2020:05

General data in accordance with the requirements in Article 37 of the Euratom Treaty

Decommissioning of the nuclear reactors Ringhals 1 and Ringhals 2 in Sweden

Report number: 2020:05 ISSN: 2000-0456
Available at www.stralsakerhetsmyndigheten.se

Abstract

The purpose of this report is to provide the European Commission with general data relating to the decommissioning of the two nuclear reactors, Ringhals 1 (R1) and Ringhals 2 (R2), at the Ringhals Nuclear Power Plant (NPP) in Sweden. This data, along with calculations, will make it possible to determine whether the implementation of the decommissioning plans is liable to result in the radioactive contamination of the water, soil or airspace of another European Union member state. The report follows the guideline in Annex III of the recommendation of the application of Article 37 of the Euratom Treaty (2010/635/Euratom).

The Ringhals NPP comprises four reactors in total. Ringhals 1 is a boiling water reactor that started commercial operation in 1976. Ringhals 2 is a pressurized water reactor that commenced operation in 1975. The final shut down of Ringhals 1 is planned in December 2020. Ringhals 2 was shut down in December 2019.

Using conservative assessment methods and assumptions (e.g. ignoring filtration), the effective dose in the vicinity of the plant is demonstrated to be less than 0.01 millisievert (mSv) per year at the peak radioactive release year during normal operation (Chapters 3 and 4), and below 0.5 mSv during worst case unplanned release scenarios (Chapter 6). Based on this local dose it is concluded that the decommissioning project poses no risk to any other member state.

This report has been produced by the Swedish Radiation Safety Authority, SSM, mainly based on information provided by the licence holder, Ringhals AB. SSM has checked that the general data provides the necessary information and that it complies with the guideline of the most recent recommendations of the application of Article 37 of the Euratom Treaty

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Swedish Radiation Safety Authority

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Content

0. Introduction

0.1. Purpose

Article 37 of the Euratom Treaty (2010/635/Euratom) has as its objective to forestall any possibility of radioactive contamination of another member State.

The purpose of this report is to provide the European Commission with general data relating to the decommissioning of the two nuclear reactors, Ringhals 1 (R1) and Ringhals 2 (R2), at the Ringhals NPP. This data, along with calculations based on it, will make it possible to determine whether the implementation of the decommissioning plans is liable to result in the radioactive contamination of the water, soil or airspace of another European Union member state. The report follows the guideline in annex III of the recommendation of the application of Article 37 of the Euratom Treaty (2010/635/Euratom) [1].

In order to commence dismantling and demolition of NPPs in Sweden, approval is needed from the Swedish regulatory authority SSM (Swedish Radiation Safety Authority) and from the Land and Environmental Court in Sweden.

The opinion of The European Commission on the general data, in accordance with Article 37 of the Euratom Treaty (2010/635/Euratom), is required to be taken into account before SSM can grant approval of dismantling and demolition activities.

The report has been produced by SSM, mainly based on information provided by the licence holder, Ringhals AB. The information provided by Ringhals AB included the evaluation of the effective doses. SSM has checked that the general data provides the necessary information and that it complies with the guideline of the most recent recommendations of the application of Article 37 of the Euratom Treaty.

0.2. Ringhals Nuclear Power Plant

Ringhals NPP is located in the south of Sweden, on the west coast approximately 50 km south of Gothenburg and about 20 km north of the town of Varberg. The licence holder is Ringhals AB.

The Ringhals Nuclear Power Plant comprises four reactors in total. Ringhals 1 is a boiling water reactor designed and constructed by ASEA ATOM. Commercial operation of Ringhals 1 (2540 MWt) started in 1976. The three pressurized water reactors (R2, R3 and R4) were designed and constructed by Westinghouse. Ringhals 2 (2652 MWt) commenced operation in 1975 and Ringhals 3 and 4 (3135 and 3300 MWt) commenced operation in 1981 and 1983, respectively. Ringhals 3 and 4 are planned to continue operation until the 2040s.

At the time of this report being issued, Ringhals 1 is still in operation and will be until the final shut down in 2020. Ringhals 2 was shut down in December 2019.

The spent fuel removal is performed on a regular basis at all Swedish NPPs. The spent fuel will be removed off-site to the interim storage at the Central Interim Storage Facility for Spent Nuclear Fuel (Clab), owned and operated by the Swedish Nuclear Fuel and

Waste Management Company (SKB) before the dismantling activities start. Clab is the facility used by all NPPs in Sweden for storage of spent fuel and it is located in the south of Sweden, on the east coast. An overview of the Swedish system for handling of spent fuel and nuclear waste is described in appendix 1.

The general data presented in this report cover the decommissioning of Ringhals 1 and 2 and possible pathways for release to the surrounding environment both from planned decommissioning activities and from possible unplanned releases.

0.3. Decommissioning licensing

This section gives a brief overview of the licensing process. R1 and R2 have different shutdown dates and two sets of applications are required. Each set consists of: an environmental license from the Land and Environment Court in Sweden, and an approval of the decommissioning safety analysis report (SAR) from SSM.

Licensing of facilities for activities involving radiation in Sweden is governed by the provisions of the following laws:

- The Environmental Code (1998:808),
- The Act (1984:3) on Nuclear Activities
- The Radiation Protection Act (2018:396)

A nuclear facility requires a license under the Environmental Code (1998:808) and a license under the Act on Nuclear Activities (1984:3). Also, the licensee is required to have permission for operations with ionizing radiation under the Radiation Protection Act (2018:396).

A specific authorization for decommissioning is needed from the Land and Environmental Court in Sweden according to the Environmental Code.

Decommissioning activities are regarded as obligations in the Act on Nuclear Activities that are included in the license for operation of the facility. Therefore, no new license is needed according to the Act on Nuclear Activities. Instead, before commencing the dismantling and decommissioning phase, an application including an updated safety analysis report shall be reviewed and approved by the Swedish Radiation Safety Authority, SSM.

In this report, the following notions are used for stages of operation and decommissioning;

Power operation phase: This is the phase where power production occurs. At the time of this report being issued R1 is in this phase.

Post-operation phase: The post-operation phase is the phase when power production has ceased, but with fuel still present at the plant. Due to the fuel still being present, the safety measures of the facility are essentially the same as during power production. R2 entered this phase at the end of 2019.

Care and maintenance phase: The reactor enters the care and maintenance phase when the fuel has been removed from the facility, but before any dismantling and demolition phase activities have begun.

Dismantling and demolition phase: This phase contains the dismantling and demolition activities.

The licensing process is illustrated in Figure 0-1.

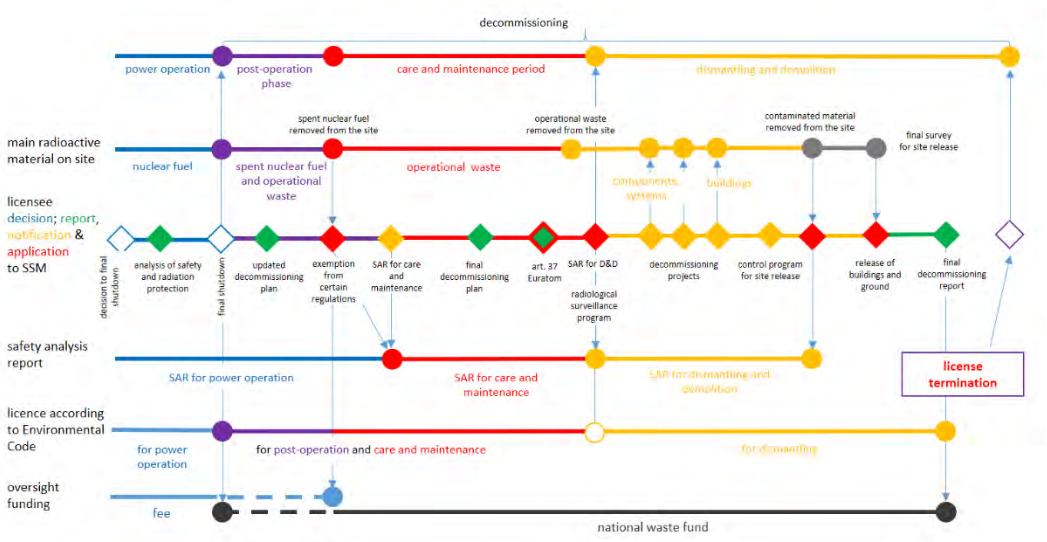

Figure 0-1. The decommissioning process including the procedures for approval and licensing (Source: SSM).

0.3.1. Radiation safety licensing for decommissioning

SSM reports to the Ministry of the Environment and has a regulatory mandate from the Swedish Government within the areas of nuclear safety, radiation protection and nuclear non-proliferation.

The authority works proactively and preventively to protect people and the environment from the undesirable effects of radiation. The Act on Nuclear Activities 1984:3 and the Ordinance 2008:452 with instructions for the SSM have been translated into English and can be found on the SSM website (www.ssm.se). The Radiation Protection Act 2018:396 is not yet published in English. These acts and ordinance, together with the Ordinance on Nuclear Activities 1984:14 and the Radiation Protection Ordinance 2018:506, stipulate the boundaries for all nuclear activities in Sweden. SSM has developed regulations (SSMFS) to give a more detailed framework for, among others, nuclear power plants.

In order to start decommissioning and dismantling, a safety analysis report (SAR) must be approved by SSM. The requirement to have an approved SAR is found in SSMFS 2008:1. The regulations in SSMFS 2008:1 apply to measures required to maintain safety in connection with the construction, possession and operation of nuclear facilities with the aim of, as far as reasonably achievable, taking into account the best available technology, preventing radiological accidents and preventing the unlawful handling of nuclear material and nuclear waste. The regulations comprise provisions on technical, organizational and administrative measures. Chapter 9 in SSMFS 2008:1 stipulates the requirements for decommissioning and dismantling. To complement and specify the regulatory framework for decommissioning, SSM has also issued specific license conditions for decommissioning, [2].

Fundamental requirements for operations with ionizing radiation are presented in SSMFS 2018:1. In SSMFS 2018:3 requirements with respect to clearance and associated criteria are presented. Requirements regarding releases to the environment and radiation protection are presented in SSMFS 2008:23, SSMFS 2008:26 and in the specific license conditions for decommissioning, [2].

The decommissioning process is described in a decommissioning plan. A nuclear power plant with more than one reactor, is also required to have a decommissioning strategy, describing the decommissioning process for the entire NPP site. The decommissioning strategy and plan are required by SSM, together with a waste management plan. These reports shall supplement the safety analysis report (SAR) in the application to enter the dismantling and demolition phase. The safety analysis report shall be reviewed and approved by the Swedish Radiation Safety Authority (SSM) in order to grant an approval for dismantling and demolition activities.

