work, year	Work period								
		2016-2039	2040	2041	2042	2043	2044	2045	2046	2047
Implementation of the FPNS reactor compartments decommissioning										
Preparatory stage	23,0									
Works execution stage on the site (territory of the FPNS)	8,0									
Preparatory work	2,0									
Main works	6,0									
Dismantling of the sarcophagus and reactor compartment - 346B	1,6									
Handling of dismantled fragments and RC equipment - 346B	2,0									
Dismantling of the sarcophagus and reactor compartment - 346A	1,8									
Handling of dismantled fragments and RC equipment - 346A	2									
Dismantling of the existing main technological building	1									
Dismantling of other buildings and structures on the site, as well as external utilities, roads and fences, reclamation of the territory	0,5									
Stage of site withdrawal from regulatory control and transfer to "green field"	0,5									

Fig. 4.1.2 RCs decommissioning implementation schedule

4.2 Types of Work and Methods Applied during Decommissioning

Implementation of the decommissioning project assumes dismantling of the RCs with the main process equipment. A detailed plan for decommissioning the RCs was developed within sub-activity 4.8.

The dismantling concept provides for complete disassembly of the RCs and cutting of the obtained components into relatively small fragments (with the exception of reactor vessels). This concept is aimed at minimising the amount of radioactive waste requiring final disposal through the use of methods such as fragmentation, decontamination, compaction, etc.

4.2.1 Dismantling of RCs

Work will begin with the dismantling of RC No. 2, which has lower radioactive contamination. After the dismantling and disposal of RC No. 2 equipment and structures, the RC No. 1 will be dismantled.

It is supposed that work on dismantling the RC No. 2 will be carried out from top to bottom with gradual access to the equipment installed in the iron-water protection tank. The work will be performed by the stages in the following sequence:

Stage 1

Dismantling of the reinforced concrete ceiling and partially external cast-in-place concrete walls of sarcophagus No. 2 to ensure free access to the cylindrical enclosure of the RC. Monolithic floor and wall sections will be cut into free of radiation contamination fragments.

Stage 2

Dismantling of the upper part of the cylindrical enclosure up to the 3rd floor decking of the RC. The metal structures of the enclosure will be cut into free of radiation contamination fragments.

Stage 3

Dismantling of the side parts of the cylindrical enclosure up to the 2nd floor decking of the RC to provide access to the pumpwells. The metal structures of the enclosure will be cut into free of radiation contamination fragments.

After that, the concrete laid during conservation on the upper room decking of the instrumental enclosure will be removed to provide access to the standard biological protection above the reactor vessel. It includes the dismantlement of the 3rd floor decking, the head of the reactor, and the protruding parts of the steam generators with the primary circuit pumps and the cooling pump.

Fragments of the top concrete layer generated during cutting may contain radioactive waste (tools, loading equipment, electrical equipment, etc.) that may have ended up in concrete during the conservation of RW.

Stage 4

Dismantling of concrete with a volume of about 31 m³ laid on the flooring of the upper room of the instrumental enclosure will provide access to the standard shielding, thus, releasing the reactor head, protruding parts of the steam generators with primary circuit pumps and the cooling pump.

Stage 5

Dismantling of the 3rd floor decking, standard shielding and concrete laid on the cover of the iron-water protection tank during the RCs conservation. In the course of this work, all auxiliary equipment and pipelines (cooling pump, primary circuit valves, feed water, etc.), which are located between the standard shielding and the cover of the iron-water tank will be removed so as to provide access to the equipment and pipelines, which are installed in the iron-water protection tank caissons. Concrete on the cover of the iron-water protection tank will be disassembled layer by layer, followed by the cleaning of the equipment, which is located in the iron-water protection tank caissons.

Stage 6

Dismantling of the main process equipment is carried out after its release from the concrete laid above and arrangement of access to the fastening elements. The work includes the dismantling of the: reactor vessel, steam generators with a pump, pressurisers, filter-cooler, and process pipelines. This equipment was exposed to neutron fluxes and is activated.

Dismantling of the main process equipment is carried out gradually by removing it from the protective tank caissons. After cutting the fasteners on the lid of the protective tank, the equipment is placed into special containers and transported to the radioactive waste treatment facility with the exception of the reactor vessel, which is to be disposed entirely. The cover of the protective tank will be cut along the outline and the steam generators will be cut into two parts.

The relocation of dismantled equipment within the MB is carried out using an overhead crane. The equipment is removed from the RC vertically (design position), then tilted into a horizontal position and loaded into special containers.

Stage 7

Dismantling of the side parts of the cylindrical body up to the 1st floor decking of the RC to provide access to the protective tank structures. In the course of the work, the following structures will be dismantled: concrete laid on the port and starboard pump room deck hatches; the 2nd floor metal decking; equipment and pipelines located in the hold; biological protection shields around the protective tank; metal structures of the cylindrical enclosure up to the 1st floor decking.

Standard shielding located around the tank was exposed to neutron fluxes during the operation of the RC and may be radioactively contaminated. The metal of the reactor enclosure is unlikely to be contaminated since it was located behind the shielding.

Stage 8

Dismantling of the protective tank metal structures by cutting them into fragments. The protective tank was exposed to neutron fluxes during the operation of the reactor and may be radioactively contaminated in some places. Before cutting the protective tank, the radioactivity of its walls will be measured.

Stage 9

Dismantling of the lower part of the RC cylindrical body, its metal supports, reinforced concrete shielding walls and sarcophagus walls. Metal structures and reinforced concrete structures will be cut into fragments. Some places of the metal structures of the cylindrical body lower part, its metal supports as well as the reinforced concrete walls of the shielding may be contaminated. The walls of the sarcophagus are unlikely to be contaminated.

The dismantling of the RC No. 1 will be carried out at the following stages:

Stage 1

Dismantling of the reinforced concrete ceiling and part of the sarcophagus external monolithic reinforced concrete walls to provide free access to the RC cylindrical body. Monolithic sections of floors and walls will be cut into free of radiation contamination fragments.

Stage 2

Dismantling of the upper part of the cylindrical enclosure up to the 3rd floor decking of the RC to provide access inside the RC. The metal structures of the hull will be cut into free of radiation contamination fragments.

Stage 3

Dismantling of concrete laid on the instrumental enclosure upper room decking during the RC conservation to provide access to the standard shielding above the reactor vessel. This is expected to result in having access to the 3rd floor decking and to the head of the reactor. Special equipment will be used to dismantle concrete layer by layer to get access to the containers filled with ionised radiation sources and radioactive waste, which may have been embedded in the concrete.

Stage 4

Dismantling of the pumpwells' structures within the 2nd and 1st floors will provide access to the end of the lower room of the instrumental enclosure (the steam generators' room on the port and starboard sides and the pressurisers' room in the aft part), which locates the main equipment. Pumping equipment will be cleaned of concrete.

It is expected that fragments generated during the dismantling work in the U-shaped enclosure may contain radioactive waste (rags, metal waste, tools, etc.) embedded in concrete during the conservation of the RC. Special equipment will be used to dismantle the concrete on the 2nd and 1st floors to locate the containers filled with ionised radiation sources, which may have been placed there during the conservation of the RC and filled with concrete.

During the dismantling, the remaining water will be pumped out of the RC systems into special liquid radioactive waste (LRW) containers. Pumping equipment and pipelines are contaminated whereas concrete fragments may remain uncontaminated.

Stage 5

Dismantling of the port and starboard steam generators enclosure rooms to provide access to the instrumental enclosure's lower room housing the reactor vessel. In the course of the work, the following will be dismantled: concrete laid on the hatch above the port side steam generator room during the pressure compensator conservation; the 3rd floor decking above the steam generator rooms, vertical metal partitions and concrete bioprotection blocks of the port side steam generator room; dismantling of steam generators (8 pcs.) and pipelines/ fittings from the port side steam generator enclosure room; dismantling of serpentine concrete blocks from the starboard side steam generator enclosure.

The shielding blocks are removed from the RC and being placed into containers are relocated outside the work areas by an overhead crane. Dismantling of concrete structures is carried out by cutting or breaking them into fragments. Concrete will be broken with jackhammers and cut with a circular saw mounted on a robot.

Dismantling of steam generators is carried out by their successive retrieval from the SG room of the RC. After cutting the fasteners, the equipment is placed into special containers and transported to the RW treatment facility.

Stage 6

Dismantling of 6 pressure compensators located in the bow section and 2 activity filters located in the aft section to provide the bow and stern access to the instrumental enclosure where the reactor vessel is installed. This equipment was exposed to neutron fluxes and is contaminated. In the course of the work, the following will be dismantled: lead screen and the 3rd floor decking above the bow pressure compensator room, the bow and stern vertical metal partitions; shielding between pressure compensators (graphite masonry); dismantling of "special concrete" between the filters.

Stage 7

The dismantling of the reactor vessel will be carried out after dismantling the process equipment, which is located along the perimeter, and providing access to the fastening elements. In the course of the work, the standard radiation shielding above the reactor and the vertical metal partitions of the equipment enclosure room (around the reactor) will be dismantled. The radiation shielding was exposed to neutron fluxes during the operation of the RC and may be contaminated.

Dismantling of the reactor vessel is carried out by removing it from the protective tank. After cutting the fasteners, the reactor vessel is relocated outside the work areas by using an overhead crane. The reactor vessel is removed from the RC vertically (design position), then tilted into a horizontal position and loaded into a special transfer and disposal container.

Stage 8

Dismantling of the protective tank metal structures by cutting them into fragments. The protective tank was exposed to neutron fluxes during the operation of the reactor and is contaminated.

Stage 9

Dismantling of the lower part of the RC cylindrical body, its metal supports, biological protection reinforced concrete walls and sarcophagus walls. Metal structures and reinforced concrete structures will be cut into fragments. The metal structures of the lower part of the cylindrical body, its metal supports as well as the biological protection reinforced concrete walls may be contaminated in some places. The sarcophagus walls most likely are not contaminated.

The work areas inside the RCs and sarcophagi could not be accessed for examination. In the course of the dismantling works, a permanent radiation survey will be carried out in the work areas to monitor changes in the radiation situation. Based on the available data, the decisions made in the design documentation for the safe performance of work are updated.

The reinforced concrete structures will be cut into fragments using a wall saw equipped with a diamond circular saw or a wire saw. The concrete will be broken with the help of jackhammers or mechanisms equipped with a hydraulic hammer. Metal structures will be cut into fragments using a circular saw mounted on a remote-controlled electro-hydraulic robot.

During the performance of the work, it is planned to install a mobile system for capturing and filtering smoke and gas generated during metal cutting as well as a vacuum system for collecting SRW generated during concrete breaking.

As the main process equipment is dismantled, the residual water will be pumped out of the RC systems. Water will be pumped to a special LRW container. The same LRW container will be used to collect water generated during the operation of mechanisms for cutting of the contaminated reinforced concrete structures.

To reduce the gamma radiation background during equipment dismantling, protective screens and other local radiation protection means against radiation will be used.

Dismantling and loading/ unloading work will be carried out using two existing overhead cranes with a lifting capacity of 50 tons (or using a new overhead crane with a lifting capacity of about 80 tons if these cranes cannot be restored to the design capacity during preparatory work).

4.2.2 Infrastructure for RW management

SRW management systems

RW preliminary management during dismantling will take place in the MB. After that RW final management will be done in the new RW treatment center in the separate building, which will be constructed before the start of the decommissioning. The SRW management in the MB will ensure the collection of the following SRW:

- Metal waste;
- Construction and heat-insulating materials;
- Filter elements for gas purification filters and ventilation system filters;
- Overalls, footwear, personal protective equipment are not subject to decontamination.

Pre-treatment of RW will be carried out immediately as RW is generated and include collection, preliminary characterisation, sorting followed if necessary by fragmentation and decontamination.

Collection of RW in situ will be carried out separately from non-radioactive waste. Mixing of RW with non-radioactive waste in order to reduce the specific activity of RW is not allowed.

Decontamination of reinforced concrete structures

Should the surface contamination of reinforced concrete structures be detected, it can be decontaminated directly on site before dismantling or at RW treatment center. Concrete structures will be decontaminated by chipping off the surface with standard equipment. The device is designed for mechanical decontamination of building structures (painted surfaces, brick, concrete, cement) with simultaneous collection of contaminated material in metal containers (200-liter drums). It consists of mechanical surface treatment module and a vacuum unit with a dust filtration system. It can be used with various modules both for decontamination of plates with contamination layer up to 10 mm deep and for manual decontamination of concrete floors and walls where contamination layer is up to 25 mm deep. Another option is to use removable modules for cleaning flat metal, concrete, brick and other surfaces.

Dust and debris generated during decontamination will enter the vacuum cleaner and pass through the filtration system into the container, thus avoiding the dispersion of contamination into the environment. Once filled, the container can be sent to RW treatment center.

Sorting and fragmentation

Sorting and fragmentation operations will take place in a workplace with the following equipment: beam crane, sorting table, docking station for containers to be removed from the sorting area, equipment for manual disassembly of small aggregates, fragmentation equipment. SRW will be transported by electric trolleys or by a crane.

The waste management process at the workplace begins with the preliminary radiological characterisation of the waste. After that, operations will be performed to fragment large-sized metal waste.

The following methods and equipment will be used to fragment all possible metal structures:

- Band saw machine;
- Powerful hydraulic scrap shears;
- Plasma cutting tools;
- Mechanical saws;
- Manual hydraulic shears.

Dust may be generated during fragmentation work. To ensure acceptable working conditions, the workplace will be equipped with an air purification unit.

4.2.3 Final decommissioning plan

This initial decommissioning plan should be subsequently updated to reflect information on changes of equipment or processes, unplanned events, changes in support capabilities including waste management and radiological monitoring, update of radiological conditions, changes in legislative requirements, changes in financial assumptions and improvements in decommissioning technology, etc.

The decommissioning plan should be finalised approximately three to five years before the safe enclosure phase ends. This final plan will be detailed and be approved by the regulatory body before implementation of the final decommissioning strategy, i.e. decontamination and dismantling. This plan is the basis for the development of the detailed work instructions and procedures.

4.3 Environmental and Radiation Monitoring

The programmes for environmental and radiation monitoring were developed within sub-activities 4.9 and 5.3.

4.3.1 Monitoring programme for the decommissioning of the RCs

General requirements for radiation protection control of personnel

The exposure control program for employees and workplaces and its update have to be approved by a regulator. The exposure of workers and workplaces are used to analyse the radiation status of the MB, its compliance with investigation levels, to analyse the radiation impact on workers and the environment, and to plan measures to reduce the radiation doses of personnel as much as possible and not exceed the limited personal dose. The screening levels for category "A" workers must be determined based on the actual potential exposure in accordance with the operator's radiation protection program and in order to ensure the optimal means of reducing the worker dose loads provided for in the operator's ALARA program. Taking into account the modern practice of performing similar works, it is recommended that the value of the investigation level during the first year of the dismantling work should not exceed the values given in the Table 4.3.1 during individual personnel control every month.

Table 4.3.1 Investigation levels during the first year of the dismantling work

No.	Job title	Irradiation dose, mSv	
		Investigation level	Threshold dose
1	Work manager	0.5	0.75
2	Forklift driver	0.7	1.0
3	Truck driver	1.7	2.5
4	Dosimetrist	4.6	7.0
5	Worker dismantling RC	2.0	3.0
6	Crane operator	0.1	0.2

The following personnel will use special dosimeters to assess the dose to the lens of the eye: work manager, dosimetrist, worker dismantling RC. Workplace control test levels are set in order to determine such values of ionising radiation dose rate, radionuclide activity or contamination per unit of area or volume, when exceeded, it is necessary to carefully measure the data and related parameters and determine the reasons for exceeding the investigation level. Worker exposure controls are carried out to accurately determine and record exposure doses and to keep the operator’s personnel and posted workers’ doses as low as possible (ALARA principle). Internal and external exposure doses of employees are carried out by PDC laboratory staff. Control of workplaces according to radiation dose power and control of radionuclide activity is carried out by the operator’s dosimetrists. The operators’ radiation protection engineer analyses the results of employee and workplace exposure controls and develops corrective measures and measures to reduce personnel exposure levels. Information about doses of radiation exposure to the operator's employees, third-party organisations, inspectors, specialists, students and pupils is provided in accordance with the requirements for ensuring the confidentiality of the results of personal control of the exposure of employees. The results of workplace controls are recorded and made available for review by the operator's staff and posted personnel.

Specific requirements for radiation protection control

Measured values and measurement methods

During the exposure control of employees and workplaces, the following values are measured and the results evaluated:

- Worker's external and internal radiation dose,
- Dose rate,
- Radioactive contamination of surfaces,
- Radioactive air pollution.

Methods of radiation control:

- Control of external exposure of personnel performing work in the CA is carried out using such dosimeters as TLD of the RADOS system (main dosimeter), TLD-500K dosimeters of the KDT-02M set (emergency control), and personal electronic dosimeters (operational control).
- Personal control of internal exposure of personnel performing work with radioactive materials and sources of ionising radiation in the CA is carried out by the HRC gamma spectrometric measurement system such as 2250 FASTSCAN™ High-Throughput Whole Body Counter or 2280 ACCUSCAN II. The purpose of the control is to obtain information about the amount of radionuclides that have entered the body and individual human organs, to determine the dose of internal radiation, and to timely identify cases of increased levels of radionuclides in the body and prevent exceeding the prescribed limit dose.
- Periodic monitoring of the equivalent dose of gamma and neutron (if necessary) radiation is carried out by portable devices equipped with an extension rod and an external detector.
- Continuous control of the equivalent dose of gamma and neutron (if necessary) radiation in the CA is carried out by stationary RCs equipment.
- Control of removable (unfixed) contamination of the surface with alpha and beta radionuclides is carried out by the smear method, using a smear activity monitor (laboratory monitor).
- Control of non-removable (fixed) contamination of the surface with alpha and beta radionuclides is carried out by direct measurement with portable devices for measuring the flux density of alpha, beta radiation.

- Continuous monitoring of the volumetric activity of aerosols released into the atmosphere during operation is carried out by the stationary RCS alpha-beta aerosol volumetric activity and radon monitor.
- The mobile RCs alpha-beta aerosol volumetric activity and radon monitor carry out periodic monitoring of the volumetric activity of aerosols in the premises and in the CA.
- Continuous monitoring of aerosol volumetric activity in exhaust ventilation air is carried out by a stationary RCs alpha-beta aerosol volumetric activity and radon monitor.
- Control of surface contamination of personnel's work clothes and tools with alpha and beta radionuclides when leaving the RC and CA is performed by stationary surface contamination control monitors.
- Continuous monitoring of the equivalent dose of gamma radiation is carried out by TLD along the perimeter of the Paldiski site territory.
- Radiological control of vehicles when leaving the Paldiski site territory is performed by a stationary portal monitor.
- Personnel radiological control on the way out from the MB territory performs stationary portal monitor.

Exposure control equipment for employees and workplaces

Specific types of exposure control equipment for workers will be determined in the work project. The equipment used to control the exposure of workers and workplaces is tested, checked and calibrated in accordance with the procedures established by IAEA recommendations (Calibration of radiation protection monitoring instruments, IAEA, Safety reports series no. 16) and Estonian legislation.

Exposure levels and measures to be taken when exposure levels are exceeded

The dose limit for the category "A" personnel:

- Annual effective dose - 20 mSv,
- Equivalent dose rate of 20 mSv or 100 mSv in the eye lens during any five consecutive years, provided that the dose in one year does not exceed 50 mSv,
- Average equivalent dose rate of 500 mSv per 1 cm^2 of the skin surface, without taking into account the actual surface area of the irradiated skin,
- Equivalent dose rate of 500 mSv in the limbs.

Category "A" personnel performing work in the CA can receive an effective dose of no more than 0.2 mSv per day.

The following restrictions apply to the category "A" personnel:

- Personnel whose difference between accumulated and limited annual dose is less than 3.0 mSv is allowed to work receiving 0.05 mSv/day,

- Personnel whose difference between accumulated and limited annual dose is less than 1.0 mSv is allowed to work receiving 0.01 mSv/day,
- When granting permission to work with less than 0.05 mSv/day, the employee's personal dose is continuously monitored with an electronic dosimeter (in addition to the TLD dosimeter) at the CA until the end of the reporting year, regardless of the nature of the CA work performed,
- If it is necessary to perform work with a planned dose higher than 0.2 mSv/day, a special permit is issued for a one-time planned increased exposure of personnel.

Planned increased daily exposure of workers may be allowed only when it is not possible to apply measures that prevent exceeding the established dose limits. Only certain persons authorised to do so can prescribe higher doses for workers. For employees it is forbidden to exceed day's personal doses limit without special permission. This is considered as a violation of radiation protection rules and norms. In case of exceeding the daily dose limits, the causes of exceeding the dose level are determined and preventive measures are prepared to avoid exceeding the dose level in the future, and radiation safety is improved. In case of exceeding the annual limit dose or in cases where the annual limit dose may be exceeded due to unforeseen circumstances, the holder of the license or permit must immediately, but not later than within 1 day, inform the regulator about it.

Registration of workplace and personnel exposure control results

The registration of workplace control results is carried out in accordance with the operator's valid procedures.

The results of the personnel's individual external and internal exposure control are recorded in the APDCS database. The APDCS database collect, stores, processes, manages and stores data on personnel exposure. The record of issuance of personal electronic dosimeters and registration of dose values, when performing work in an ionising environment, is carried out in the logbook of personal electronic dosimeter doses and in the APDCS database in manual or automatic mode. A scanning device is used when leaving the work performance area and reading the dose from the electronic dosimeter. When an employee is released from PDC, the radiation exposure dose report is stored in the PDC laboratory and transferred to the operator's archive for safekeeping at the end of the calendar year. The term for keeping reports in the operator's archive is until the employee reaches the age of 75 (or would reach the age of 75), as well as at least 30 years after the completion of work related to occupational exposure.

Organisation of exposure control of the operator's employees

All employees of the operator and third-party organisations visiting CA, performing permanent, temporary or one-time work, must have a personal TLD dosimeter and must undergo PDC. During the decommissioning of the MB, for the control of doses of radiation exposure to personnel when performing work in an ionising environment according to instructions, and in the conditions of a change in the radiation situation at the workplace during a shift, with the main TLD dosimeter, a personal electronic dosimeter is additionally used, which is equipped with an indicator of the accumulated dose, sound and light alarm of exceeding the set threshold of external exposure. Employees are prohibited from being in CA without a personal dosimeter. This is considered a violation of the rules and requirements of the radiation safety and sanitary control regime applied by the operator. A separate room must be provided where personal TLDs are issued to the operator's employees. The room must be designed for the storage of TLD during use (during work breaks). After work in the MB, dosimeters are returned to the room where they are issued. Periodicity of measurements - according to the workplace control program. Electronic portable dosimeters are issued and returned in the same room where TLDs are issued. The dosimeter for determining the effective dose of external radiation must be worn over the personal protective equipment in the chest area. The dosimeter for external ocular exposure should be worn at eye level as close as possible to the most affected eye.

Control of radiation of workplaces

The control measures of workplaces during the decommissioning of the RCs have to be performed for the following objects: Cementing plant (No. 1 on Fig. 4.3.1); Compaction plant (No. 2 on Fig. 4.3.1); Measurement facility for RW characterisation (No. 3 on Fig. 4.3.1); Final RW packages /Bulk RW (No. 4 on Fig. 4.3.1); Metal RW sorting workplace (No. 5 on Fig. 4.3.1); Decontamination workplace (No. 6 on Fig. 4.1.1);.High-pressure washer (No. 7 on Fig. 4.3.1); Measurement facility for RW clearance (No. 8 on Fig. 1); Empty drums and containers storage place (No. 10 on Fig. 4.3.1).

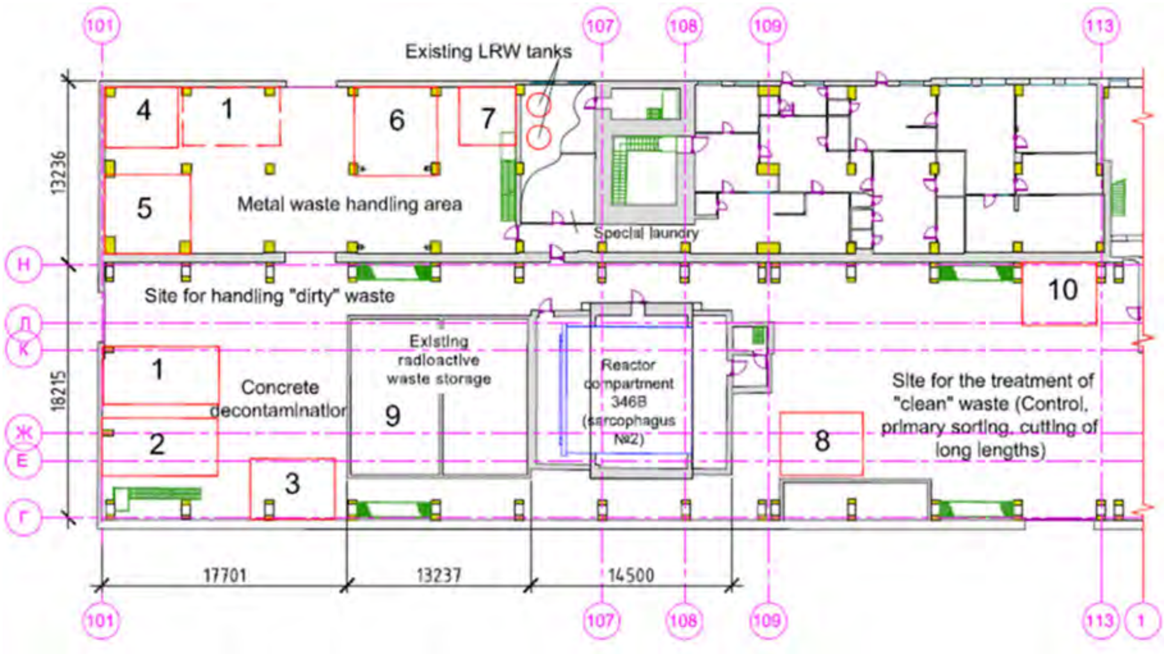


Fig. 4.3.1 Scheme of the MB part 302A

Following values have to be controlled as necessary according to the controlled object: Gamma and neutron (if necessary) radiation dose rate; Surface activity by smear method; Contamination of surfaces with alpha-beta radionuclides; Volumetric activity of alpha, beta aerosols and radon; Effective dose of external/internal radiation; Gamma radiation dose; Total atmospheric precipitation activity; Total activity of groundwater.

Frequency of measurements: Automatic continuous measurements with record keeping in the database; According to the specific operations performed, e.g. each time the container is placed for storage, changing the location of the container, but at least once a month, or after the decontamination of the transport container.

Measurement equipment or methods: Radiation control system; Mobile devices for measuring gamma, neutron (if necessary) radiation dose rate, alpha, beta radiation flux density; Mobile alpha-beta aerosol and radon volumetric activity monitors; Portal transport pollution monitor; Handfoot-fibre monitor; Full-body contamination monitor; Radiation portal monitor; RADOS TLD-system; Electronic dosimeter; Human radiation counter; Smears; Sampling from cuvettes or from stationary wells and laboratory measurements.

Records

The results of the MB workplace control and mapping are recorded, according to the established procedure. The results of personal control of external and internal exposure of personnel are recorded in the APDCS database. The APDCS database collects, stores, processes, manages and stores data on personnel exposure during the operator’s operation. Records are stored in electronic format, security is ensured by the service requirements of the automated PDC system ASRM2A server. The retention period of records is limited to the lifetime of APDCS in the operator’s project. All entries must be clear. Documents with control results are registered according to the procedure established by the company and stored at the radiation protection department engineer's workplace.

4.3.2 Environmental monitoring programme for the decommissioning of the RCs

Gamma monitoring

Continuous measurements of ionising radiation intensity have to be performed in representative locations of FPNC and in directions to the nearest residential areas. The gamma monitoring system performs a function of early warning in case of emergencies. Automatic electronic devices have to be used for measuring the gamma-ray dose rate in the environment. The data are transmitted to the central station. Typical characteristics of the gamma-monitoring system are presented in Table 4.3.2.