After approval of the safety analysis report (SAR), each dismantling and/demolition package/project of parts (components, systems or building) of the plant that contains contaminated or activated systems must be notified to SSM. SSM conducts regular oversight activities, using a graded approach principle during the decommissioning and dismantling process. The site will only be released from regulatory control from the Act on Nuclear Activities and the Radiation Protection Act by Governmental decision, upon the recommendation of SSM once the final state report is approved. The final state report comprises a summary of the decommissioning and a foundation for the decision on completed obligations to the laws governing radiation safety.

SSM will not allow dismantling and demolition activities to occur until the European Commission has provided an opinion on the article 37 report for the decommissioning project. The purpose of this is to enable the Commission to give its opinion whether the implementation of the decommissioning plan is likely to involve radiological consequences in another member State.

This document and any requested references are part of the submission.

0.3.2. Environmental licensing

The operator must apply for a license to decommission an NPP in accordance with the Swedish Environmental Code. The Swedish Environmental Protection Agency, the County Administrative Board, the local Environmental and Public Health Committee and SSM are consulted in the licensing procedure and are given the opportunity to propose specific licensing conditions. The environmental license is issued by the Land and Environment Court.

The environmental impact assessment, which is the main submittal of the environmental licensing process, for the R1 and R2 decommissioning project is submitted in two steps;

a) One application applies to the post-operational phase (and any potential care and maintenance phase) of both R1 and R2. This license has been granted by the Land and Environmental Court in July 2018.
b) A second application has been submitted in 2018, which applies to the dismantling and demolition phase of both R1 and R2.

0.4. Decommissioning funding

According to Swedish law it is the owners of the NPPs that must pay for all the costs of dealing with spent nuclear fuel and its final disposal. They also must pay the costs of decommissioning the NPPs and other nuclear installations. Since the mid-1970s the nuclear power operators have been allocating funds to cover these costs. These funds are administered by the Nuclear Waste Fund. The Swedish Nuclear Fuel and Waste Management Company, SKB, performs regular calculations of the future costs for dealing with nuclear waste and submits the updated budget to SSM every three years. After the calculations have been reviewed, the Swedish National Debt Office then proposes the surcharge for the next few years to the Government, which decides on the amount to be charged.

The funds are distributed to the decommissioning projects as they progress.

0.5. Decommissioning planning

This section gives an overview of the decommissioning planning. In Figure 0-2 the main phases, operation, post operation (defueling) and dismantling and demolition (D&D), of the decommissioning process are shown for Ringhals 1 and 2.

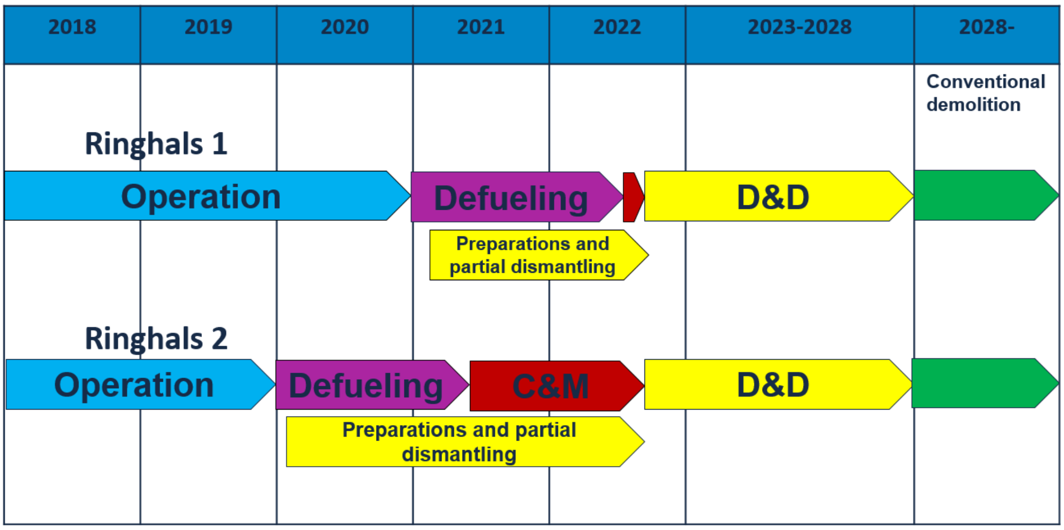

Figure 0-2. Overview of the decommissioning phases, operation, post operation (defueling), Care and Maintenance (C&M) and dismantling and demolition (D&D), for Ringhals 1 and 2.

0.5.1. Decommissioning of R1 and R2

Licensees of nuclear reactors are responsible for identifying and estimating all measures that are needed to manage and dispose of spent nuclear fuel, as well as other radioactive residual products from nuclear activities, and to decommission and dismantle the reactor plants.

The decommissioning plan comprises the general Swedish system for the disposal of spent nuclear fuel and radioactive waste and the specific decommissioning activities at Ringhals 1 and 2.

The project planning phase for decommissioning and dismantling of Ringhals 1 and 2 began in 2014 and is still ongoing. This phase includes site characterization, general waste management and D&D planning based on prerequisites for the site, logistics and the criteria and plans for the final depositories. The licensing process including environmental and safety assessments to receive authorization for the decommissioning and demolition of the facilities is an important part of this phase.

The plan is to dismantle Ringhals 1 and 2 in parallel. The final shutdown of Ringhals 2 happened at the end of 2019. The final shutdown of Ringhals 1 is expected one year later, at the end of 2020. During the post-operational phase, the fuel will be stored in the fuel pools in the reactor building at Ringhals 1 and the fuel building at Ringhals 2. The post-operational phase continues until approximately mid 2022 when all fuel from both Ringhals 1 and 2 has been removed and transported to the intermediate storage facility CLAB. During this time general preparations are made before the start of the full-scale dismantling and demolition phase, see section 0.5.2. The dismantling and demolition phase starts after the fuel is transported offsite, i.e. there is no intervening period for planned care and maintenance.

The dismantling and demolition phase with its dismantling and waste management of contaminated systems and components will be carried out approximately between 2022 and 2026. Decontamination and measurements of buildings according to a free release control program will be carried out before the conventional demolition will commence in 2028. Decontamination of other areas, soil and ground etc. will also be commenced in 2026. The free release of the site will wait until the decommissioning of Ringhals 3 and 4.

The current waste management infrastructure will be complemented with local waste management stations and additional equipment for decontamination, compaction, clearance measurements etc. to handle the increased volumes of waste. A site survey has been completed to complement the plant radiological and inventory characterization. Based on these studies, the waste handling and clearance processes can be integrated into the D&D-planning. The disposal of radioactive waste and the release of materials from radiation regulations through clearance is described in chapter 5 of this report.

Waste handling adaptations are needed to handle special items such as the reactor internals, the primary systems and the steam generators. The waste management of special items will be based on experience from previous similar projects at Ringhals.

The most active parts such as the internals of the primary system are planned to be segmented and removed early to reduce the radiation levels and facilitate the dismantling of the remaining systems. The planned sequence of activities is given in Table 0-1. Note that the stated year is the initiation year for the activity, and that an activity may last several years.

The decommissioning sequence will be developed further during the subsequent project phases, with respect to radiological characterization after defueling and decontamination and based on the experience of dismantling contractors.

Table 0-1. Planned decommissioning activities at R1 and R2.

Initiation year	Planned activity	
	Ringhals 1 (BWR)	Ringhals 2 (PWR)
2020		Shutdown, Preparation, Chemical decontamination
2021	Shutdown, Preparation, Chemical decontamination	
2022	Dismantling of reactor internal components, Dismantling of turbine components, Dismantling of cooling systems, parts of the waste systems, chemical systems	Dismantling of reactor internal components[1], Dismantling of fuel pool and associated cooling systems, Dismantling of steam generators, Dismantling of cooling systems and chemical volume control system
2024	Dismantling of reactor pressure vessel and primary systems	Dismantling of reactor pressure vessel and primary systems
2025	Dismantling of biological shield, Dismantling of remaining systems[2].	Dismantling of primary shield, Dismantling of remaining systems

Ringhals 1 and 2 are planned to be released for conventional demolition by 2028.

0.5.2. Early decommissioning activities

After the final shutdown of the plant, general preparations will be made for the start of the full-scale dismantling and demolition.

During the post-operation and care and maintenance phase, activities that are necessary to ensure that the decommissioning does not affect the continued operation at Ringhals 3 and 4 will be performed. These activities include separating the site infrastructure and support systems. Other preparatory activities that are planned are removal of combustible materials such as oil and hydrogen, adaptations of lifting equipment as well as isolation and drainage of systems that are not necessary during decommissioning.

The Environmental license for the post-operation and care and maintenance phase includes the option of limited partial dismantling and demolition activities. This option primarily relates to dismantling equipment in the turbine hall for Ringhals 1 and segmenting reactor internals at Ringhals 2. In addition, removal of insulation, asbestos and other hazardous materials will also facilitate the subsequent dismantling activities.

[1] Note that the dismantling of reactor internals for Ringhals 2 is limiting with respect to the releases of radioactive effluents during normal conditions. In order to maximize the activity inventory in the reactor internals and thus the calculated theoretical release to the environment (see chapters 3 and 4), it is assumed that the reactor internals are dismantled during the first year after final shutdown. The actual start of dismantling of reactor internals for Ringhals 2 is planned during 2022.

[2] This includes the dismantling of systems in the waste handling building, which is considered as an option If it is not needed to handle operational waste for Ringhals 3 and 4.

Partial dismantling activities are also subject to the licensing procedure described in section 0.3.

1. The site and its surroundings

1.1. Geographical, topographical and geological features of the site and region

Ringhals NPP is located on the Swedish west coast at coordinates 57° 15′ 35″ N, 12° 6′ 39″ E.

A map of the location and its relation to other countries is presented in Figure 1-1.

Figure 1-1. Location of Ringhals NPP (Source Kartdata, GeoBasis-DE/BKG, Google, 2019).

In Table 1-1 the distances to the borders as well as to metropolitan areas in surrounding EU member states are given. The closest EU member state is Denmark, with the closest (sparsely) populated area, the island Læsø, at a distance of about 50 km across the Kattegat from Ringhals.

Table 1-1. Distance to EU member states and metropolitan areas[3]

Country	Approximate distance to border (km)	Metropolitan area	Population (million)	Approximate distance to metropolitan area (km)
Denmark	50	Aalborg	0.2	150
Germany	300	Hamburg	5.1	400
Poland	400	Szczecin	0.8	450
Lithuania	550	Klaipeda	0.2	600
Latvia	550	Riga	1.0	700
Estonia	600	Tallinn	0.6	800
Finland	550	Turku	0.3	700
Netherlands	550	Groningen	0.3	550

In Figure 1-2 a map of the vicinity of Ringhals is presented. Near the plant there are scattered houses and small communities at a distance of a few hundred meters. The closest villages with a larger population (~1000) are Bua and Väröbacka on the order of a few km. The closest town with a significant population (~30 000) is Varberg, approximately 20 km south of the plant. The closest metropolitan area is Göteborg (see Figure 1-1) 50 km north of the plant, with a population of approximately half a million.