Table 4.3.2 Proposed characteristics of the gamma-monitoring system

	Characteristics
Type of probe	Gamma probe, calibrated in ambient dose equivalent units
Measuring range	10 nSv/h to 15 mSv/h
Energy range	40 keV to 2.5 MeV
Measuring uncertainty	Less than 15%
Data	Real time data

Recording	Data storage memory

MB of FPNC should be surrounded by a perimeter consisting of four gamma-monitoring stations inside the metallic fence. It is also proposed to install an additional station at the entrance into the territory of FPNC and two additional stations in directions to the nearest residential areas. The exact positions of the gamma stations have to be defined in the detailed design of the decommissioning of the RCs. The gamma monitoring system should be installed and measurements should start before the decommissioning of the RCs (preferably about 1 year earlier).

Air monitoring

The gamma monitoring network also has to include monitoring of radioactive particles and aerosols in the air. The release of radioactive substances into the atmosphere is below permissible levels of gaseous and airborne radioactive emissions into the environment during normal operation. However, high concentrations of radioactive contaminants can be detected in the air in case of accidents. Therefore, an automatic air monitoring station located near the MB has to be installed. The location for this station has to be selected taking into account the prevailing wind direction. It has been found (within Sub-activity 2.11) that the region of Pakri Peninsula is dominated by south-southwest winds. Therefore, the northeast is the optimal direction for the air monitoring station. The exact location has to be defined in the detailed design of the decommissioning of the RCs. Aerosol filters coupled to air pumps are capable of accumulating particles from large volumes of air onto a small surface, their radioactive content can be determined with good measuring efficiency thus allowing advantageously low detection levels. The presence of non-natural radioactivity on the air filter is detected by means of alpha-, beta- and gamma-counting. The required minimal sensitivity for measuring of radioactive aerosols of artificial nuclides is presented in Table 4.3.3.

Table 4.3.3 Sensitivity of the radioactive aerosols monitoring station

Measured parameter	**Detection limit, Bq/m^3**	
	Duration of filtering 1 h	Duration of filtering 24 h
Concentration of Cs-137	0.7	0.07
Concentration of alpha particles	0.5	0.05
Concentration of beta particles	0.7	0.07

The air monitoring station has to be installed and the measurements have to start before starting the decommissioning of the RCs. The station has to be combined with an automatic meteorological station. The following meteorological parameters should be measured hourly:

- Air temperature,
- Wind speed,

- Wind direction,
- Precipitation.

Good knowledge of meteorological conditions is relevant for predicting the situation in case of emergencies. Two meteorological stations operated in the national monitoring network are in the region. Paldiski Coastal Station is the nearest one to the MB of FPNC. Distances from FPNC site to this station is about 2 km. Data from national meteorological station can be used for radionuclides transfer assessment.

Although meteorological station is close to FPNC, installation of the independent station at FPNC can be considered as an advantage of having the weather station on the site to assure effective data transfer in case of accident.

Ground water monitoring

Ground water monitoring on FPNC is already ongoing in the framework of the National Monitoring Programme. Depths of these boreholes is up to about 10 m. Sampling location “Suubla” represents a drainage water flowing from FPNC.

The ground water samples have to be regularly taken for laboratory analysis until the change in the radiological status of FPNC. The specific activities of radionuclides H-3, Co-60, Sr-90, Cs-137 have to be measured once a month during the implementation of the decommissioning of the RCs. In case of an abnormal increase of the gamma-ray dose rate, gaseous and airborne radioactive emissions, the specific activities of radionuclides H-3, Co-60, Sr-90, Cs-137 in the ground water have to be measured. If the abnormal increase of radionuclides H-3, Co-60, Sr-90, Cs-137 in the ground water is found, then specific activities of radionuclides C-14, Ni-59, Ni-63, Nb-94, Eu-152, Eu-154, Ra-226, Pu-238, Pu-239, Am-241 also have to be measured.

Monitoring of terrestrial ecosystem

During eventual accidents, radioactive fallouts can contaminate the surrounding soil and grass. Therefore, the accumulation of the radioactive substances has to be controlled by taking the samples and analysing them in the laboratory. In order to determine the background contamination, the monitoring of soil and grass shall be started at least one year before the start of the decommissioning of RCs. The sampling plot has to be selected on FNTC during the development of the detailed design of the decommissioning of RCs, taking into account the prevailing direction of the wind (i.e. in the direction of the highest probability of radioactive fallouts). One grass and one soil sample have to be taken from the same plot once a year. The grass has to be sampled at the end of summer. Grass samples have to be taken from 1 square metre. The upper 5 cm soil layer has to be sampled, the amount of sample should be about 1.5 kg (taking into account gamma spectrometer calibration geometry). Measurements of gamma activity in the grass and soil samples have to be done according to International Standard IEC 61452:2021 using a high-resolution gamma spectrometer, capable of identifying the relevant radionuclides (Co-60, Cs-137, Ra-226, Am-241). The spectrometer energy resolution (full width at half maximum) should be at least 2 keV, while the energy range should be 40 keV to about 2 MeV.

Alpha and beta radionuclides get importance in case of operational accidents. In these cases, these DTM radionuclides must be investigated.

Possible reference site

To determine whether a change in radioactive contamination on the FPNC was caused by the decommissioning work done on the FPNC or by an outside source, it should be considered whether a reference site for comparison should be created. This reference site should have similar geomorphological characteristics and biota as the FPNC. Radionuclide composition would be determined from grass and soil samples. Analysis of the samples is to be made using a gamma spectrometer. The frequency of sampling should be at least once per year, at the end of summer. Depending on the circumstances and necessity, the sampling could be done more often.

Monitoring of marine ecosystem

Radioactivity in the marine environment is already monitored in the framework of the National Monitoring Programme. There are two monitoring stations in the Gulf of Finland next to FPNC. Regular sampling of water, bottom sediments and biota is done each year for measurements of artificial radionuclides Sr-90 and Cs-137. These two stations of the national monitoring network are sufficient.

The marine monitoring has to start before the decommissioning of RCs and continued until the change in radiological status of FPNC.

5. STUDY OF DECOMMISSIONING SAFETY

The developed safety case provides comprehensive safety analysis of the impact of decommissioning activity on the personnel implementing the dismantling of the MB, public and environment around FPNC, neighbouring countries as well as risk analysis.

The main topics discussed in the study of decommissioning safety are:

- Identification of relevant safety criteria;
- Operational limits and conditions;
- Hazard analysis of normal decommissioning activities;
- Hazard analysis of abnormal events and incidents;
- Assessment of potential consequences;
- Results of the assessment of the effects of non-radiation factors;
- Possible impact of the decommissioning of the RCs on neighbouring countries;
- Risk assessment;
- Comparison of analysis results with relevant safety criteria.

5.1 Safety Assessment

A preliminary safety assessment was performed within sub-activity 4.8 and was finalised within sub-activity 5.3.

5.1.1 Safety assessment content

Safety assessment for the decommissioning of the MB, its internal structures, systems and components including the RCs was performed in accordance with the IAEA recommendations stated in the safety guides WS-G-5.2, SRS-45 and SRS-77.

This safety assessment describes the MB that locates the reactor sarcophagi and analyses the following stages:

- Dismantling of reactors and reactor components,
- Dismantling the sarcophagi that are built over the reactors,
- Dismantling of the internal systems of the building,
- Dismantling the building itself.

The safety assessment of the planned activity (dismantling works) was carried out on the basis of technical descriptions of the existing state of structures, systems and components of the main technological building of the FPNC. Moreover, to make a decision on the techniques of dismantling and waste management, consideration was given to the equipment and techniques commonly applied for industrial purpose that were well proven in performing the similar activities for the other nuclear facilities.

This safety assessment does not include the RW disposal facilities located off-site, as well as the transfer of packaged and characterised RW placed for temporary storage in a storage facility inside the MB. The infrastructure for radioactive waste disposal is supposed to be built before the start-up of the reactors dismantling in 2039. The specified RW will be transferred for disposal in the appropriate repository.

The final state of the site is characterised as follows:

- Facility liquidation;
- Full dismantlement and decontamination of constructions;
- Partial removal of RW, disposal of the remaining RW.

Within the framework of the safety assessment:

- Requirements for safety assessment and acceptance criteria are defined;
- Provided description of FPNC including historical background and current status of facilities buildings and constructions;
- Decommissioning activities for the MB and existing internal structures are developed;
- An engineering assessment of the structures, systems and components to be used during decommissioning has been completed;
- A safety analysis of the presented decommissioning technology was carried out, as a result of which a list of initiating events and the frequency of their occurrence was determined;
- Scenarios for the implementation of these initiating events were developed and analysed, their qualitative screening was carried out;
- The analysis of radiation safety for the scenarios of normal operation and scenarios of the implementation of initiating events, which were not screened out at the previous stage, was carried out;
- Safety assessment of non-radiation factors was carried out;
- An assessment was made for the conformity of the results of the analysis with the regulatory requirements and corresponding acceptance criteria;
- Developed technical and organisational measures to control normal operation and measures to prevent and eliminate the MB accident scenarios.

To fulfil the specified requirements, the SAR contains a numerical assessment of the impact on workers and the environment due to exposure factors (radiological and non-radiological) under normal conditions (accident-free) of decommissioning activities and accident conditions.

5.1.2 Approach applied for safety assessment

In accordance with the IAEA Safety Guide (WS-G-5.2), when developing the safety assessment, the following were performed:

- **Justification of compliance with the relevant safety requirements and criteria of the regulatory body to support approval for the proposed decommissioning activities;**
- **Systematic assessment of the nature, extent and likelihood of hazards and their radiological consequences for workers, the public and the environment in connection with the planned activities and accident conditions;**
- **The planned and gradual reduction of radiological hazards to be achieved in the course of decommissioning activities, etc. is quantified.**

The SAR takes into account the adopted decommissioning strategy, as well as the current MB dismantling organisation plan and technical solutions for the decommissioning of RC. The safety assessment will help develop the final decommissioning project with the possibility of updating the proposed solutions at subsequent stages of project development.

For the analysis of accidents during decommissioning, the deterministic approach to safety assessment recommended by the IAEA experts was applied. This approach to the safety assessment is considered as the most effective method for provision of protection levels for workers and the public during decommissioning activities for nuclear facilities.

Deterministic analysis deploys a design analysis of initiating events related to abnormal operation and design basis accidents using the computer codes to assess safety and fulfil the acceptance criteria adopted for calculations. Obviously, at the stage of decommissioning of reactor facilities from which nuclear fuel is retrieved, the main criteria for the analysis to perform is the fulfilment of radiation criteria.

A validated and verified computer code GENII 2.10 was used for assessment of radiological criteria for workers. The computer code was developed by Pacific Northwest National Laboratory, Richland, Washington. The GENII system includes capabilities for calculating radiation doses following chronic and acute releases. Radionuclide transport via air, water, or biological activity may be considered. Air transport options include both puff and plume models, each allowing use of an effective stack height or calculation of plume rise from buoyant or momentum effects (or both).

The safety assessment was performed on the basis of inputs regarding the description of the facilities, the engineering analysis of the systems deployed and the dismantling of the facilities to be undertaken in accordance with the proposed decommissioning plan. This data to be used to identify existing and potential hazards inherent in the facility and new hazards that may arise due to the nature of the decommissioning work being carried out.

Relevant hazards were quantified and their possible consequences for workers and the public identified. The expected effective doses and the risks associated with these hazards were then compared with the relevant safety requirements and criteria to determine whether these safety requirements and criteria are met.

The extent of analysis of potential accidents during decommissioning activities depends on the probability of occurrence of initiating events and on the potential consequences. Therefore, the initiating events were detailed assessed to ensure the completeness of the analysis of facility responses to postulated disturbances of process parameters and equipment failures. The analysis of potential accidents took into account all decommissioning activity-specific internal and external initiating events, as well as human errors.

According to the specifics of the facility under consideration, the initial accident events are grouped as follows:

- Release of radionuclides during dismantling of structural elements and equipment;
- Release of radionuclides during management of various categories of RW on site;
- Release of radionuclides during transportation of RW on site;

Each initiating event, depending on the expected frequency of occurrence, was classified into one of the following categories:

- Abnormal operation - a category of expected initiating events that may occur at least once during the operation of the facility (i.e. the expected frequency of occurrence of which is not less than 2×10^{-2} 1/year);
- Design basis accident – a category of postulated initiating events that form the design basis for safety systems, the expected frequency of which is less than 2×10^{-2} 1/year.
- Frequent events at P1 frequency more than 10^{-2} 1/year;
- Probable events at the frequency of $10^{-6} \leq P2 < 10^{-2}$ 1/year;
- Incredible events $P3 < 10^{-6}$ 1/year.

The value of the frequency of initiating events was determined by the methods of probabilistic safety analysis in the presence of appropriate statistical data or by an expert method.

Acceptance criteria were established to perform the analysis of each initiating event. The selection of acceptance criteria took into account the group, category and expected consequences of the initiating event. It was chosen to consider that the initiating events (IE) with the lowest expected frequency of occurrence can have the most severe consequences.

To perform the design analyses, the IE-representatives were selected from the generalised list of initiating events. The IE-representatives assume such initiating events, whose consequences are the most adverse in comparison with the consequences of other initiating events from this group.

Performance of a design analysis of an IE- representative includes the following stages:

- Definition of a task;
- Performance of calculations and analysis of results;
- Documentation of analysis results.

The definition of a task assumes determination of initial and boundary conditions that determine the scenario (i.e. the expected sequence of events) of the transient process.

Initial conditions include numerical values of parameters that are directly measured (e.g. pressure, temperature, etc.) or can be calculated, as well as the expected operating conditions of systems and equipment at the time of occurrence of the initiating event analysed.

When analysing operational events and design basis accidents for facilities of this type, the limit values of parameters out of the range of values specified for normal operation (i.e., operational limits) are used as initial values:

[operating limit] = [nominal value] ± [deviation],

where [deviation] is the deviation from the parameter nominal value acceptable for normal operation, caused by the operation of control systems and/or the measurement error of the current value of the parameter.

When selecting initial conditions, the analysis considers the parameter values and characteristics (operating conditions) of systems and equipment that lead to the worst consequences of the initiating event being analysed and characterised by the smallest margin for non-excess of established acceptance criteria.

The boundary conditions include the expected configuration, characteristics and operating conditions of systems and equipment, as well as the actions of workers upon the occurrence of the initiating event in question.