[3] Distances have been rounded with 50 km increments. Metropolitan area is defined here as a city with more than 200 000 inhabitants.

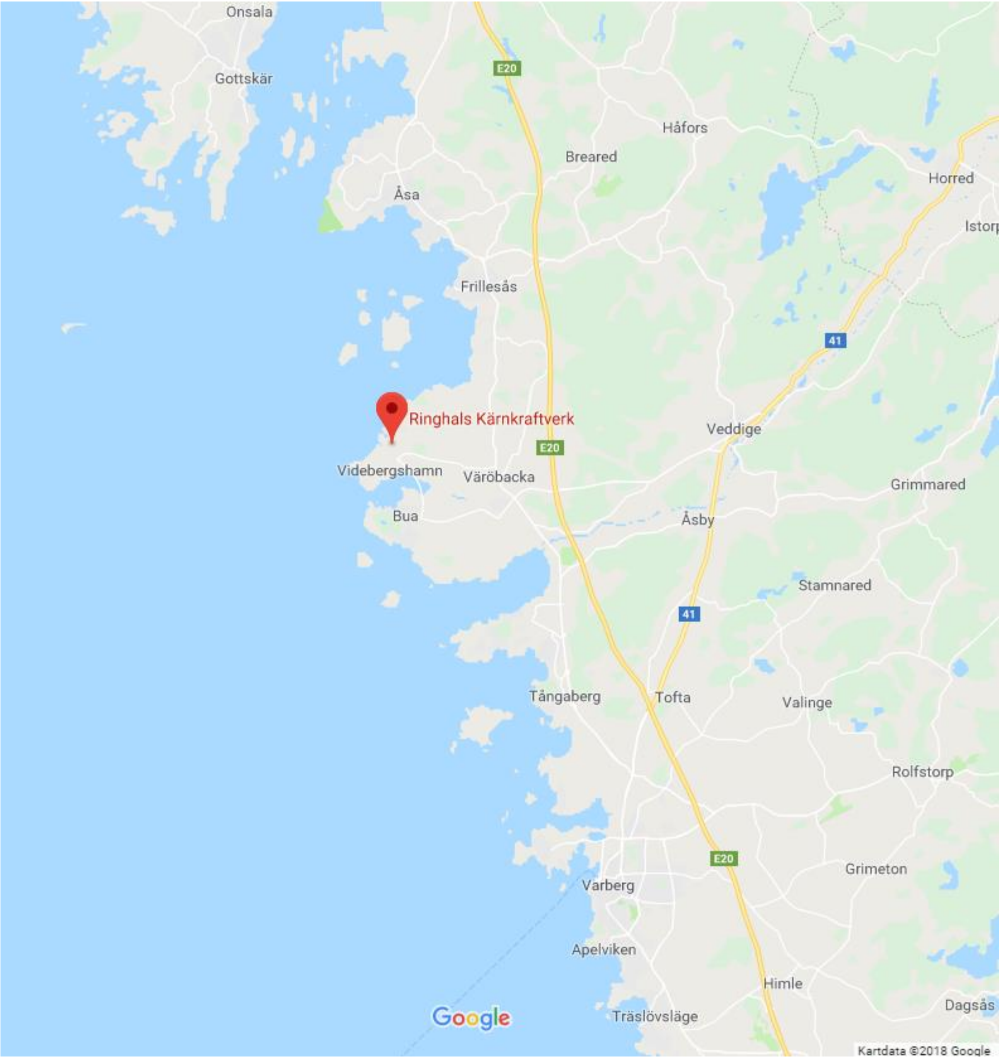

Figure 1-2. Map of the vicinity of Ringhals (source Kartdata, Google, 2018).

In Figure 1-3 a land use map of the vicinity of Ringhals is presented. In this map the populated, forested and barren areas are illustrated together with agricultural land.

Figure 1-3. Land use map of the vicinity of Ringhals (red dot). (Green: Grass/forest, Yellow: Agricultural, Brown: Populated, Beige: Barren/sparsely vegetated) (Source Lantmäteriet, 2018).

The topography of the area is varied but in general low. The highest points in the vicinity are approximately 50 m above sea level on local hills.

Figure 1-4 shows an aerial photograph of the Ringhals NPP. The industrial area covers approximately 2.5 km2 of land area. Apart from the four reactors and buildings housing shared systems, the site also contains office buildings, conventional workshops and storage buildings. There is also a hotel, an information center and a conference Centre near the site area.

On the photograph Ringhals units 1, 2, 3 and 4 as well as the operational waste surface repository are marked.

R3 and R4 will continue to produce electricity during the dismantling of R1 and R2. As such, they will be the main source of radioactive release in the area through continued operational release. For this reason, they are included in the effluent release calculations presented in chapters 3 and 4.

Apart from the operating reactors and the waste management and storage facilities, the very low-level operational waste surface repository is the only other facility in the area that could potentially release radioactive effluents. In this surface repository metallic and compressed organic waste (plastics, cloth etc.) from operation of the four units is disposed of in intermittent disposal campaigns. The surface repository has been designed to not allow significant infiltration of precipitation, and hence no leachate has been possible to detect during the two decades it has been in operation.

The operational waste surface repository is not licensed to accept decommissioning waste. As is discussed later in this report, a corresponding surface repository for decommissioning waste is currently in the early stages of planning. The location is planned to be either southwest of R3, near the operational waste surface repository, or northeast of R1.

There are no other facilities or activities in the area which have any significant impact on the radiological safety of the decommissioning project.

Figure 1-4. Aerial photograph of Ringhals NPP (Source Kartdata, Google, 2018).

1.1.1. Geology of the site and region

In Figure 1-5 and Figure 1-6 the soil types and bedrock types, respectively, of the surrounding area are given. The latter also shows deformation zones in the area.

Most of the coastline consists of exposed granite eroded to a smooth surface.

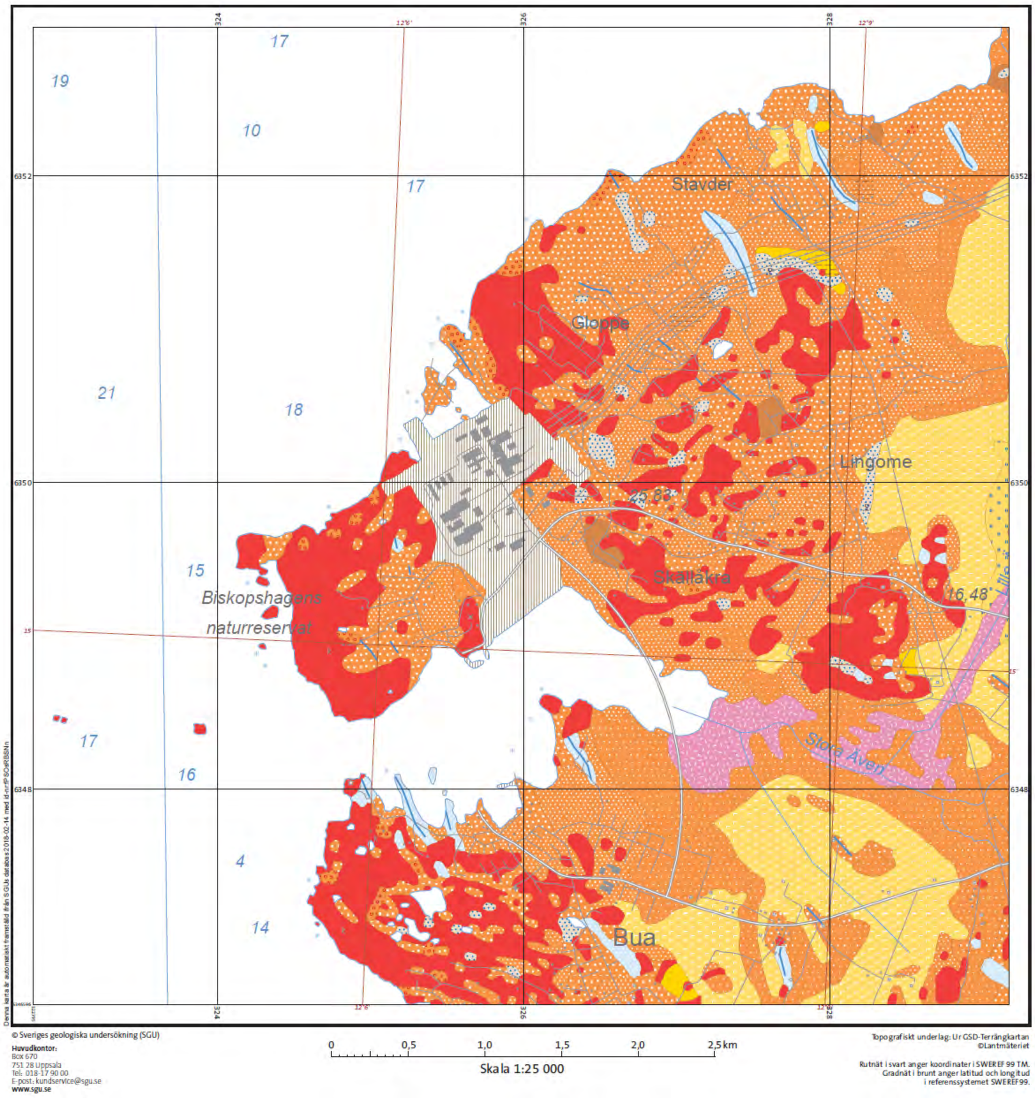

Figure 1-5.Soil types in the area surrounding Ringhals[4] . Red is bedrock, orange is post-glacial sand and gravel, purple is fluvial clay silt, and yellow is clay. (Source SGU, retrieved 2018, map based on data from 1981).

[4] Note that the source map used in the SGU map service is from a date before some extension projects, which means that the extent of the Ringhals area is not entirely up to date. This does not affect the general data in the map.