In assessing the safety of decommissioning, attention was paid to the fact that a large number of uncertainties exist at the time of report development. For instance, changes in the regulatory framework or the characteristics and types of equipment used for demolition and handling the dismantled items may not be well defined. The operator of a facility may need to clarify the type of equipment based on available technologies at the start of decommissioning. The scenarios outlined in the decommissioning plan may need revising based on knowledge gained during the preparation stage for the decommissioning process.

Considering possible uncertainties during safety assessment, a conservative approach was applied taking into account the knowledge and data available during development of the SAR.

The choice of both boundary and initial conditions should be based on the application of the principle of conservatism. Accounting for all conservative assumptions enables the receipt of the most conservative result, which provides for consideration of the effect of all uncertainties on the analysis results.

5.1.3 Assessment of radiation factors for normal operation scenarios

In terms of impact on personnel

When dismantling the RC No. 1, in order to fulfil the condition of not exceeding the dose limit for the category A personnel, the following number of workers are required for individual stages:

- 2 workers are needed for Stage 4;
- 13 workers are needed for Stage 7.

When performing Stage 8 of the dismantling of the RC No. 2, a minimum of four workers are required to meet the condition of not exceeding the dose limit for the category A personnel.

The maximum dose rate for workers handling packaged RW will not exceed 2 mSv/h at the workplace.

In terms of impact on the public and the environment

Under normal work conditions radiation exposure of the population and the environment can occur only due to gas-aerosol emissions into the atmosphere since there are no sources of any significant water discharges during work. The total release activity under normal conditions of dismantling work will not exceed 1.1E+04 Bq. The comparison of the release levels leads to conclude that the magnitude of the radioactive release, under normal work conditions, will not exceed the release levels specified in the Regulation No. 40 “Conditions for Exclusion and Release of Radioactive Substances Used or Generated in Radiation Operations and Requirements for Requests for Exclusion and Release”.

5.1.4 Identification of hazards and initiating events

The hazard identification process identifies all locations in the facility where radioactive material is present (e.g. intentional and unintentional accumulation of radioactive material and radioactive waste, surface contamination, contaminated soil, radioactive sources, activated components and ventilation system filters). Particular attention was paid to radioactive materials which, as a result of planned decommissioning activities, create new sources of exposure to personnel, for example, as a result of a change in the ventilation system due to decompression of the containment during the dismantling of the facility or the dismantling of the shielding wall.

Consideration was given to future accumulations of material on the site, such as radioactive waste storage facilities, which will be progressively replenished and must be assessed based on the maximum level of radioactivity expected to exist at any time.

This process took into account all potential initiating events that could have an adverse impact:

- External initiating events;
- Natural phenomena such as adverse meteorological conditions (e.g. wind, snow, rain, ice, temperature, flood, lightning), earthquakes or biological interference;
- Man-made events such as aircraft crashes (with or without subsequent fires), explosions, fires, loss of power or other functions, and human intervention;
- Internal initiating events at the facility or site, such as fire, explosion, structural collapse, leakage or spillage, failure of the ventilation system, heavy loads falling and failure of protective measures (e.g. failure of shielding or personal protective equipment);
- Human-caused initiating events, such as human errors and violations, and misidentification of events leading to inappropriate actions.

5.1.5 Hazard Screening

Hazards that may arise during decommissioning will be quantified without taking into account any protective or mitigating safety measures that will be implemented at the facility during decommissioning. However, consideration will be given to the benefit that can be gained from the inherent (passive) safety features of the facility (e.g. shielding walls, engineered safety features) as long as they remain operational during decommissioning. Hazards that have the potential to cause significant adverse effects through any identified pathways, or high-risk hazards identified by comparison with appropriate criteria, require further consideration.

Hazards that are not within the scope and/or do not meet the safety assessment objectives or cannot lead to consequences that exceed the threshold of the necessary requirements, will be screened out. This will lead to a reduction in the list of hazards on which the safety assessment will be focused. In facilities of low hazard or complexity, or in cases where planned decommissioning activities are limited, there may be only a few actual hazards, thereby reducing the scope of the safety assessment.

The screening process will take into account all potential exposure pathways through which certain hazards can cause harmful effects to workers, such as:

- External exposure due to contamination, activation of structures (components, buildings, surfaces, etc.) or exposure to other radioactive material (e.g. sealed sources, waste packages) through direct exposure to gamma-emitting radionuclides;
- Internal exposure by inhalation or ingestion from airborne releases (e.g. gases, aerosols and particulates) during material cutting (e.g. thermal and mechanical cutting) or decontamination or fire; as a result of aerosols from chemical decontamination baths or the use of mechanical methods of decontamination, as well as a result of exposure to other sources;
- Combination of radiation contamination and bodily injury (e.g. contamination of wounds).

Based on the review of initiating events presented in subsection 6.1 of Sub-activity 4.8 and on the data of sections 4 and 5 of Sub-activity 4.10, a list of initiating events was compiled, for which scenarios are defined in subsection 6.3 of Sub-activity 4.8. In addition to the initiating events presented in subsection 6.1, three more initiating events were defined:

- Lightning strike.
- Opening of the source of ionising radiation during dismantling works inside RC No. 1.
- Military, sabotage and reconnaissance group or terrorist attack.

The frequency of each initiating event is determined in accordance with Sub-activity 4.8 and Table 1 in IAEA Safety Reports Series No. 23. The results are presented in Table 5.1.1.

Scenarios for the implementation of these initiating events were developed and analysed, their qualitative screening was carried out.

Table 5.1.1 List of initiating events

The name of initiating event	Category (see subsection 5.1.2)
External initiating events	
Earthquake	P1
Hurricanes, storms, wind	P1
Floods	P1
External fires	P2
Snow load	P1
Extreme temperatures	P1
Lightning strike	P1
Tsunami	-
Explosions on site or nearby	P2
The fall of the aircraft/flying object	P3
External power loss	P1
Internal initiating events	
Falling loads/collapse of upper slabs	P2
Fire in the MB/fire of a leaky container with SRW	P2
LRW spill	P1
Opening of the source of ionising radiation during dismantling works inside RC No. 1	P1
Loss of power supply	P1
Human error	P1
Military, sabotage and reconnaissance group or terrorist attack	P1

5.1.6 Assessment of the impact of radiation factors for accident scenarios

Impact on personnel

The accident "Opening of the source of ionising radiation during dismantling work inside RC No. 1" was assumed as a design basis accident with the maximum radiation consequences for the personnel. If the source is destroyed, the dose rate at a distance of 1 m will be 1.1 mSv/h. It is assumed that in the event of an emergency, the personnel will leave the emergency area within 5 minutes and move to a safe place. In this case, a single individual exposure dose will not exceed 0.1 mSv.

The accident "Falling loads/collapse of the upper slabs" with a simultaneous failure of the personnel respiratory PPE was accepted as a beyond design basis accident with the maximum radiation consequences for the personnel. The 50-year lifetime dose of internal exposure for personnel, taking into account 5-minute external exposure from reactor No. 1 will be 60.1 μSv.

Impact on the public and the environment

The maximum radiation exposure of the population and the environment in case of accidents can occur only due to accidental gas-aerosol emissions into the atmosphere, since there are no sources of any significant water discharges during the work. Two scenarios were analysed: a case with precipitation and a case without precipitation.

The accident "Fire in the MB/fire of a leaky container with SRW" was assumed as a design basis accident with the maximum radiation consequences for the public. The total effective dose for the case with precipitations will not exceed 2.91E-08 Sv. The total effective dose under the scenario without precipitation will not exceed 2.92E-10 Sv.

The accident “The fall of the aircraft/flying object” with a subsequent fire and parallel destruction of an industrial ionising radiation source was taken as a beyond design basis accident with maximum radiation consequences for the population and the environment. The total effective dose for the case with precipitations will not exceed 1.644E-03 Sv. The total effective dose under the scenario without precipitation will not exceed 6.30E-06 Sv.

5.1.7 Comparison with the radiation safety criteria

Under normal operating conditions due to applying a special work technology, maintaining safety barriers and controlling for compliance by a radiation monitoring system, the protection of the public and the environment will be ensured by limiting radioactive releases and discharges to a level not exceeding the release levels specified in the Regulation No. 40 “Conditions for Exclusion and Release of Radioactive Substances Used or Generated in Radiation Operations and Requirements for Requests for Exclusion and Release”. The analysis of normal decommissioning activities and abnormal events and incidents shows that the radiation safety criteria are not exceeded:

- **THE LIMITS OF DOSES FOR PERSONNEL AND THE PUBLIC AS PER RADIATION ACT AND ESTONIAN GOVERNMENT DECREE NO. 97 ARE NOT EXCEEDED, BOTH UNDER NORMAL CONDITIONS AND IN ACCIDENTS;**
- **THERE IS NO NEED TO APPLY ACTION LEVELS: EVACUATION, RESETTLEMENT, AS SPECIFIED IN THE ESTONIAN GOVERNMENT DECREE NO. 95.**

Assessment of the impact of radioactive factors on the public and the environment under scenarios of emergencies taking into account a conservative approach demonstrated that the exposure dose of the public as a result of radioactive contamination of the territory will not exceed 1.0 mSv/year. Therefore, the survey of radiological conditions including all types of radiation and optimisation of measures aimed at radiation protection at the territories considered is not required. Since specific radiation can be prevented via protective measures and dismantling technologies planned, there is no need to carry out the protective measures associated with the interference of the normal public life as well as the economic and social activities at this territory.

5.2 Impact of the Decommissioning on Neighbouring Countries

Possible impact of the decommissioning of RCs on neighbouring countries was investigated within sub-activity 4.11.

5.2.1 Methodology

The assessment of the impact of transboundary atmospheric and marine transport of a radioactive release in the event of a radiation accident during the decommissioning of the RCs in the FPNC is carried out for Finland as the state closest to the potential source of the release. The city of Helsinki with a population of more than 650 thousand people, located at a distance of about 110 km from the source, was chosen as the object for conducting assessments of environmental pollution and the consequences for public health. Since Finland is the nearest country to FPNC and Helsinki is the largest city, the application of the conservative scenarios assures the maximum possible dose of human irradiation. The conservative scenario includes the specially selected meteorological conditions to provide the maximum possible effective dose of exposure to a reference person living in the considered settlement at a given release rate of radionuclides. Increasing the distance from the release point, while maintaining all other conditions, can only reduce the dose. Therefore, if the dose obtained under such conservative scenario for Finland is below the allowable limits, this assures that it will be below these limits for all other countries, which are at a greater distance, in particular Sweden (distance from FPNC to Stockholm is 340 km), and Latvia (distance to the country border is 155 km), Lithuania (distance to the country border is 330 km) and Russian Federation (distance to the border is 220 km while distance to St. Petersburg is 360 km).

The assessment of the consequences of the transboundary transport of a radioactive release in the event of a radiation accident in FPNC is carried out using the Lagrangian-Eulerian diffusion model of pollutant transport in the atmosphere LEDI.

Assessment of individual doses for the population was made taking into account the following pathway of the population radiation exposure dose formation after the accidental radioactive releases into the atmosphere: external exposure from radionuclides in the atmospheric air and deposited on the soil; internal exposure caused by the radionuclides intake into the human body with inhaled air and the consumption of contaminated local terrestrial food products and seafood. The population individual exposure doses were calculated for two reference groups of the population - “Adult” and “1-year old” - at the reference point located in Helsinki. The doses were calculated for 2 periods separately: 1) for the acute period of the accident (during the 1st day after the accident beginning) and 2) during the first year after the accident.

Special attention is paid to modelling the deposition of radionuclides from an accidental release on the water surface, their further migration in the Gulf of Finland, and the assessment of the corresponding contribution to the doses of internal exposure of the population due to the consumption of radioactively contaminated seafood. This problem was solved using the compartment model POSEIDON-R, which is a part of the European Decision Support System for emergency response to nuclear accidents RODOS.

5.2.2 Atmospheric dispersion modelling

Accidental radionuclide release scenarios description

Two release scenarios with maximum radiation consequences for the population and the environment were used to analyse the consequences of transboundary radionuclide transport due to accidental emissions into the atmosphere.

The accident "Fire in the MB/fire of an unpressurized SRW container" (fire in the reactor room of a RC accompanied by the burning of heat insulation) was considered the most severe design basis accident. The accident “Crash of an airplane/flying object” with a subsequent fire and parallel destruction of an industrial ionising radiation source was assumed as an accident with the maximum radiation consequences for the public. For both scenarios, the release duration was assumed to be 1 hour as a conservative estimate. The effective source height was estimated as 35 m due to a convective updraught formed over the fire area.

Meteorological scenarios

Three meteorological scenarios were formed that would provide conservative estimates of the radionuclide volume activity concentrations in the near-ground atmosphere and the radionuclide deposition density on the territory of Finland.

MeteoScenario_1 provides the transport of released radionuclides in the direction of Helsinki without precipitation under small values of wind speed in the lower atmosphere and the presence of stably stratified layers in the lower atmosphere. MeteoScenario_2 assumes the rainfall with the constant intensity of 1 mm per hour during the entire period of emission transport towards Finland.

MeteoScenario_3 assumed intensive local rain over Helsinki (22.5 mm of rain during 15 min) ensuring the local deposition spot formation in the reference point.

Results of simulation of radioactive contamination of the atmosphere and the underlying surface

Model results for the design basis accident

The model maximum 1-h averaged volume activity concentration of radionuclides in the surface air and the radioactive deposition density after the design basis accident for the reference point in Helsinki for 3 meteorological scenarios is presented in Table 5.2.1.

Table 5.2.1 Model results for the design basis accident

Nuclide	Maximum 1-h averaged volume activity concentration of nuclides in the air* (Bq/m^3)			Deposition density (Bq/m^2)		
	Meteo-Scenario_1	Meteo-Scenario_2	Meteo-Scenario_3	Meteo-Scenario_1	Meteo-Scenario_2	Meteo-Scenario_3
Co-60	8.78E-10	2.50E-10	8.78E-10	5.32E-08	3.53E-08	1.06E-05
Cs-137	1.08E-09	3.08E-10	1.08E-09	6.56E-08	4.36E-08	1.31E-05
Eu-152	6.17E-08	1.75E-08	6.17E-08	3.74E-06	2.48E-06	7.47E-04
Eu-154	5.46E-08	1.55E-08	5.46E-08	3.31E-06	2.20E-06	6.62E-04

* 14 hours after the accident beginning

Even the maximum value of 1-hour averaged Cs-137 volumetric activity in the case of transboundary transport of the release from FPNC is 3 orders of magnitude less than the background values of about 10^{-6} Bq/m^3 observed in Finland.

Model results for the hypothetical beyond design basis accident

The model maximum 1-h averaged volume activity concentration of radionuclides in the surface air and the radioactive deposition density after the hypothetical beyond design basis accident for the reference point in Helsinki for 3 meteorological scenarios is presented in Table 5.2.2.

Table 5.2.2 Model results for the hypothetical beyond design basis accident

Nuclide	Maximum 1-h averaged volume activity concentration of nuclides in the air* (Bq/m^3)			Deposition density (Bq/m^2)		
	Meteo-Scenario_1	Meteo-Scenario_2	Meteo-Scenario_3	Meteo-Scenario_1	Meteo-Scenario_2	Meteo-Scenario_3
Co-60	2.80E-03	7.95E-04	2.80E-03	1.69E-01	1.13E-01	3.39E+01
Pu-238	3.00E-05	8.54E-06	3.00E-05	1.82E-03	1.21E-03	3.64E-01
Sr-90	2.15E-06	6.10E-07	2.15E-06	1.30E-04	8.65E-05	2.60E-02

* 14 hours after the accident beginning

The maximum volume concentration of Co-60 activity in the surface air of Helsinki averaged over 1 hour, for meteorological scenarios 1 and 3 is 2800 $\mu Bq/m^3$. For Sr-90, this value is 2.15 $\mu Bq/m^3$, and the daily-averaged volume activity concentration during the first day after the accident is 0.25 $\mu Bq/m^3$. Both of these values exceed the background values of the Sr-90 volume activity concentration in the surface air in Finland (the measurement results were under a detection limit of 0.12 $\mu Bq/m^3$ in 2011).

Marine dispersion modelling

The POSEIDON-R model was used for the simulation of the transport of radionuclides in the marine environment including water and sediments, their uptake by biota, and the estimation of doses to humans from marine pathways of irradiation. This is the compartment (box) model where the marine environment is represented as a system of compartments (boxes) for the water column, bottom sediment, and biota. The exchange of radioactivity between the water column boxes is described by fluxes of radionuclides due to advection, sediment settling, and turbulent diffusion processes. POSEIDON-R uses the dynamic food web model for the simulation uptake of radionuclides by marine organisms. The model includes pelagic and benthic food chains.

NEMO-Nordic model as a source of water circulation data for the POSEIDON-R model

Correct values of water fluxes between boxes, which are calculated from available 3D velocity fields, are important for model customisation. Here 3D-currents from the circulation model NEMO-Nordic, which are available online, were used. To check the water velocities provided by the NEMO-Nordic model, they were compared with available measurement data in the locations of 4 buoys near the coast of Poland. The comparison showed that, in general, model currents are in agreement with measurements. This means that currents from the NEMO-Nordic model can be used for the calculation of water fluxes between boxes in the POSEIDON-R model.

POSEIDON-R customisation for the Gulf of Finland

The resolution of the NEMO-Nordic model is 1/30 degree in latitude and 1/18 degree in longitude which is about 3.7 km in both directions. Based on this resolution, the optimal size of boxes in the POSEIDON-R model is 15 × 15 km (4 × 4 calculation nodes). Such compartments were created in the Gulf of Finland between FPNC, Tallinn and Helsinki. Larger boxes were placed around them to prevent excessive mixing of contamination in the large volumes of seawater. The volume and average depth of each box were calculated based on the bathymetry data. Deep boxes were vertically subdivided on a surface layer (from the surface to a depth of 25 m) and a bottom layer (from a depth of 25 m to the bottom) to describe the activity stratification in the water column.

The water fluxes between boxes were calculated by averaging currents on their faces over a 10-year period (2009-2018) from the circulation model NEMO-Nordic. The water inflow of main rivers (Neva, Narva, Kymijoki) was also taken into account to have the correct dominant flow of water from the Gulf of Finland to the Baltic Sea. Parameters describing the water-sediment interaction in each box such as suspended sediment concentration and sedimentation rate, which are typical for the Gulf of Finland, were adopted from previous studies. The salinity of the Baltic Sea is lower than ocean salinity due to large river runoff and low water exchange with the Atlantic Ocean. It increases the uptake of radionuclides (especially isotopes of Cs and Sr) by marine organisms due to decreasing competition ions concentration. In the model, salinity changes from 1.5 in the Neva Bay to 8 in the western part of the Gulf according to data described in the literature.

Data for consumption rates of marine organisms are needed for the estimation of the doses to people from seafood consumption. In this study, the average annual human consumption of fish equal to 30.5 kg was used according to Food and Agriculture Organization of the United Nations data. However, this value includes the consumption of domestic marine and freshwater species and the consumption of imported fish. For conservative dose assessment, we assume that there is a reference person (group) that consumes only fish from the Gulf of Finland.

Sources of radionuclides in the POSEIDON-R model

The new system of boxes were integrated into the JRODOS system. Fields of atmospheric deposition of radionuclides, which are described above, were interpolated on the POSEIDON-R boxes and included in the JRODOS system. The activity deposited on each box was calculated by summing the deposition densities multiplied by the computation grid areas for grid nodes, which are located inside this box.

Results of modelling

The greatest contamination of the Gulf of Finland by radionuclides and corresponding doses to humans from seafood consumption will be for the scenario of accidental release with atmospheric precipitation, accompanied by the transfer of the radioactive cloud over the gulf (MeteoScenario_2 for hypothetical beyond design basis accident). Simulation results show that the highest concentration of radionuclides will be near the Estonian coast close to FPNC. Since, Co-60 is the dominant radionuclide, the concentration of Co-60 will exceed concentrations of Sr-90 and Pu-238 in the first months after the atmospheric deposition. However, these concentrations will be quite small – the maximum will not exceed 0.01 Bq/m^3. Later the concentration of each radionuclide will change in different ways. Due to the high ability of Co-60 to be adsorbed by sediments, the concentration of Co-60 in water will decrease faster than other radionuclides. Therefore, 1 year after the atmospheric deposition, the concentration of Co-60 in water will be lowest. Water currents directed mostly from the Gulf of Finland to the Baltic Sea also lead to a decrease in the concentration of radionuclides in water due to the outflow of radionuclides from the Gulf of Finland and their dilution by seawater.

Intensive deposition of Co-60 on the bottom will contaminate bottom sediments more than other radionuclides. But, as in the case of water, the concentration will be quite low and not exceed 0.1 Bq/kg. The lowest contamination of bottom sediments will be from Sr-90 due to its low ability to interact with sediments.

For most radionuclides and heavy metals, there is an inverse relationship between trophic levels and the concentration of radionuclides in aquatic organisms. This means that the higher concentration of radionuclides will be in organisms from lower trophic levels. They are pelagic and demersal non-predatory fish in the POSEIDON-R model. The concentration of radionuclides in them will be higher than in predatory types of fish. Among the three considered radionuclides, the concentration of Pu-238 will be the highest for pelagic fish, while the concentration of Co-60 will be the highest for demersal fish. The concentration of Sr-90 in fish will be the lowest.

Simulated concentrations of Co-60, Sr-90 and Pu-238 in water, bottom sediments and marine biota for MeteoScenario_1 and MeteoScenario_3 in the case of hypothetical beyond design basis accident were obtained very similar to corresponding concentration for MeteoScenario_2 described here. The reason is in very similar deposition densities provided by the atmospheric dispersion model. Simulated concentrations of Co-60, Cs-137 and Eu-152, Eu-154 in the marine environment for the case of design basis accident were 6 orders of magnitude less that is in accordance with atmospheric deposition densities.

5.2.3 Calculated exposure doses of the population due to the possible accidents

Population exposure doses from the atmospheric and terrestrial pathways of irradiation

Using the results of modelling of the air radioactive contamination and the radionuclide deposition density as a result of emissions into the atmosphere, radiation doses for the acute period of the accident (during the 1st day after the accident beginning) and for the first year after the accident were calculated for the population categories “Adult” and “1 year old” in the reference point in Helsinki for 3 considered meteorological scenarios.

Acute period of the accident for the design basis accident

The exposure dose of the population (for 2 age groups) for the design basis accident during the acute period (1 day after the accident beginning) is calculated taking into account the external exposure from radionuclides in the atmospheric air (radioactive cloud immersion), and internal exposure caused by the radionuclides intake into the human body with inhaled air. The total effective dose of the population for the design basis accident during the acute period for 3 meteorological scenarios is presented in Table 5.2.3.

Table 5.2.3 Total effective dose of the population for the design basis accident during the acute period for 3 meteorological scenarios, µSv

	Adult	**1 year old**
MeteoScenario_1	1.79E-08	7.85E-09
MeteoScenario_2	4.88E-09	2.14E-09
MeteoScenario_3	1.47E-08	6.44E-09

The maximum values of the total effective dose were obtained for meteorological scenario 1. The inhalation pathway contributes about 99% for both age groups.

The first year after the accident for the design basis accident

The exposure dose of the population (for 2 age groups) for the design basis accident during the first year after the accidental release for 3 considered meteorological scenarios is calculated. During this period, radioactive contamination of the near-surface air is determined by the resuspension of radionuclides deposited on the underlying surface during the initial period of the accident. The following pathways of population exposure were considered:

- External exposure from radionuclides deposited on the soil;

- Internal exposure caused by the radionuclides intake into the human body with air contaminated due to resuspension (inhalation pathway);
- Internal exposure caused by the consumption of contaminated local food products (ingestion pathway).

The total effective dose of the population for the design basis accident during the first year after the accident for 3 meteorological scenarios is presented in Table 5.2.4.

Table 5.2.4 Total effective dose of the population for the design basis accident during the first year after the accident for 3 meteorological scenarios, μSv

	Adult	1 year old
MeteoScenario_1	2.95E-07	3.11E-07
MeteoScenario_2	1.96E-07	2.07E-07
MeteoScenario_3	5.90E-05	6.21E-05

The external dose from depositions contributes about 88% to the total effective dose for adults and about 84% for 1-year-old infants. The part of the ingestion pathway is about 11% of the total effective dose for adults and about 16% for 1-year-old infants, and the part of the inhalation pathway is about 0.4% for adults and about 0.2% for 1-year-old infants.

The europium isotopes contribute 95-96% to the total effective dose for both age groups. They contribute almost 70% of the internal exposure dose due to the consumption of contaminated food (without seafood) for adults and above 86% for 1-year-old infants.

The main contribution to the internal dose due to the food consumption (without seafood) for the design basis accident during the first year after the accident for each of all considered meteorological scenarios is determined by the consumption of root vegetables (31-34%), grain (29-31%) and fruit (11-19%) for both age groups. The largest difference between the two age groups for the relative contribution of considered products is obtained for meat (15.0% for adults against 1.5% for infants). The main reasons are the low consumption of meat by infants compared to adults relative to other products, and the significantly smaller (7 times) ingestion dose coefficient for infants compared with adults for Cs-137.

Unlike for the initial period of the design basis accident, the "1-year-old age" is obtained to be the critical group. The total effective dose for 1-year-old infants under the worst-case MeteoScenario_3 is obtained at $6.21 \cdot 10^{-5}$ µSv, for adults - $5.90 \cdot 10^{-5}$ µSv. These values are significantly lower than the established limit of the individual effective dose of 1 mSv year^{-1}.

Acute period of the accident for the hypothetical beyond design basis accident

The exposure dose of the population (for 2 age groups) for the hypothetical beyond design basis accident during the acute period (1 day after the accident beginning) for 3 considered meteorological scenarios is calculated due to external exposure from radionuclides in the atmospheric air, and internal exposure caused by the radionuclides intake into the human body with inhaled air. The contribution of the Y-90 nuclide resulting from the radioactive decay of Sr-90 to the values of total external exposure doses from deposits and cloud immersion was taken into account. The total effective dose of the population for the hypothetical beyond design basis accident during the acute period for 3 meteorological scenarios is presented in Table 5.2.5.

Table 5.2.5 Total effective dose of the population for the hypothetical beyond design basis accident during the acute period for 3 meteorological scenarios, µSv

	Adult	1 year old
MeteoScenario_1	4.75E-03	1.33E-03
MeteoScenario_2	1.29E-03	3.62E-04
MeteoScenario_3	3.89E-03	1.09E-03

As for the initial period of the design basis accident, the critical group of the population is found to be the age group "Adult". The maximum values of the total effective dose were obtained for meteorological scenario 1. The inhalation pathway contributes over 99% of the total effective dose for both age groups. The part of the Pu-238 nuclide in the total dose is 90-94%.

The first year after the accident for the hypothetical beyond design basis accident

The exposure dose of the population (the age groups “Adult” and “1 year old”) for the hypothetical beyond design basis accident during the first year after the accident for 3 considered meteorological scenarios is calculated taking into account the external exposure from radionuclides deposited on the soil, and internal exposure caused by the radionuclides intake into the human body with air contaminated due to resuspension, and the consumption of contaminated local food products. The total effective dose of the population for the hypothetical beyond design basis accident during the first year after the accident for 3 meteorological scenarios is presented in Table 5.2.6.

Table 5.2.6 Total effective dose of the population for the hypothetical beyond design basis accident during the first year after the accident for 3 meteorological scenarios, µSv

	Adult	1 year old
MeteoScenario_1	1.56E-02	1.76E-02
MeteoScenario_2	1.04E-02	1.17E-02
MeteoScenario_3	3.12E+00	3.51E+00

The external dose from depositions contributes about 81% to the total effective dose for adults and about 72% for 1-year-old infants. The part of the ingestion pathway is about 16% of the total effective dose for adults and about 27% for 1-year-old infants, and the part of the inhalation pathway is about 1.9% for adults and about 0.5% for 1-year-old infants.

The Co-60 contributes 93-97% to the total effective dose for both age groups. The part of Pu-238 is about 7% for adults and about 3% for infants. Co-60 contributes almost 68% of the internal exposure dose due to the consumption of contaminated food (without seafood) for adults and about 90% for 1-year-old infants. The part of Pu-238 is above 31% for adults and almost 10% for infants.

The main contribution to the internal dose due to the food consumption (without seafood) for the hypothetical beyond design basis accident during the first year after the accidental release for each of all considered meteorological scenarios is determined by the consumption of root vegetables (31-32%), grain (28-29%) and fruit (10-18%) for both age groups. For adults, the input of meat in the internal dose due to food consumption increases up to about 24%, while for infants the contribution of milk to the internal ingestion dose is about 11%. Differences in the relative contribution of various foodstuffs to the total exposure dose obtained for the two accidental release scenarios are explained by their different nuclide composition.