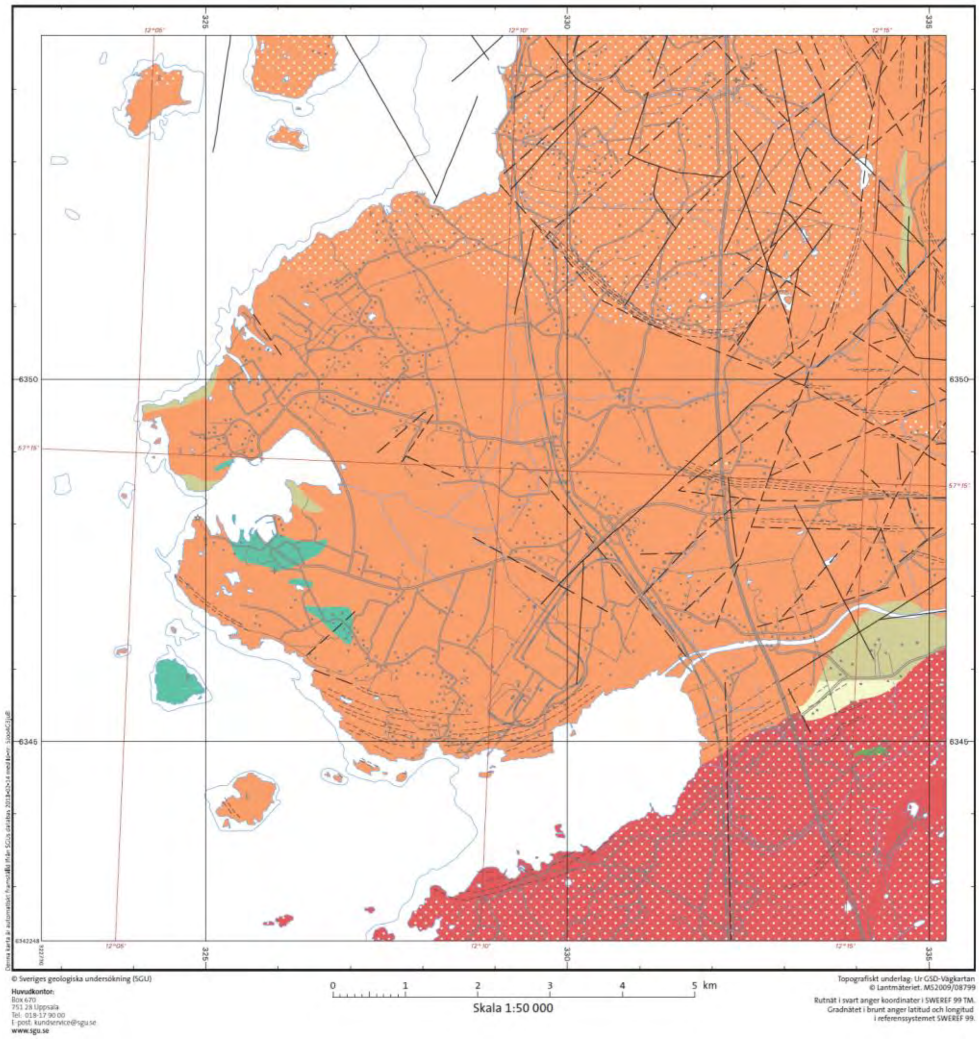

Figure 1-6. Bedrock types of the area surrounding Ringhals. Orange and red are acidic intrusive rock (granite, monzonite etc.) while green and turquoise are basic or intermediate intrusive rock (gabbro, diorite, diabase etc.). Black lines and dashed lines show deformation zones (Source SGU, retrieved 2018, map based on data mainly from 2016-2018).

No earthquake large enough to jeopardize the safety of a nuclear facility has been documented in Sweden in recorded history. Smaller earthquakes do, however, occur. According to [3] eight earthquakes with a magnitude 2, three with magnitude 3 and five with magnitude 4 according to the Richter's magnitude scale have been measured in Sweden since the year 2000.

Ringhals NPP is located on bedrock with several deformation zones. These have been determined to be open, and due to their location behind the Göta älv-deformation zone it is unlikely that any large pressure or tension can build up with an earthquake as consequence in the area. The presence of unperturbed deformation zones with an age of more than 100 million years in the vicinity of the Ringhals site confirms that no significant movements occur in the region [4].

1.2. Hydrology

The Ringhals site is adjacent to the Kattegat, which is a body of water between Sweden and Denmark. The Kattegat is also where Ringhals collects and discharges its cooling water.

In Figure 1-7 the depth in the area is shown together with the cooling water discharge point.

The water flow in the region is determined by wind as well as the larger flow patterns of the Kattegat. Brackish water from the Baltic Sea flows into the Kattegat as a surface layer over the saltier water in the Kattegat. This creates a surface flow, called the Baltic stream, which follows the Swedish coast north toward Norway. The flow patterns are also affected by other meteorological effects such as air pressure changes, salt level variations etc., which can result in southbound streams. For this reason, the surface flow can switch, often more than once per day. Flow measurements have shown that the flow is northbound approximately 40-50% of the time, while it is southbound 30-40% of the time. The flow speed is on average a few dm per second. In deeper waters, below 20 m depth, southbound flow is dominant [5].

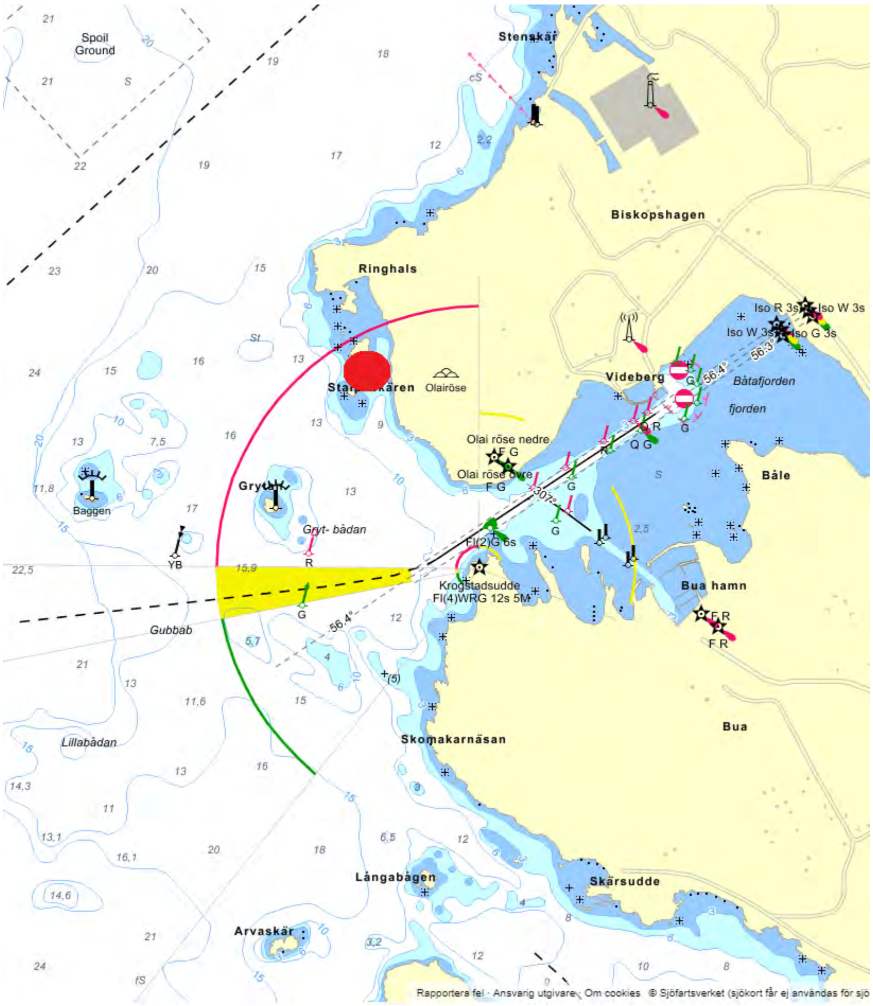

Figure 1-7. Depth in the area. The cooling water discharge points (separate for R1-R2 and R3-R4) is shown as a red dot. (The figure is from a navigational chart and contains additional information not of relevance to this report) (Source Eniro, Sjöfartsverket, 2018).

In Figure 1-8, the marine geology of the area (the dominant material in the top one meter of the sea bottom) is shown. As can be seen in this figure, most of the sea bottom is dominated by clay, sand and gravel.

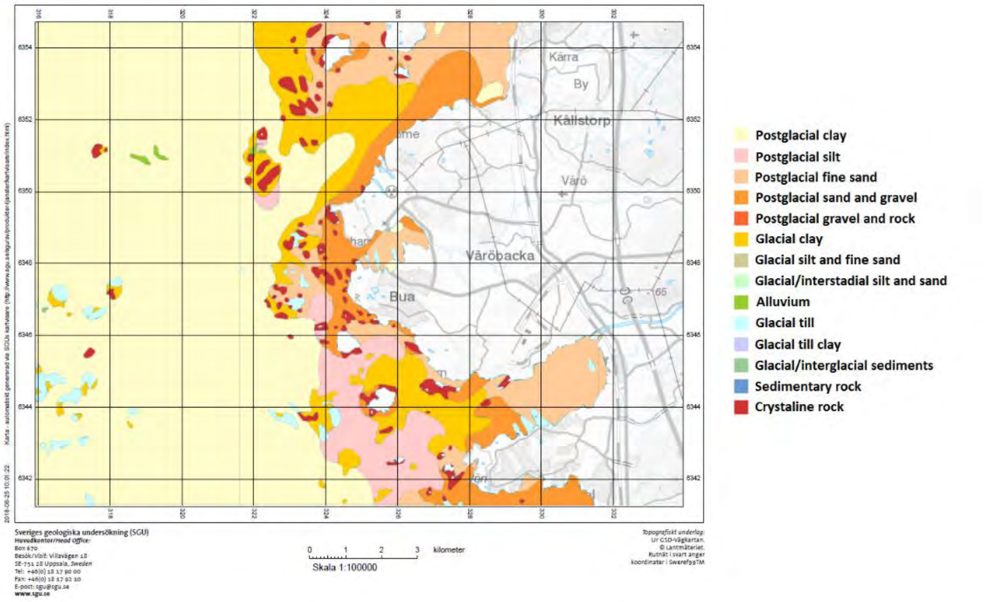

Figure 1-8. Marine geology (the dominant material in the top one meter of the sea bottom) in the vicinity of Ringhals. (Source SGU, retrieved 2018, map based on data from 2002).

The sea water level at Ringhals is measured automatically by the Swedish meteorological institute, SMHI. Hourly water levels are publicly available and shows a great variation. The highest measured value was 168.5 cm in January 2005, and the lowest -98.7 cm in January 1976. Both measurements are according to the Swedish RH 2000 reference system, which uses the Normaal Amsterdams Peil (NAP) as reference level. The reactor ground levels correspond approximately to 310 cm in the RH 2000 system.

Tidal levels in the region vary by at most about 5 cm but can vary up to approximately 20 cm with spring flooding. Taking wind and pressure into account the sea level can further increase to approximately 30 cm [6].

In [6] modelled results of the effects of sea level rise along the Swedish coast are presented. The model is mainly based on altitude, which means that any low-lying area under a given altitude will be considered to be flooded, even though local effects might mean that this would not be the case in reality.

In Figure 1-9 and Figure 1-10 the expected flooding due to a postulated sea level increase of 2 and 3 m respectively are presented. As can be seen, an extreme flooding (an addition of 130 cm to the maximum recorded height during the 2005 storm) is required in order to reach flooding events where a radiological release might occur (flooding of the waste management areas). This means that the Ringhals site is well protected against flooding.