The "1-year-old age" is obtained to be the critical group for this accident. The total effective dose for 1-year-old infants under the worst-case MeteoScenario_3 is obtained at 3.51 µSv, and for adults - 3.12 µSv. These values are significantly lower than the established limit of the individual effective dose of 1 mSv year^{-1}.

Population exposure doses from the marine pathways of irradiation

Exposure doses to humans from seafood consumption were calculated based on simulated by POSEIDON-R model concentrations of radionuclides in marine organisms. A maximal dose was obtained for the MeteoScenario_2 for the reference person consuming all fish in their diet from the Gulf of Finland. The conservative estimations of the dose were made assuming that the reference person consumes fish from the modelling box closest to FPNC, where maximal concentrations of radionuclides in fish were obtained in simulations. The effective doses to "Adult" due to seafood consumption from all considered radionuclides for 1st year after the accident are presented in Table 5.2.7.

Table 5.2.7 Effective doses to "Adult" due to seafood consumption from all considered radionuclides for 1st year after the accident, µSv

	Design basis accident	Beyond design basis accident
MeteoScenario_1	6.2×E-09	6.4×E-03
MeteoScenario_2	1.0×E-08	1.1×E-02
MeteoScenario_3	6.2×E-09	6.4×E-03

The maximal dose was obtained for MeteoScenario_2 (when precipitation occurred), which is approximately 2 times higher than for MeteoScenario_1 and MeteoScenario_3.

Total population exposure doses from all pathways of irradiation

The total effective doses to "Adult" from all pathways of irradiation for 1st year after the accident are presented in Table 5.2.8.

Table 5.2.8 Total effective doses to "Adult" from all pathways of irradiation for 1st year after the accident, μSv

	Design basis accident	Beyond design basis accident
MeteoScenario_1	3.01×E-07	2.20×E-02
MeteoScenario_2	2.06×E-07	2.14×E-02
MeteoScenario_3	5.90×E-05	3.12

The contributions for the design basis accident and the hypothetical beyond design basis accident are different due to different sets of radionuclides released in these scenarios. For the design basis accident, the doses from seafood consumption contribute from 4.9% to <0.01% in the total dose only depending on the meteorological scenario. For the hypothetical beyond design basis accident, the input of marine pathways varies from 51.4% (MeteoScenario_2 when atmospheric precipitation occurred along the path of the contaminated cloud) to 0.2% (For MeteoScenario_3 with intensive precipitation over Helsinki city).

5.2.4 Conclusions

The total effective doses of public exposure during the acute period of the accident and the first year after it for Finland (at the reference point in Helsinki) will be significantly lower than the established limit of the individual effective dose 1 $mSv*year^{-1}$. Calculations of radioactive contamination of the air, the earth's surface, the marine environment and the corresponding doses to the population as a result of the transboundary transport of accidental radioactive releases from FPNC showed no significant negative effects on the environment and public health in Finland. The conclusions about the fulfilment of the established safety criteria for the population, obtained as a result of calculations for Finland, will be all the more likely for other countries, as including Latvia, Lithuania, Sweden, and Russian Federation.

5.3 Risk analysis and assessment

Risk assessment of RCs decommissioning was done within sub-activity 4.10.

5.3.1 Risk assessment methodology

Risk assessment for decommissioning of RCs at FPNC was performed using a graded approach directed to available data about the planned decommissioning process and following Regulation No. 28 of the Minister of the Interior of 19 June 2017, "Requirements for an emergency risk assessment and procedure for the preparation of a risk assessment" (further on – "Regulation 28"). The procedure provided by Regulation 28 was adopted for the purpose of this study and extended with the methods and data used in Probabilistic Safety Analysis for nuclear facilities, following best available international practice and recommendations. Risk analysis included the following steps:

- Development of emergency scenarios;
- Assessment of the probabilities of the emergency scenarios;
- Assessment of the consequences of the emergency scenarios;
- Risk categorisation of the emergency scenarios;
- Overview (including the development of a risk matrix) and conclusions regarding risks related to the dismantling of the RCs.

5.3.2 General assumptions

The risk analysis provided in this report was based on the following approach and assumptions:

- Information regarding FPNC, including a description of decommissioning process together with considered emergency scenarios was obtained from SAR developed in the course of Sub-activity 4.8.
- Information used for the analysis of external emergency scenarios regarding the area and FPNC characteristics including climate, and seismology, was obtained from reports of Activity 2 „Studies of the three repository locations".
- The worst possible consequence is assumed for each emergency scenario.
- Specific assumptions, data used and supporting calculations are provided in the scope of analysis of each emergency scenario.

5.3.3 Development of scenarios

A list of emergency scenarios subject to the risk analysis is provided in Table 5.3.1 below.

Table 5.3.1 List of emergency scenarios subject to the risk analysis

SAR Scenario	Frequency from SAR, 1/year	Consequences from SAR	Scenario ID and title for Risk Analysis
Internal emergency scenarios (IDS)			
Falling loads/collapse of upper slabs	10^{-6}=< P2 <10^{-2}	The 50-year lifetime dose of internal exposure for personnel taking into account 5-minute external exposure from reactor No. 1 will be 60.1 µSv	IDS-01 Dropping the reactor vessel
Fire in the MB /fire of a leaky SRW container	10^{-6}=< P2 <10^{-2}	The total effective dose under this scenario will not exceed 2.91E-08 Sv	IDS-02 Fire in the MB
LRW spill	P1 >10^{-2}	Local dose rate 5.6·10-2 µSv/h	IDS-03 LRW spill
Exposing the ionising radiation source during dismantling works inside RC No. 1	P1 >10^{-2}	A single individual exposure dose will not exceed 0.1 mSv	IDS-04 Exposing IRS
Power loss	P1 >10^{-2}	Related to falling loads	IDS-05 Internal power loss
Human error	P1 >10^{-2}	Lead to other initiating events, both internal and external	Contribution to other scenarios IDS-01 – IDS-05
Military, sabotage and reconnaissance group or terrorist attack	P1 >10^{-2}	Not identified. Consequences like an aircraft crash are assumed for risk analysis	IDS-06 Terrorist attack
External emergency scenarios (EDS)			
Earthquake	P1 >10^{-2}	Related to falling loads	EDS-01 Earthquake
Hurricanes, storms, wind	P1 >10^{-2}	Related to falling loads	EDS-02 Extreme wind
Floods	P1 >10^{-2}	Related to LRW spill	EDS-03 External flood
External fires	10^{-6}=< P2 <10^{-2}	Related to fire	EDS-04 External fire
Snow load	P1 >10^{-2}	Related to falling loads	EDS-05 Snow load
Extreme temperatures	P1 >10^{-2}	Related to fire	EDS-06 Extreme temperature

SAR Scenario	Frequency from SAR, 1/year	Consequences from SAR	Scenario ID and title for Risk Analysis
Lightning strike	P1 $>10^{-2}$	Related to fire	EDS-07 Lightning strike
Explosions at or near the site	10^{-6}=< P2 $<10^{-2}$	Related to fire	EDS-08 External explosion
Airplane/flying object crash	P3 $< 10^{-6}$	The total effective dose under 1.644E-03 Sv	EDS-09 Aircraft crash
External power loss	P1 $>10^{-2}$	Related to falling loads	EDS-10 External power loss

A scenario includes three main parts:

- Initiating event (or initiator, which starts a scenario);
- Scenario or events sequence;
- Consequences.

In many cases scenarios having different initiators, have the same consequences.

5.3.4 Assessment of probabilities

Initiators are random stochastic events characterised by probability (or frequency, i.e. expected number of events over a time period). Initial values of initiators' frequencies were taken from SAR (see Table 1 above) and, where possible, were refined based on available data, such as generic reliability databases provided by the US NRC and IAEA, historical data, expert judgement. In cases when different values were provided by several data sources, the highest („the worst") values were used in accordance with the conservative approach.

Obtained values were then subdivided using the five grades scale, based on Annex 2 to the Regulation 28 where 'A' corresponded to the lowest, 'E' – to the highest probability (see

Table 5.3.2 below).

Table 5.3.2 Probabilities assessment

Value in acronym	A	B	C	D	E
Value in words	very low	low	average	high	very high
Criterion	less than once every 100 years	once every 50÷100 years	once every 20–50 years	once every 5÷20 years	more than once every 5 years
Frequency, events/year	$< 10^{-2}$	$[10^{-2}; 2\cdot10^{-2}]$	$[2\cdot10^{-2}; 5\cdot10^{-2}]$	$[5\cdot10^{-2}; 2\cdot10^{-1}]$	$> 2\cdot10^{-1}$

5.3.5 Assessment of consequences

Consequences of emergency scenarios are evaluated for the considered scenarios based on calculations provided in the SAR, Sub-activity 4.8. The main criterion used to evaluate the severity of the consequences was radiation dose uptake by personnel or population. An additional criterion was direct financial cost.

The severity of the consequences was then subdivided using a five-grade scale from '1' to '5' based on Annex 3 to the Regulation 28 where '1' corresponds to the lowest, '5' – to the highest severity (see Table 5.3.3 below).

Table 5.3.3 Consequences assessment

Severity	1	2	3	4	5
Value in words	insignificant	minor	severe	very severe	catastrophic
I Life and health					
Deceased (number)	≤ 5	6–15	16–50	51–200	> 200
Injured or taken ill (number)	≤ 15	16–45	46–150	151–600	> 600
Evacuated (number)	≤ 50	51–200	201–500	501–2000	> 2000
II Property					
Direct financial cost (MEUR)	< 1	1–10	11–50	51–100	> 100

5.3.6 Risk estimate categorisation

Risk category R can be expressed as a "value of risk" and is simply a multiplication of risk probability P and risk consequences C: $R = P \cdot C$.

In accordance with the requirements and the criteria provided in Annex 4 of the Regulation 28 risk category for each emergency scenario is defined based on the probability and severity of the considered scenario using 5 × 5 grid known as „risk matrix" (see Table 5.3.4 below).

Table 5.3.4 Risk estimate categorisation

		CONSEQUENCE				
		Insignificant (1)	**Minor (2)**	**Severe (3)**	**Very severe (4)**	**Catastrophic (5)**
PROBABILITY	**Very high (E)**	average	significant	high	very high	very high
	High (D)	average	significant	significant	high	very high
	Average (C)	low	average	significant	high	high
	Low (B)	low	average	significant	significant	high
	Very low (A)	low	low	average	significant	high

Risk categories then can be used for comparing the risks and defining priorities of applying risk preventive measures.

5.3.7 Risk analysis for decommissioning of RCs

Overview of risks due to internal emergency scenarios is presented in

Table 5.3.5. Overview of risks due to external emergency scenarios is presented in Table 5.3.6. The risk matrix for the decommissioning of the RCs is provided in Table 5.3.7.

Table 5.3.5 Risk categories of the internal emergency scenarios

ID	Description	Probability	Severity	Risk category
IDS-01	Dropping the reactor vessel	A – very low	1 – insignificant	LOW
IDS-02	Fire in the MB	A – very low	1 – insignificant	LOW
IDS-03	LRW spill	D - high	1 – insignificant	AVERAGE
IDS-04	Exposing IRS	D - high	1 - insignificant	AVERAGE
IDS-05	Internal power loss	A – very low	1 – insignificant	LOW
IDS-06	Terrorist attack	B – low	2 – minor	AVERAGE

Table 5.3.6 Risk categories of the external emergency scenarios

ID	Event leading to a scenario	Probability	Severity	Risk category
EDS-01	Earthquake	A – very low	1 – insignificant	LOW
EDS-02	Extreme wind	E – very high	1 – insignificant	AVERAGE
EDS-03	External flood	B – low	1 – insignificant	LOW
EDS-04	External fire	C – average	1 - insignificant	LOW
EDS-05	Snow load	C – average	1 – insignificant	LOW
EDS-06	Extreme temperature	C – average	1 – insignificant	LOW
EDS-07	Lightning strike	B – low	1 – insignificant	LOW
EDS-08	External explosion	B – low	1 – insignificant	LOW
EDS-09	Aircraft crash	A – very low	2 – minor	LOW
EDS-10	External power loss	A – very low	1 – insignificant	LOW

Table 5.3.7 Risk matrix of the emergency scenarios

		CONSEQUENCE (SEVERITY)				
PROBABILITY		Insignificant (1)	Minor (2)	Severe (3)	Very severe (4)	Catastrophic (5)
	Very high (E)	EDS-02				
	High (D)	IDS-03 IDS-04				
	Average (C)	EDS-04 EDS-05 EDS-06				
	Low (B)	EDS-03 EDS-07 EDS-08	IDS-06			
	Very low (A)	IDS-01 IDS-02 IDS-05 EDS-01 EDS-10	EDS-09			

Risk preventive measures were developed for the considered scenarios based on the findings of risk analysis, namely the main factors of the risk, weak points of design and operation. Developing risk preventive, recommendations provided in IAEA documents were considered as well, adopting them to the decommissioning of the RCs. Overview of preventive measures is provided in Table 5.3.8.

Table 5.3.8 Overview of preventive measures

ID	Scenario	Risk category	Preventive measures
IDS-01	Dropping the reactor vessel	LOW	1. Crane operators' skills and training 2. Detailed procedure for the reactor body handling operations 3. Reliable lifting equipment provided by the decommissioning design
IDS-02	Fire in the MB	LOW	Fire safety measures to be described in the dismantling design

ID	Scenario	Risk category	Preventive measures
IDS-03	LRW spill	AVERAGE	Design of the residual water pumping unit preventing LRW spill possibility, including: 1. Sturdy positioning of outlet hose/pipe 2. Preventing overfilling of the drum/container. 3. Special tray for spill collection
IDS-04	Exposing IRS	AVERAGE	1. Use of ultrasonic scanning (or other state-of-the-art technology) for IRS location 2. Ensuring high skills and a high level of safety culture 3. In-situ radiation monitoring 4. Detailed special procedure for IRS retrieval
IDS-05	Internal power loss	LOW	. Ensuring power supply reliability depending on the equipment safety class 2. Technical means preventing load drop in case of loss of power supply
IDS-06	Terrorist attack	AVERAGE	1. Security measures (fence, gates, etc.) 2. Speed limitation equipment and anti-ram barriers equipment 3. Access control for persons and vehicles
EDS-01	Earthquake	LOW	See IDS-01 (internal event “Dropping the reactor vessel”) plus: 1. Seismicity measurements/monitoring system; 2. Seismically qualified external equipment; 3. Preventive preparations.
EDS-02	Extreme wind	AVERAGE	See IDS-01 (internal event “Dropping the reactor vessel”) plus: 1. Wind monitoring system; 2. Preventive preparations; 3. Autonomous power supply.