Figure 1-9. Flooding at a postulated sea level rise of 200 cm (RH 2000). The picture is a slightly edited version of the original in [6] as the R1 reactor building has been marked for reference.

Figure 1-10. Flooding at a postulated sea level rise of 300 cm (RH 2000). The picture is a slightly edited version of the original in [6] as the R1 and R2 reactor buildings have been marked for reference.

The groundwater conditions vary depending on soil types, topography, season, climate and weather conditions. The runoff (or the total flow of water) in the area is calculated annually by SMHI using the S-HYPE model for Sweden, which takes into account measured temperature and precipitation levels in the country. In Figure 1-11 the specific runoff, i.e., water amount per unit area, for 2012 is shown for the area.

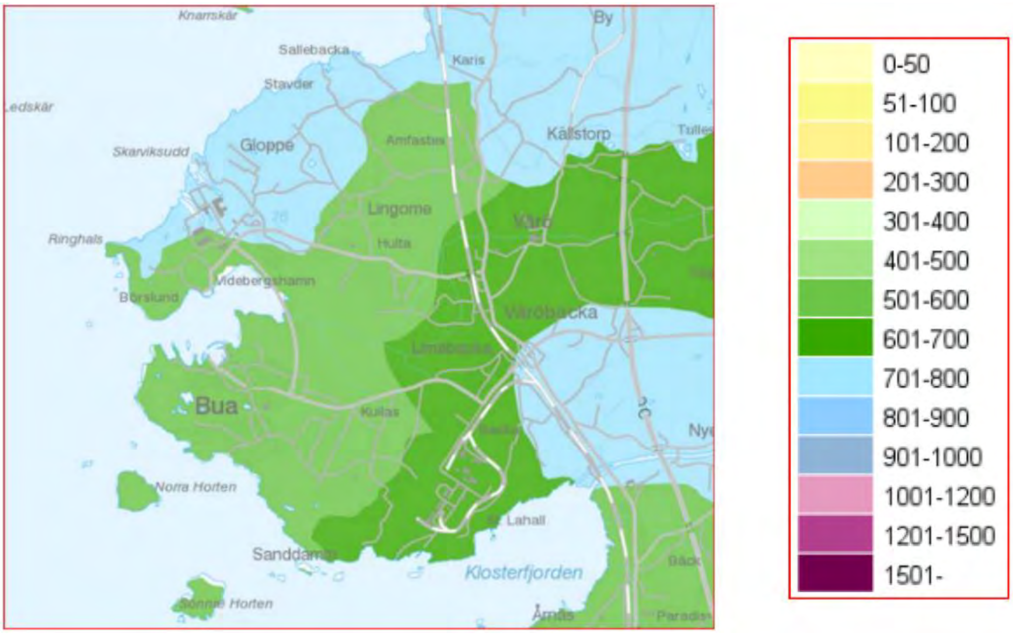

Figure 1-11. Specific runoff in mm for the area during 2012, which was a high runoff year [6].

1.3. Meteorology

1.3.1. Wind

In [6] data from 2007 – 2017 at two meteorological stations have been gathered. These stations are located on the island of Nidingen (13 km north west of Ringhals) and in Varberg (20 km south east of Ringhals), which have been judged to be representative of Ringhals.

In Figure 1-12 a wind rose showing the relative occurrence of wind direction in this dataset is shown. South-westerly to westerly winds dominate over the year.

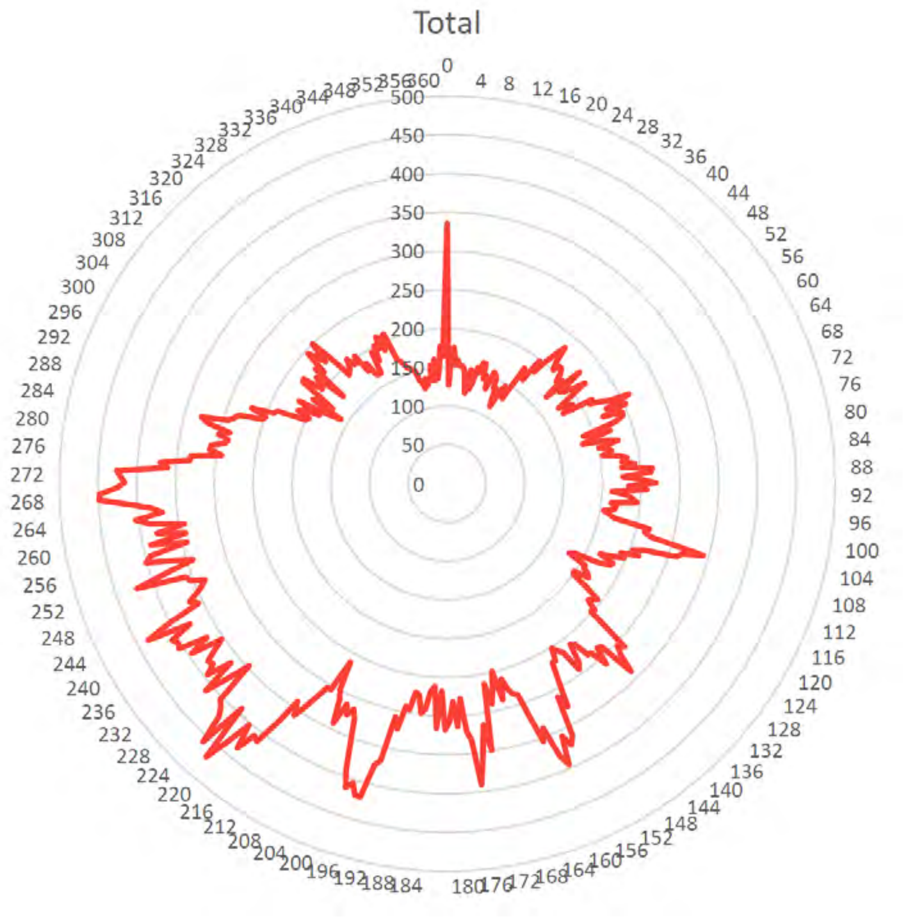

Figure 1-12. Wind rose displaying the direction of wind at the Nidingen measurement station for the period 2007-2017. 0 degrees (up) is north, 90 degrees (right) is east etc. The direction of each sector shows the wind direction while the amplitude shows the relative occurrence of wind from that direction.

In Figure 1-13 and Figure 1-14 the average and maximum annual wind speed measured at the Nidingen station are shown. It should be noted that the wind speed presented is based on an hourly average. For this reason, data on wind gust speeds have also been collected, as presented in Figure 1-15.

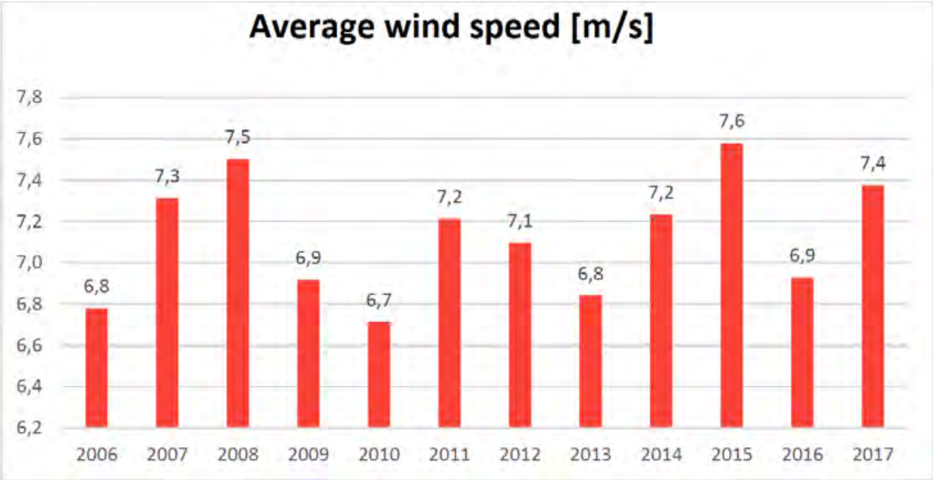

Figure 1-13. Average annual wind speed measured at the Nidingen station. The figure is an edited version of the original in [6] as the heading has been translated.

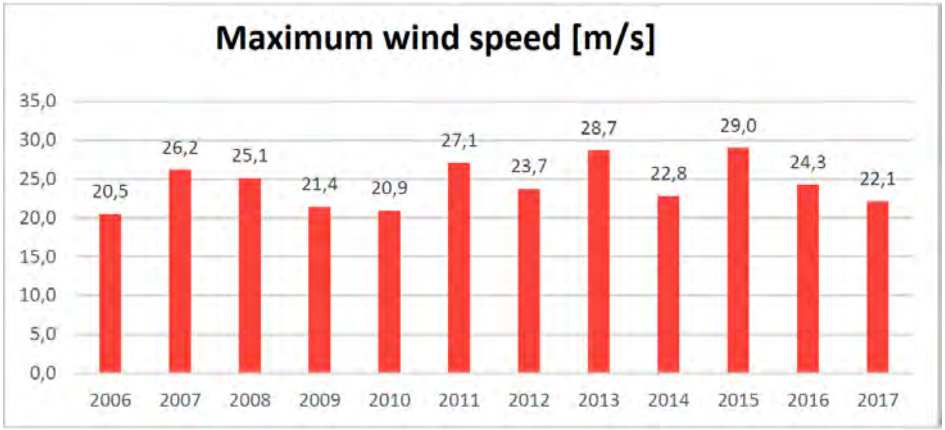

Figure 1-14. Highest annual wind speed measured at the Nidingen station. The figure is an edited version of the original in [6] as the heading has been translated.

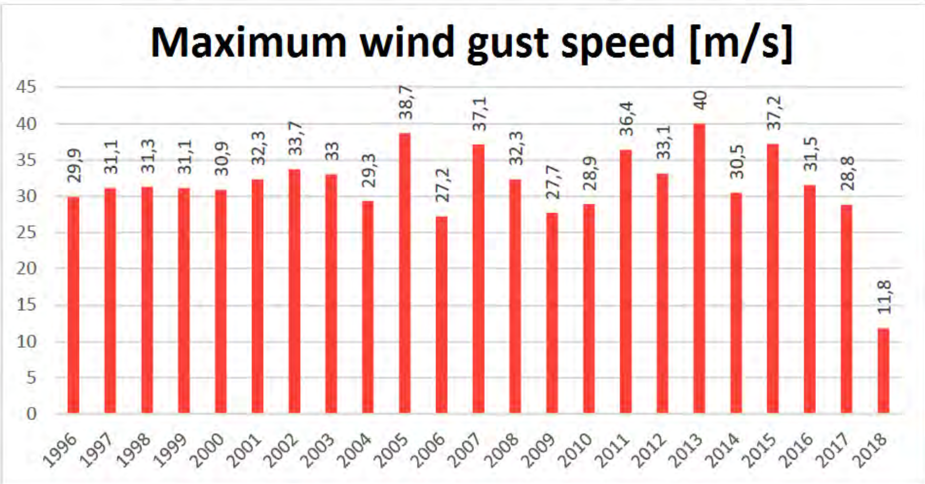

Figure 1-15. Maximum wind gust speed measured at the Nidingen station. The figure is an edited of the original in [6] as the heading has been translated.