ID	Scenario	Risk category	Preventive measures
EDS-03	External flood	LOW	See IDS-03 (internal event “LRW spill”) plus: 1. Proper drainage system; 2. Floods warning system; 3. Preventive preparations.
EDS-04	External fire	LOW	See IDS-02 (internal event “Fire in the MB”) plus: 1. Fire/smoke notification system; 2. Passive fire protection barriers; 3. Smoke spread limitation systems.
EDS-05	Snow load	LOW	See IDS-01 (internal event “Dropping the reactor vessel”) plus: 1. Route and building cleaning; 2. Inlets or outlets clearing.
EDS-06	Extreme temperature	LOW	See IDS-02 (internal event “Fire in the MB”) plus: 1. Temperature monitoring/predicting; 2. Extreme temperature management.
EDS-07	Lightning strike	LOW	See IDS-02 (internal event “Fire in the MB”) plus: 1. Lightning strike mitigation/notification system; 2. Electromagnetic interference management.
EDS-08	External explosion	LOW	See IDS-01 (internal event “Dropping the reactor vessel”), IDS-02 (event “Fire in the MB”) plus: 1. Preventive distance; 2. Explosion notification system.
EDS-09	Aircraft crash	LOW	See IDS-01 (internal event “Dropping the reactor vessel”), IDS-02 (event “Fire in the MB”) plus: 1. Flight prevention/prohibition zone; 2. Aircraft crash management.
EDS-10	External power loss	LOW	See IDS-05 (event “Internal power loss”) plus: 1. Redundant power supply lines; 2. Emergency power system.

5.4 Conclusions

The developed safety case allows concluding that the decommissioning technology of the FPNC complies with Estonian and international safety standards.

To exclude uncertainties in estimates associated with possible changes in regulatory requirements, conditions for the implementation of on-site activities taking into account the knowledge gained as a result of other similar work at other facilities or sites (including international experience) and considering the characteristics of equipment to be selected through the design, the SAR should be revised after the development of the final decommissioning design for the FPNC.

6. EXECUTIVE SUMMARY

Engineering study of the condition of the MB of the Paldiski site (MB) found that the load-bearing structure of the building is sufficient to withstand the effective loads that it is subjected to. No design flaws or construction flaws were discovered. The building envelope, however, has deficiencies. Wall panels are the largest source of problems. The concrete is of very uneven quality there, and the surface is cracked in places. These wall elements do not play a very important role in terms of the load-bearing capacity of the building as a whole, but the crumbling of the protective layer can primarily be dangerous for the occupants of the buildings. Due to existing damage, the residual life of several panels may not exceed 10 years. However, it must be taken into account that it is very likely that this damage has largely occurred already in the period that preceded the last major repair, when extensive leaks occurred.

Also, some roof-ceiling panels have visible damage. They are not extensive and large. They originate from the time before the last leaks occurred. For example, around the in-building drainage pipes and roof outlets. At the same time, the load condition of the roof ceiling panels has become significantly more favourable, due to the fact that they are no longer affected by snow load, since the base structure of the pitched roof allows the load to concentrate on the roof ceiling beams. There is salt and moisture damage in the masonry, which also dates back to the time before repairs and demolition works, when the building still had adjacent parts. These joining parts to the building were likely the places of leaks. Plaster has fallen off from the walls and the compressive strength of the masonry mixture has decreased. The situation has been remedied by installing additional anchors to the frame posts. There were no signs visible of foundation sinking, but given that the foundations are resting on limestone, this is rather expected to find, since no consolidation of soil cannot occur.

Based on the findings, it can be said that the load-bearing structures are in good condition and the initial building envelope elements are in satisfactory condition. But since there is practically a new building envelope built around the initial one, then the deterioration has stopped. Still the situation must be regularly monitored. Therefore, it can be said that with proper maintenance the building will be durable at least until 2040. Also, with proper maintenance, the cranes can be used by 2040, but that also requires regular inspection by a certified partner. For decommissioning, it should be noted that the lifting capacity of the cranes is currently limited to 30 tonnes, but it would be possible to restore it to 50 tonnes as it was initially.

Engineering-technical study of the building materials and structure of the interim storage facility of radioactive waste (RW) showed that the load-bearing structure of interim storage is sufficient to withstand the effective loads that it is subjected to. There no design flaws or construction flaws were discovered. However, there were visible cracks that may occur for example shrinking of concrete and these must be repaired. There were no signs visible of foundation sinking, but given that the foundations are resting on limestone, this is rather expected to find, since no consolidation of soil cannot occur. Based on the findings, it can be said that the concrete structures are in very good condition. Taken into account that the structure is not subjected to external climate, it can be said that it will be durable at least until 2040. The doors, which need regular maintenance, however, need replacing before 2040.

Study of the structure of the reactor sarcophagi and the reactor compartments (RCs) found that the main problems are related to the upper surfaces of the ceilings of the sarcophagus No. 1. Compressive strengths there are relatively low and, in addition, cracks can be seen on the upper surface of the ceiling of sarcophagus No. 1 in the old part of the building. This seems to be a levelling layer with an apparent shrinkage crack, so this damage should not affect the bearing capacity. However, it is advisable to fill the cracks and monitor their possible future development. There were no signs visible of foundation sinking, but given that the foundations are resting on limestone, this is rather expected to find, since no consolidation of soil cannot occur. Based on the findings, it can be said that the concrete structures are in good condition. Taken into account that the structure is nor subjected to external climate, it can be said that it will be durable at least until 2040. The doors can also last until 2040, since their default position is closed position, and they are rarely used.

The **3D CAD models of RCs No. 1 and No. 2** were created using the computer program SolidWorks version 2013. The models contain all the available information on the geometry and placement of equipment/components found in previously produced reports and drawings.

Radiological study in the MB of the Paldiski site included a survey of gamma-ray dose rates in the CA, spectrometric analysis of gamma emitters' activity in situ, measurement of the total gamma and beta, the total alpha surface contamination, determination of nuclide specific activity in samples. It was found that there are two contaminated areas in the MB near the RC No. 1, a wall adjacent to the NFP and structures of the NFP. The gamma-ray dose rate inside the NFP is up to 1.8 µSv/h at the bottom. The dose rate is 0.24 ± 0.03 µSv/h on a wall adjacent to the NFP and exceeds the dose rate in all other areas when measurements are done with shielded devices to exclude of the influence of radiation sources stored in MB. 87 smears and 51 volume samples have been taken on every wall for the determination of difficult-to-measure radionuclide concentrations in the laboratory. The results of the destructive analysis performed in the samples from contaminated areas have been used for the determination of the NV. The results of the analysis of the ratios of nuclide activities show that the contamination is of the reactor origin. The rest of the CA can be classified as non-contaminated as being below exclusion and release levels of activity concentrations for all samples. The analysis of historical data, measurement of nuclide specific activity and a simulation of nuclide generation in the reactors has shown that 13 nuclides are relevant to long-term radiation safety: C-14, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Cs-137, Eu-152, Eu-154, Ra-226, Pu-238, Pu-239, Am-241. Conservative activity concentration values of radionuclides C-14, Cs-137, Eu-152 and E-154 in the contaminated areas are higher than the ECL. The total volume of both waste streams is about 216 m^3 and the total mass is about 521,000 kg. All other areas in MB after the removal of radioactive sources related to current waste management can be classified as non-contaminated in the year 2041.

The same approach (methods, measurement instruments) to the **radiological survey of the RCs** as one used for the radiological survey of the MB was applied. The gamma-ray dose rates have been measured on walls, floor, ceiling of RCs as well as on the surface of the metal shell of the RCs. The gamma-ray dose rate measurements show that the value (0.13±0.02) µSv/h is typical for all places in sarcophagi. Higher levels of a gamma-ray dose rate were measured in the rooms below reactors. The maximum value of the gamma-ray dose rate in the room below reactor No. 2 was 10 µSv/h. The highest gamma-ray dose rate is in the room below reactor No. 1. The maximum value of the gamma-ray dose rate in this room was 0.4 mSv/h. In situ gamma spectrometry, the determination of nuclide concentrations in samples in the laboratory showed that the increase of gamma-ray dose rate is caused not by surface contamination but by Co-60 radiation from activated reactor structures.

The results of the radiological study of the RCs showed that no surface contamination is present in all area inside sarcophagi accessible without the destruction of a submarine metal shell. As a result, it is not anticipated any waste streams from decommissioning of constructions of RCs, which are outside the submarine shell.

There are inaccessible regions inside submarines' shells. Characterisation of these regions can be only based on available historical data and simulation of the generation of radionuclide specific activity by creating a radiological model. For this purpose modelling of nuclear fuel depletion in the reactors with SCALE6.2 code package and by modelling reactor constructions activation with MCNP6 code have been done. Simulated ratios of radionuclide activities were used for the determination of the NV. The NV was applied for conservative estimation of radionuclide inventory in the waste streams produced from metal equipment and RW embedded in concrete. Characteristics of the waste streams produced after dismantling of all equipment inside submarine shells for the year 2039 are summarised in Table 1.

Table 1 Characteristics of waste streams from decommissioning of RCs

Waste type	Volume, m^3	Mass, kg	Nuclide	Activity, Bq
LLW-SL	89.36	1.53E+05	Co-60	1.89E+07
			Ni-59	2.51E+07
			Ni-63	1.80E+09
			C-14	3.07E+08
			Sr-90	1.18E+07
			Nb-94	5.63E+05
			Cs-137	5.63E+08
			Eu-152	2.88E+08
			Eu-154	1.35E+07
			Pu-238	3.16E+04
			Pu-239	6.76E+04
			Pu-240	1.97E+04
			Am-241	3.94E+04
LILW-LL	147.13*	2.28E+05*	Co-60	4.26E+11
			Ni-59	1.34E+12
			Ni-63	7.98E+13

			C-14 Sr-90 Nb-94 Cs-137 Eu-152 Eu-154 Pu-238 Pu-239 Pu-240 Am-241	1.16E+13 1.76E+09 4.14E+08 8.38E+10 1.51E+12 4.81E+11 4.69E+06 1.01E+07 2.93E+06 5.87E+06

It is supposed at least 0.58 m^3 of waste can be filled in 1 m^3 concrete container with an outer volume of 1.728 m^3. For about 226 m^3 {28.0 (spent sealed sources in concrete + very low-level waste embedded in concrete) + 89.36 (LLW-SL) + 108.25 (LILW-LL)} of raw waste is needed 390 containers with an outer volume of about 673 m^3**.** The estimated disposal volume from decommissioning of RCs in previous studies (2014-2015) with reactors (2 × 35 m^3) was 987 m^3. Disposal volume estimation from current studies including reactors is 743 m^3 (673+70), i.e. 24.7 % less than from studies 2014-2015.

The laboratory determination of activity concentrations of alpha emitters Pu-238, Pu-239+240, beta emitters C 14, Ni-63, Sr-90 and gamma emitters Co-60, Nb-94, Ag-108m, Cs-137, Am-241 in 4 samples taken from the steam generators removed from the RC No. 1 and 16 samples of concreted sediments of liquid waste generated from the operation of both reactors have been done.

In samples of concreted sediments of liquid waste, the most active gamma emitter is Cs-137, whose activity concentration is in the range 6370 ÷ 317,000 Bq/kg, the most active beta emitter is C-14, whose activity concentration is in the range 29,600 ÷ 126,000 Bq/kg. The activity concentration of alpha emitters is far below the exclusion and exemption levels.

In samples of steam generators, the most active gamma emitter is Co-60, whose activity concentration is in the range 40 ÷ 246,000 Bq/kg, the most active beta emitter is C-14, whose activity is in a range of 900 ÷ 404,000 Bq/kg. The activity concentration of alpha-emitting nuclides in 3 samples is above the exclusion and exemption levels, typical value is equal to a few hundreds of Bq/kg.

Measurement and analysis of gamma-ray spectra from the concrete containers BE 251 and BE 252 with spent sealed sources removed from the Tammiku waste storage, which are stored at Interim Storage now, has been done by a portable spectrometer (Ortec instrument ISO-Cart-85). Simulation of the gamma spectra from containers has been performed by MCNP6.2 code package for various localisation of a radiation source inside the containers, the density of containers' wall concrete, the density of waste fill.

Comparing measurements of BE 251 container with simulated distributions, the closest configuration corresponds to 2.2 g/cm^3 density of wall concrete, 0.5 g/cm^3 density of waste fill and a Cs-137 source placed 15 cm sideways from the center of the container. This conservatively could correspond to 1000 GBq of Cs-137 source activity. For the BE 252 container, the similar source distribution gives 30 GBq of Co-60 source activity and 35,000 GBq of Cs-137 activity.

New determination of the activity of main gamma emitters (Eu-152 and Eu-154) of decommissioned control rods by modelling using MCNP6.2 code package and reassessment of the results of gamma-ray spectrum measurements outside containers with the control rods have been performed. The previously measured gamma spectra have been analysed by automatised ISO-CART procedure, and, most probably, analysis was not performed based on the highest gamma yield but low gamma energy peaks.

The new total activity of radionuclides Eu-152 and Eu-154 was determined with an uncertainty lower than the available one by simulation of radionuclide generation in the reactor and absorption of gamma rays in the containers.

The new values of Eu isotopes activity for the present time (12/1/2023) are as follows:

CRC-1:	Eu-152	4.6 ± 1.6 TBq,	Eu-154	0.6 ± 0.2 TBq;
CRC-2:	Eu-152	31 ± 11 TBq,	Eu-154	3.1 ± 1.4 TBq;
CRC-3:	Eu-152	6.8 ± 2.8 TBq,	Eu-154	1.0 ± 0.4 TBq;
CRC-4:	Eu-152	0.2 ± 0.1 GBq,	Eu-154	< 0.05 GBq.