1.3.2. Temperature

In [6], annual average, maximum and minimum temperatures measured in Varberg during 2006 – 2017 have been collected. The data are presented in Figure 1-16, Figure 1-17 and Figure 1-18.

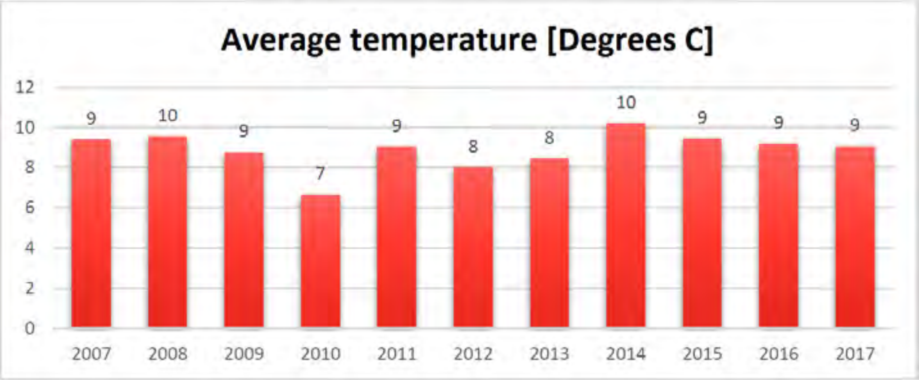

Figure 1-16. Average annual temperature measured in Varberg. The figure is an edited version of the original in [6] as the heading has been translated.

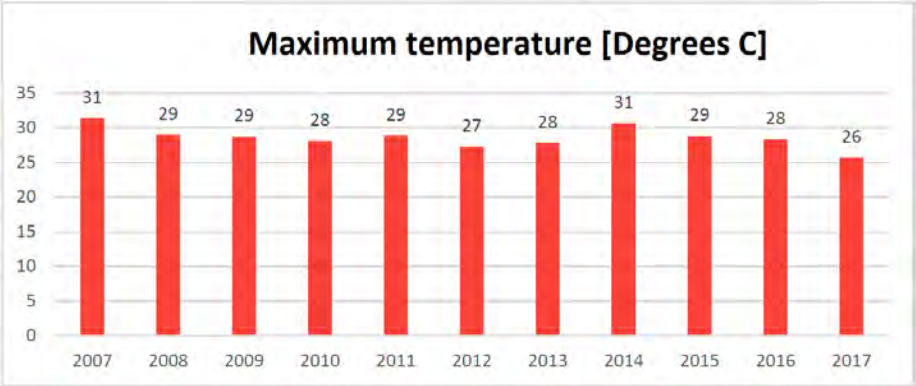

Figure 1-17. Highest measured temperature in Varberg. The figure is an edited version of the original in [6] as the heading has been translated.

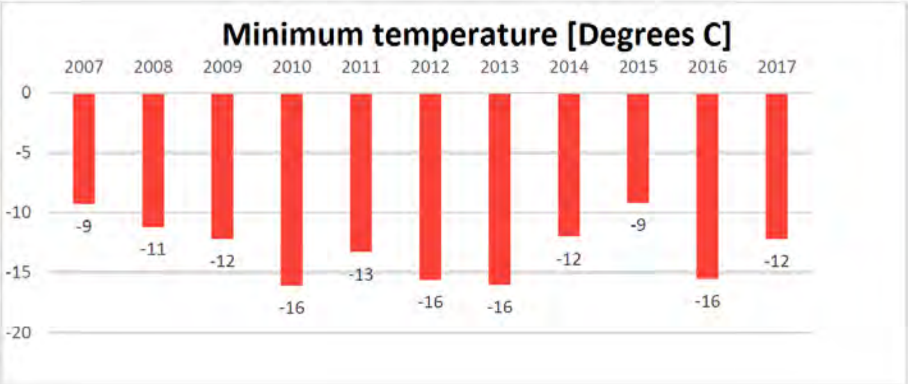

Figure 1-18. Lowest measured temperature in Varberg. The figure is an edited version of the original in [6] as the heading has been translated.

1.3.3. Precipitation

In [6], precipitation data from local measurement station have been gathered. Average and maximum daily precipitation are shown in Figure 1-19 and Figure 1-20. The total annual precipitation is shown in Figure 1-21.

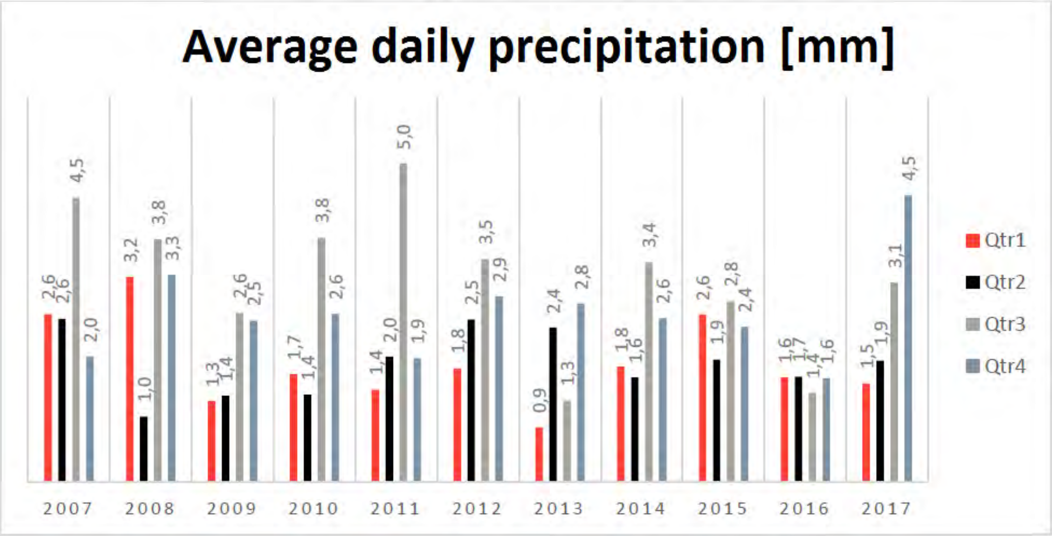

Figure 1-19. Average daily precipitation in the area per year and quarter. The figure is an edited version of the original in [6] as the heading has been translated.

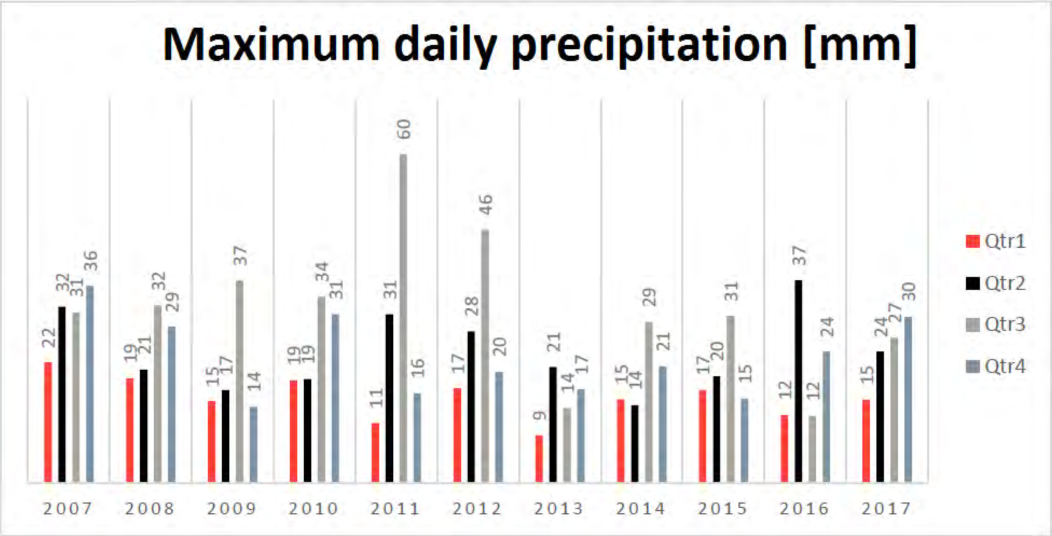

Figure 1-20. Highest daily precipitation in the area per year and quarter. The figure is an edited version of the original in [6] as the heading has been translated.

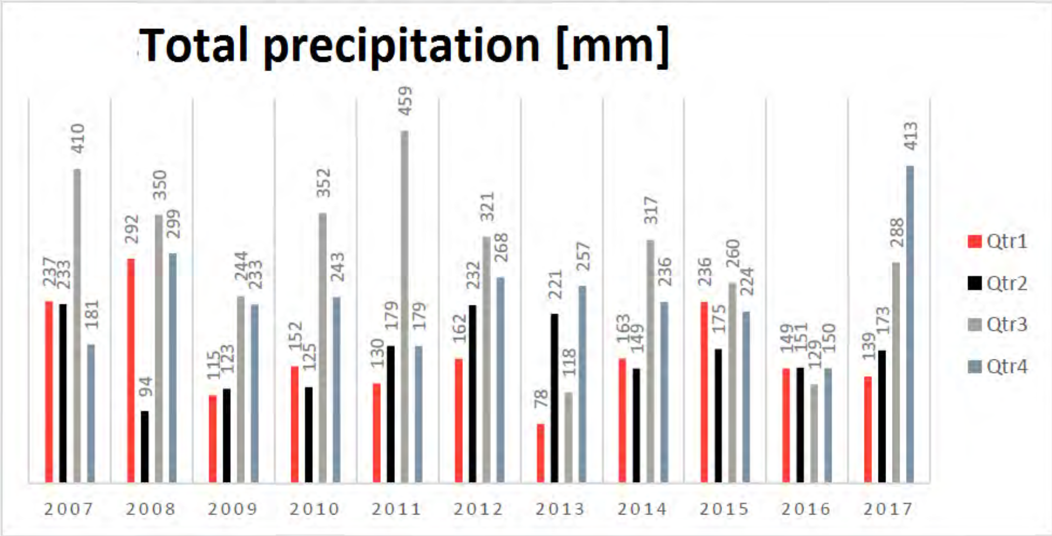

Figure 1-21. Total precipitation in the area per year and quarter. The figure is an edited version of the original in [6] as the heading has been translated.

1.3.4. Extreme weather

SMHI has performed an analysis of extreme weather conditions at the Ringhals site [7].

In the study it is noted that only one weather station in the region has measured more than 90 mm of daily precipitation over a 50-year period. The broader region has, however, suffered from relatively extreme rain weathers locally. During the year 2002 the Orust region (approximately 100 km north of Ringhals) received up to 200 mm of rain in a 24-hour period. Such weather is, however, very rare[5].

While Sweden generally is not subject to intense weather phenomena, such as tornados and similar, storms do occur. In 2005 for example, the storm Gudrun caused sustained wind speeds of between 30 and 40 m/s in the southern part of the country (including the area where Ringhals is located). SMHI concludes that storms of this magnitude might occur a few times per century.

1.4. Natural resources and foodstuffs

1.4.1. Water utilization

There are no lakes within a 10-km radius from the Ringhals site. There are, however, two rivers, Viskan and Löftaån that flow towards the sea. Neither of these is used as sources of drinking water.

There are several protected water resource areas in the region, see Figure 1-22. The closest of these are approximately 15 km northeast of the plant.

[5] An assessment made in [7] for a similar storm over Ringhals is that it happens with a frequency of once per 10 000 years.

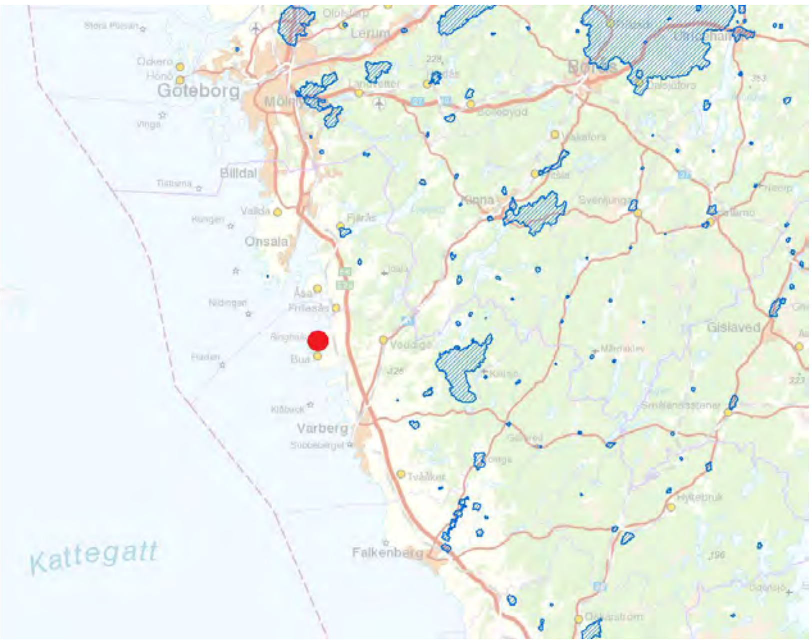

Figure 1-22. Protected water resources in the vicinity of the Ringhals site (red dot) (Source Naturvårdsverket, 2018).

1.4.2. Principal food resources

In addition to the sea the two rivers Löftaån and Viskan are also used for fishing activities. The rivers are mainly used for small-scale recreational fishing while the sea is a resource for commercial fishing. In Table 1-2 the principal fishing resources are given.

Table 1-2. Principal species for fishing per body of water.

Waterbody	Species
Löftaån (river)	Salmon, trout, pike, perch, roach and bream
Viskan (river)	Perch, bream, chub, pike, trout, ide, salmon, whitefish, burbot and eel
Kattegat (sea)	Herring, sprat, cod, mackerel, haddock, whiting, coalfish, flatfish, shellfish and blue mussels

The area around Ringhals is also used for agricultural production. In [8] data regarding animal and farmland use within 10 km from the Ringhals site have been compiled. The data are presented in Table 1-3 and Table 1-4.

Table 1-3. Number of production sites for various livestock within 10 km radius from the Ringhals site [8].

All production sites	Cow	Pig	Fowl	Sheep	Goat	Other
66	19	13	7	19	1	7

Table 1-4. Crop cultivation, grazing and other uses of farmland within a 10-km radius from the Ringhals site [8].

Use/crop	Arable land units
Grassland on cropland	830
Pasture	280
Barley	200
Oats	190
Fallow land	170
Wheat	90
Protective zone	50
Vegetable farming	40
Rape seed	20
Potato	10
Field for birds etc.	15
Garden crops	10
Rye	5
Triticale	5
Silage	5
Field bean and peas	5
Strawberry farming	2
Hay meadow	2
Fruit farming	1
Green manure	1

1.4.3. Foodstuff distribution

Agricultural activities account for approximately 4% of the land use in Halland County. This is slightly less than the national average.

The main share of the Swedish agricultural and fishing exports goes to the neighbouring counties. Statistics on the amount of foodstuff for export from the specific area around Ringhals is, however, not available.

In chapters 3 and 4 it is demonstrated that the local dose due to the ingestion of locally produced foodstuffs is insignificant (conservatively assessed to be on the order of 0.1 μSv/year during the peak year) which means that any dose due to the export of foodstuff to other member states is negligible, and therefore does not justify any further investigation into foodstuff export statistics.

By monitoring the releases from the plant (see chapter 2 and 8), it is ensured that there are no releases to the environment around the plant that could significantly affect the fishing or agricultural products in the area around the plant.

2. The installation

This chapter will give a short description of the facilities to be dismantled and their waste management systems. The purpose of the descriptions is to give an understanding of the factors of relevance for evaluation of the risk for contamination of other member states. Details that do not serve this purpose have been omitted from this report.

2.1. Brief description and history of the installation to be dismantled

Ringhals 1 is a boiling water reactor (BWR) designed by Asea-Atom. The unit was commissioned in January of 1976. The turbine side consist of two turbine strings. The thermal capacity is 2540 MW.

Ringhals 2 is a three-loop pressurized water reactor (PWR) designed by Westinghouse. The unit was commissioned in May of 1975. The turbine side consist of two turbine strings. The thermal capacity is 2652 MW.

Ringhals 1 and 2 share water inlet buildings, diesel building and personnel buildings as well as some waste handling facilities. In Figure 2-1 a layout of R1 and R2 showing the areas that will be decommissioned, is presented.

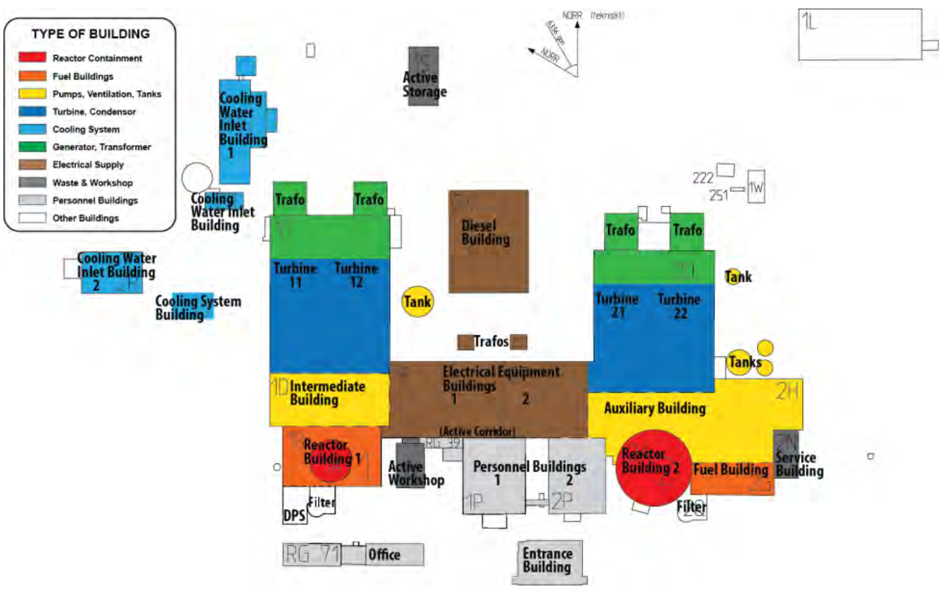

Figure 2-1. The buildings and installations that will be decommissioned during decommissioning of R1 and R2. A description of the supporting waste management infrastructure at the Ringhals site is presented in section 2.4.

2.1.1. Brief description of R1

Below follows a brief description of the main buildings at R1.

Reactor containment

The reactor containment is enclosed in the reactor building. The interior of the containment is separated into two different volumes: the dry well, where the reactor pressure vessel and all connecting piping are located, and the wet well, which is an annular chamber in the bottom of the containment containing the condensation pool.

The containment contains the reactor pressure vessel, the concrete biological shield, recirculation pipes and pumps as well as main steam piping. Inside the reactor pressure vessel are the reactor internal components. The building is made of reinforced steel-lined concrete and has a length of 31 m, a width of 50 m and a height of 51 m. The depth below ground is 18 m.

The entire building constitutes a radiologically controlled area.

Reactor building

The reactor building comprises the reactor containment, process systems close to the reactor, spent fuel storage pools, and the handling pool. The upper level is the reactor hall, were the spent fuel storage pools, handling pool, and the space for storing the reactor vessel head and dome during the outages is located. The building consists of nine floors above ground and three floors below ground.

The entire building constitutes a radiologically controlled area.

Intermediate building

Between the reactor building and the turbine building is the intermediate building. This building contains the feed water pumps, the ventilation outlet from the controlled areas, condensate water clean-up filters, etc. On top of the intermediate building is the main ventilation stack which has a height of 110 m above ground level. Culverts to the waste management building also connect to the intermediate building. The building consists of six floors, four of which are above ground. To a large extent, the building is made of concrete.

The entire building constitutes a radiologically controlled area.

Turbine building

Installed in the turbine building are the main turbines, generators, condensers, main steam and feed water systems, condensate systems, main cooling pumps and the auxiliary systems for turbines and generators. The building has a length of 78 m, a width of 61 m and a height of 26 m. The depth below ground is 12 m. To a large extent, the building is made of concrete. The turbine building is physically separated from the reactor building.

The entire building constitutes a radiologically controlled area.

Electrical building

This building is shared between R1 and R2 and contains the main control rooms as well as rooms with electrical equipment, such as relays, batteries, rectifiers and computer systems. The building also has some rooms for service. The building consists of three floors above ground and two floors below ground. The supporting body is made of concrete.

In this building the only controlled area is a corridor that links the R1 intermediate building and the R2 auxiliary systems building.

Cooling system building
This building contains systems for diversified component cooling, water treatment and distribution, demineralized water, fire protection water, etc. The building has a length of 33 m, a width of 22 m and a height of 13 m. The depth below ground is 4 m. The supporting body is made of concrete.

There are no controlled areas in the building.

Service building
The service building consists of two parts: a mechanical workshop and a service area for maintenance of control rod drive mechanisms during outage. The latter part has a passageway to the reactor building. The building has a length of 25 m, a width of 14 m and a height of 9 m. The depth below ground is 3 m. To a large extent, the building is made of concrete.

The entire building constitutes a radiologically controlled area.

Personnel building
This building contains mainly offices and changing/shower rooms. There are, however, controlled areas as the building contains systems for distribution of contaminated water.