A **radiological survey of the territory of Former Paldiski Nuclear Center** (FPNC) has shown that all the territory, except about 30 m^2 area, is unaffected by contamination from activity with radiation sources in FPNC. The average ambient dose rate value was determined as 0.13 ± 0.02 μSv/h in the unaffected area. Radionuclide C-137 is the only gamma emitter, which is related to the fission of uranium and plutonium isotopes and was observed in areas unaffected by local contamination. Typical sources of Cs-137 in such areas are global fallouts (deposition of radioactive aerosols from the atmosphere after nuclear weapons tests or precipitation after accidents at nuclear power plants and devices). Typical representatives of natural radioactivity are K-40, Pb-210 (represents the U-238 decay chain), Ra-226 (represents the U-238 decay chain), and As-228 (represents the Th-232 decay chain). The result of gamma-ray dose rate scanning exceeded 0.20 μSv/h in the contaminated area on the territory of FPNC. The contaminated area is located at a distance of about 30 m from the MB of FPNC to the North direction. The contaminated area includes spots on the asphalt road, and the grass lawn as well as a storm drain well. Additional measurements of the gamma-ray dose rate and other measurements were done to define the affected area with better than 0.5 m uncertainty. The dose rate on the grass surface varied from 0.11 μSv/h to 0.22 μSv/h. This shows that the contamination on the area is non-homogeneous, there are smaller areas with no contamination inside a larger area with enhanced levels which on the grass surface can be evaluated as high as doubled background radiation level. Furthermore, the highest dose rate measured at the spot located on the asphalt road was 0.24 μSv/h. It was decided not to demolish the asphalt road in order to examine the dose rate variation beneath the asphalt in the contaminated area. Instead, the soil samples have been taken from the grass lawn where the dose rate reached 0.2 μSv/h. The comparison of ratios of specific activity of various plutonium isotopes with the results of the simulation of specific activity of plutonium isotopes in the nuclear fuel shows that contamination of the area is of the reactor origin. This finding is supported by the fact of the presence of the pipe below the contaminated area, which was used for the transportation of contaminated water from the reactors to the liquid water treatment facility. The depth profile of the contamination (an increase of specific activity of artificial radionuclides with an increase of the depth) also points out the relation of contamination of the area with the underlying pipe. Then gamma spectrum measurements at different depths of the soil and the well have been done and samples have been analysed in the laboratory using alpha, beta and gamma spectrometry. The depth of the contaminated soil exceeds 4 m. The deeper layers are more contaminated than the surface layer. Conservative activity concentration values of radionuclides Cs-137, Eu-152 and E- 154 are higher than the ECL.

For **detailed characterisation of the bulk of the contaminated area**, the installation of 11 boreholes was carried out. The gamma spectrum at various depths of a borehole was measured with a $CeBr_3$ gamma detector. Analysis of spectra shows that Cs-137 is the gamma emitter whose activity overcomes all other artificial ones in all boreholes. The results of gamma spectrometry with $CeBr_3$ scintillation detector were interpreted using simulation of Cs-137 activity concentration in a rock inside the borehole by MCNP6 code package.

Totally 24 samples from various depths in boreholes were investigated in a laboratory. The only nuclide, which activity concentration in samples exceeds a value for exemption or clearance of materials is Cs-137.

Taking into account the information about the activity concentration of rock samples, the localisation of the contaminated area to be excavated was approximately specified. The volume of the contaminated area to be excavated is about 945 m^3. The average and maximum activity of the rock to be excavated is equal to 1.96E+09 Bq and 11.0E+10 Bq, respectively.

Several options for the “zero alternative” were analysed:

- Option 1: prolonging safe enclosure of the reactors and postponing decommissioning works up to 2100;
- Option 2: leaving the reactors not dismantled forever, i.e. prolonged storage waiting until clearance levels are reached;
- Option 3: on site disposal (entombment).

Options 2 and 3 were excluded from further consideration due to a number of reasons (impossibility to meet safety requirements, non-compliance with international standards, etc.). According to the results of the preliminary analysis, the scenario of “Prolonging safe enclosure of the reactors for additional time period and postponing decommissioning works “ was chosen for a detailed review.

A selected option should be considered based on such factors as the economic component, reducing the risk to personnel and the environment during decommissioning work, compliance with the regulatory requirements.

The main advantages and disadvantages of the "zero alternative" concept are reviewed in Table 2.

Table 2 Analysis of the "zero alternative" scenario

Considered factor	Advantages	Disadvantages
Economic factor	Maintaining the facilities in working condition and monitoring the RW until 2100 will incur only operating and refurbishing costs. The costs of performing complex dismantling works and building a RW management facility can be attributed to the period after 2100.	Maintenance works will be required if the scenario with a long-term delay of work associated with the dismantling of RCs and subsequent RW management and disposal (for 60 years starting from 2039) is considered. In this case, postponing solutions for the long term will result in a significant increase of the work cost due to an increase of the money value over time (inflation, market price growth, etc.).
Radiation and non-radiation factors	Personnel exposure doses under this decommissioning option will be very insignificant, due to decay and smaller amount of RW to be transported and disposed.	RW located in the MB contain intermediate level long-lived radionuclides, the activity of which will not decrease to the clearance levels in the foreseeable future. It means that the criteria for unrestricted use of the territory of the Paldiski site are not met. RW in the Interim Storage collected in concrete and steel containers in both conditioned and unconditioned state must be regularly inspected and reconditioned if needed. Considering limited knowledge about the condition inside RCs there is a possibility for radioactive releases from the RC due to corrosion. There are lots of safety-related uncertainties.
Compliance with the regulatory requirements	-	Without solving the issue of safe waste disposal or postponing such a decision to a distant future violates the EU policy and obligations of Estonia under the Joint Convention.
Additional risk	-	The nuclear facilities can become targets of a terrorist act or military attacks of other countries. The risk of impact on the environment, personnel and the public will be significantly higher if prolonging safe

Considered factor	Advantages	Disadvantages
		enclosure of the reactors for additional time period and postponing decommissioning works is taken.

The review of the “zero alternative” concept showed that the option “Prolonging safe enclosure of the reactors for additional time period and postponing decommissioning works“ as well as other options have many disadvantages: additional costs for the operation of the MB and an increase in the cost of implementing the RC dismantling over time, the difficulties in ensuring engineering barriers long-term safety and an additional risk due to the danger of external aggression. Also, the implementation of the "zero alternative" does not comply with the EU policy and the IAEA recommendations for nuclear facility’s decommissioning options in part of not shifting the burden to future generations. Reduction of personnel doses during decommissioning and some reduction in the RW amount requiring disposal does not have a significant impact on the consideration of the “zero alternative”.

Given the many disadvantages, this decommissioning option should not be considered as an acceptable choice.

Two **decommissioning options for the FPNC RC’s** were considered:

- Under Option A **"***Dismantling and fragmentation of reactor vessels***"** RCs, including reactor equipment and structures will be dismantled and the resulting components cut into small fragments. The reactor vessel will be fragmented into small parts and disposed of in standard containers.
- Under Option B **"***Dismantling without fragmentation of reactor vessels***"** RCs, including reactor equipment and structures will be dismantled and the resulting components cut into small fragments. The reactor vessels will not be fragmented but transferred as a whole for disposal in special containers.

Both concepts require the procurement of equipment for the demolition of building structures and concrete crushing; special facilities for decontamination, fragmentation and compaction of RW. However, the Option B provides for lesser personnel doses compared to Option A, and it is thought to result in a lesser amount of waste that will be sent to disposal.

When comparing options, a formalised decision-making technique known as multi-attribute utility theory was used. According to the analysis, option B is a preferable option. It scored more points across all groups of indicators except for “Project Schedule Capability”.

The initial **Decommissioning Plan** is to describe the planned decommissioning activities of two naval training reactors that are installed at the FPNC and describes the current situation in all areas of activity.

This initial decommissioning plan should be subsequently updated to reflect information on changes of equipment or processes, unplanned events, changes in support capabilities including waste management and radiological monitoring, update of radiological conditions, changes in legislative requirements, changes in financial assumptions and improvements in decommissioning technology, etc.

The decommissioning plan should be finalised approximately three to five years before the safe enclosure phase ends. This final plan will be detailed and will be approved by the regulatory body before implementation of the final decommissioning strategy, i.e. decontamination and dismantling. This plan is the basis for the development of the detailed work instructions and procedures.

The **radiation monitoring programme** for the decommissioning of the RCs is intended for the personnel of the radiation protection department, involved in ensuring the radiation safety of employees and workplaces during the MB decommissioning, and FPNC management body. Taking into account the modern practice of performing similar works, recommendations on the values of the investigation level during the first year of the dismantling work for individual personnel control every month are provided. Exposure control equipment for employees and workplaces and measured values are described. Exposure levels and measures to be taken when exposure levels are exceeded are detailed. Measured values, measuring equipment, and measurement frequency at the territory of FPNC are provided.

Environmental radioactivity monitoring must begin before the start of the decommissioning of the RCs and continue until the end of the activity. Monitoring results should be stored until the change in radiological status of FPNC.

The environmental radioactivity monitoring should cover the normal operation of the decommissioning of the RCs as well as emergencies. Periodic revisions of the programme are recommended taking into account the available results of monitoring.

Installation and operation of monitoring system should not create additional pathways of radionuclide spread. If a disposal facility is established on FPNC, the same monitoring system could serve both the disposal facility and decommissioning of the reactor. Therefore, in this case system optimisation is possible.

The **safety case** was performed in accordance with the recommendations of IAEA. Relevant hazards were quantified and their possible consequences for workers and the public identified. For the analysis of accidents during decommissioning, the deterministic approach to safety assessment was used.

When dismantling the RC No. 1, in order to fulfil the condition of not exceeding the dose limit for the category A personnel, the following number of workers are required for individual stages:

- 2 workers are needed for dismantling equipment, containers filled with ionised radiation sources and RW (rags, metal waste, tools, etc.) embedded in concrete at the end of the lower room of the instrumental enclosure;
- 13 workers are needed for dismantling of the reactor vessel.

When dismantling of the protective tank metal structures of the RC No. 2, a minimum of four workers are required to meet the condition of not exceeding the dose limit for the category A personnel.

The maximum dose rate for workers handling packaged RW will not exceed 2 mSv/h at the workplace.

Under normal work conditions radiation exposure of the population and the environment can occur only due to gas-aerosol emissions into the atmosphere since there are no sources of any significant water discharges during work. The total release activity under normal conditions of dismantling work will not exceed 1.1E+04 Bq. The comparison of the release levels leads to conclude that the magnitude of the radioactive release, under normal work conditions, will not exceed the release levels specified in the Regulation No. 40 “Conditions for Exclusion and Release of Radioactive Substances Used or Generated in Radiation Operations and Requirements for Requests for Exclusion and Release”.

Assessment of the impact of radioactive factors on the public and the environment under scenarios of emergencies taking into account a conservative approach demonstrated that the exposure dose of the public as a result of radioactive contamination of the territory will not exceed 1.0 mSv/year. Therefore, the survey of radiological conditions including all types of radiation and optimisation of measures aimed at radiation protection at the territories considered is not required.

The analysis of normal decommissioning activities and abnormal events and incidents shows that the radiation safety criteria are not exceeded:

- **THE VALUES OF THE LIMITS OF DOSES FOR PERSONNEL AND THE PUBLIC AS PER RADIATION ACT AND ESTONIAN GOVERNMENT DECREE NO. 97 ARE NOT EXCEEDED, BOTH UNDER NORMAL CONDITIONS AND IN ACCIDENTS;**
- **THERE IS NO NEED TO APPLY ACTION LEVELS: EVACUATION, RESETTLEMENT, AS SPECIFIED IN THE ESTONIAN GOVERNMENT DECREE NO. 95.**

To exclude uncertainties in estimates associated with possible changes in regulatory requirements, conditions for the implementation of on-site activities taking into account the knowledge gained as a result of other similar work at other facilities or sites (including international experience) and considering the characteristics of equipment to be selected through the design, the SAR should be revised after the development of the final decommissioning design for the FPNC.

Possible impact analysis of the decommissioning of RCs on neighbouring countries in accordance with of International Conventions and Treaties was also performed. Simulation of the transboundary transport of radionuclides released in the event of radiation accidents at the FPNC showed that the total effective doses of public exposure during the acute period of the accident and during the first year after it for Finland (at the reference point in Helsinki) will be significantly lower than the established limit of the individual effective dose 1 mSv*year^{-1}.

Calculations of radioactive contamination of the air, the earth's surface, the marine environment and the corresponding doses to the population as a result of the transboundary transport of accidental radioactive releases from the FPNC showed no significant negative effects on the environment and public health. The highest dose that was obtained for the specific meteorological conditions with precipitations accompanied the transfer of radioactive cloud over the Gulf of Finland in a hypothetical beyond design basis accident had the value of 0.01 µSv for 1st year after the accident.

Risk analysis and assessment for decommissioning of RCs at FPNC were performed using sixteen emergency scenarios based on emergency scenarios provided by SAR, including six internal scenarios and ten external scenarios. Most of the scenarios have negligible probabilities and severities, sometimes several orders of magnitude below the criteria for "very low" probability and "insignificant" severity provided by the Regulation No. 28 "Requirements for an emergency risk assessment and procedure for the preparation of a risk assessment".

There are no internal or external emergency scenarios of significant, high or very high-risk categories.

Four scenarios, including three internal and one external emergency scenario, belong to the "average" ("yellow") risk category. Twelve scenarios, including three internal and nine external events scenarios, belong to the "low" ("green") risk category.

Two internal events scenarios belonging to the "average" risk category, namely „Liquid radioactive waste spill" and "Exposing ionising radiation source", have "high" probabilities conditioned by human errors, although their severities were evaluated as "insignificant". Preventive measures for these scenarios shall be aimed at reducing human error probability. One internal event scenario belonging to the "average" risk category, namely "Terrorist attack" has "low" probability combined with "minor" severity. Another three internal events scenarios, namely "Dropping the reactor vessel", "Fire in the MB" and "Internal power loss" have very low probabilities and insignificant severities.

One external events scenario belongs to the "average" risk category, namely "External wind". The other nine external events scenarios have very low, low or average probabilities and insignificant severities and belong to the "low" ("green") risk category.

Risk preventive measures were suggested for all considered scenarios. Implementation priority might be given to risk preventive measures for the higher-risk category.