Filter building
The filter building contains a scrubber and related equipment to help reduce the radioactivity in an emergency resulting in release during operation. The building is four floors high and to a large extent made of concrete. The system has never been used for filtration of radioactive gases. Thus, the building and equipment are not radioactively contaminated.

There are no controlled areas in the building.

Cooling water inlet buildings
The cooling water inlet buildings consist of three separate buildings: inlet building 1 (1R), inlet building 2 (2R) and a coupling chamber (2R, adjacent to 1R). The buildings provide cooling water for both reactors. Part of this function is to clear the water of debris and biological matter. The supporting body of the buildings is made of concrete.

There are no controlled areas in the building.

Diversified Protection System building
The Diversified Protection System building contains electrical, instrument and control equipment forming the diversified protection system, a system installed as a complement to the original protection system. The building consists of four floors above ground and one floor below ground. The supporting body of the buildings is made of concrete.

There are no controlled areas in the building.

Waste management building
The main waste management building is a separate building, placed technical west of Ringhals 1, see Figure 2-5. The building contains waste management systems, e.g. liquid

waste management equipment, dry scrap management areas, etc. The building is connected to the reactor building in Ringhals 1 through culverts below ground. The supporting body of the buildings is made of concrete.

Active storage
The active storage is a steel shed which was used to store low level material and equipment used during refuelling etc.

2.1.2. Brief description of R2

Below follows a brief description of the main buildings at R2.

Reactor containment
The reactor containment is cylindrical and contains the reactor pressure vessel, the concrete biological shield, steam generators, main circulation pumps, pressurizer tank, main steam piping and other associated pipes. Inside the reactor pressure vessel are the reactor internal components. The containment is built of reinforced steel-lined concrete. It provides radiation protection around the main sources of radioactivity in the plant, such as the reactor fuel, the reactor coolant and the structural materials in and around the core. The containment is dimensioned to enclose radioactive materials and fission products and prevent it from reaching the surroundings. The wall consists of a 55-m high concrete cylinder with an inner diameter of 35.4 m and a wall thickness of 1.1 m.

The entire building constitutes a radiologically controlled area.

Turbine building
Installed in the turbine building are the main turbines, generators, condensers, main steam and feed water systems, condensate systems, main cooling pumps and the auxiliary systems for turbines and generators. The building has a length of 73 m and a width of 61.5 m. To a large extent, the building is made of concrete. The building consists of two almost identical parts, one on the east side and one on the west side. Below the north part of the building the cooling water inlet is found and in the central part of the building the cooling water outlet. The upper part of the building consists of a large hall that is divided in two parts where each part comprises a turbine placed in the center and a generator placed north of the turbine. The condensers are placed below the turbines. The salt water pumps are located in the building.

There are no controlled areas in the building.

Electrical building
This building is shared between R1 and R2 and contains the main control rooms as well as cable vaults, relay rooms, I&C systems and switchgear rooms. The building has a length of 52 m and a width of 39 m. The building is made of concrete and separated from adjacent buildings.

Inside the building there is a corridor leading from the changing room to the R1 intermediate building and R2 auxiliary systems building. Beneath the corridor there is a culvert for process pipes. These areas are controlled areas.

Fuel building
The fuel building contains two spent fuel pools and a canal with a sluice for fuel transport to the reactor containment. The building also contains equipment for cooling and clean-up of the fuel pools. The building consists of three floors with the total height of approximately 26 m and it occupies an area of 42.5 m x 16.0 m. All walls and slabs are made of reinforced concrete or prefabricated concrete elements. The thickness of the slabs and walls varies between 0.2 and 2.0 m, and have in some cases been designed with respect to radiation protection. The fuel building is physically separated from the containment.

The entire building constitutes a radiologically controlled area.

Auxiliary building
The structure of the auxiliary building is of reinforced concrete. The thickness of walls and slabs are between 0.25 and 1.2 m. mostly depending on requisite thickness for shielding.

The auxiliary building consists of three parts:

- Part 1 contains mainly the pumps, heat exchangers and tanks associated with the emergency core cooling systems and containment spray. There are also tanks and pumps associated with the liquid waste management system as well as the ventilation system. On the roof of part 1 is the main stack.
- Part 2 contains mainly the steam line pipes and associated safety valves.
- Part 3 contains e.g. an area for intake of equipment.

Parts 1 and 2 constitute radiologically controlled areas. Part 3 does not constitute any radiologically controlled area.

Diesel building
The diesel building is shared between R1 and R2 and contains four emergency diesel generator sets for each unit and distribution systems for demineralized water, fire protection water, steam boiler etc. The building is 52 m long and 40 m wide. The building height over ground level is 6.5 m and below ground level 8 m.

There are no controlled areas in the building.

Service building
The service building contains workshops as well as an area for storing concrete waste moulds.

The entire building constitutes a radiologically controlled area.

Personnel building
The Personnel building contains mostly offices, a lunch room, changing/shower rooms, etc.

The building contains passageways to controlled areas and has also previously been used as a laundry room for contaminated clothing. There are therefore controlled areas in the building.

Filter building
The filter building contains a scrubber and related equipment that will help reduce the radioactivity in an emergency resulting in release during operation. The building is 15 m long, 14 m wide, has a total height of 21 m and consists of four floors. It consists of two major parts. One is a standing cylindrical cistern (scrubber) with a diameter of 7 m and a height of 10 m. The other is the auxiliary structure over four floors. The structure is made of concrete. The thickness of walls varies up to 1.4 m, mostly depending on requisite thickness for shielding. The system has never been used for filtration of radioactive gases. Thus, the building and equipment are not radioactively contaminated.

There are no controlled areas in the building.

Cooling water inlet building
The cooling water inlet buildings consist of three separate buildings: inlet building 1 (1R), inlet building 2 (2R) and a coupling chamber (2R, adjacent to 1R). The buildings provide cooling water for both reactors. Part of this function is to clear the water of debris and biological matter. The supporting body of the buildings is made of concrete.

There are no controlled areas in the building.

2.2. Ventilation systems and the treatment of gaseous and air-borne wastes

R1 and R2 have separate ventilation systems with somewhat different designs owing to the different reactor types. The philosophy of the systems is, however, the same. Air is ventilated in a direction from areas with less risk of airborne contamination to areas with a higher risk, and eventually out through the main stacks. All evacuated air is measured and recorded during discharge to the surroundings.

During the care and maintenance period, the ventilation of the areas with radioactive and contaminated material will continue to be in operation in the same manner as during normal power operation. During the decommissioning, the requirements regarding filtration of exhaust ventilation from controlled areas will be regulated by license conditions issued by SSM.

Under normal conditions during decommissioning and dismantling the ventilation of Ringhals 1 is planned to be unfiltered. If there is a risk of radioactive release, it is possible to manually connect the ventilation discharge to both carbon and HEPA-filters. Ventilation of active parts of the waste management building are filtered by HEPA-filters before release through the ventilation stack of Ringhals 1.

At Ringhals 2, the ventilation discharge passes HEPA-filters continuously, even during decommissioning and dismantling.

Principal diagrams of the ventilation systems for R1 and R2 during decommissioning are shown in Figure 2-2.

At some point during the decommissioning the ventilation system itself will be decommissioned. At this point the system, or sections of it, will be replaced by e.g. temporary mobile solutions that meet applicable criteria.

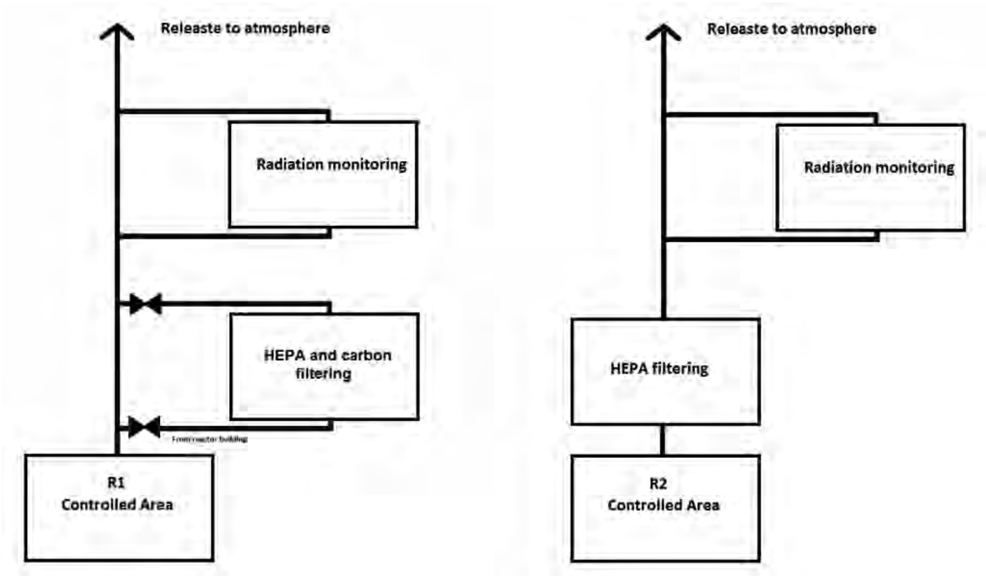

Figure 2-2. Principal diagram of the ventilation systems for R1 and R2 during decommissioning.

2.3. Liquid waste treatment

Within the controlled areas at Ringhals 1 and Ringhals 2, all water that is removed from the process is routed to the waste handling systems on each unit. Radioactive liquid waste is treated using precoat filters as well as ion exchange resins (IOX) and evaporation (at R1).

To facilitate the treatment of waste water, the choice of handling method is made with respect to the origin.

The treated water is either reused or released to the sea, while the residual radioactivity (e.g. in filters and ion exchange resins), is transported to the waste management building for handling as solid waste or for solidification in concrete moulds. The waste management building solidifies waste from the entire Ringhals site.

After shutdown, the unit-specific waste treatment systems will continue to be used to treat the water to reduce the discharge of radioactivity to the sea. These systems will, however, be decommissioned during the project, and at that point other solutions will have to be deployed. This could, for example, be implemented using mobile tanks with subsequent treatment. It should be noted that parts of the R1 processing systems are also located in the waste management building. As R1 is decommissioned, these parts will be disconnected from the system, but may be repurposed for other uses.

The waste management building, however, is situated outside the decommissioning area, and is not subject to decommissioning.

The overall principle of the liquid waste treatment is to purify the water so that it can be reused in the process systems or released to the sea. Residual waste is solidified and treated in accordance with the requirements for final disposal.

In the following sections a short description of the unit-specific liquid waste management systems is given.

2.3.1. Ringhals 1

The liquid waste management system at Ringhals 1 consist of several subsystems: the liquid waste collection system, the system for processing liquid waste, the system for processing solid waste and the liquid waste discharging system.

An overview of the liquid waste management system at R1 is presented in Figure 2-3.

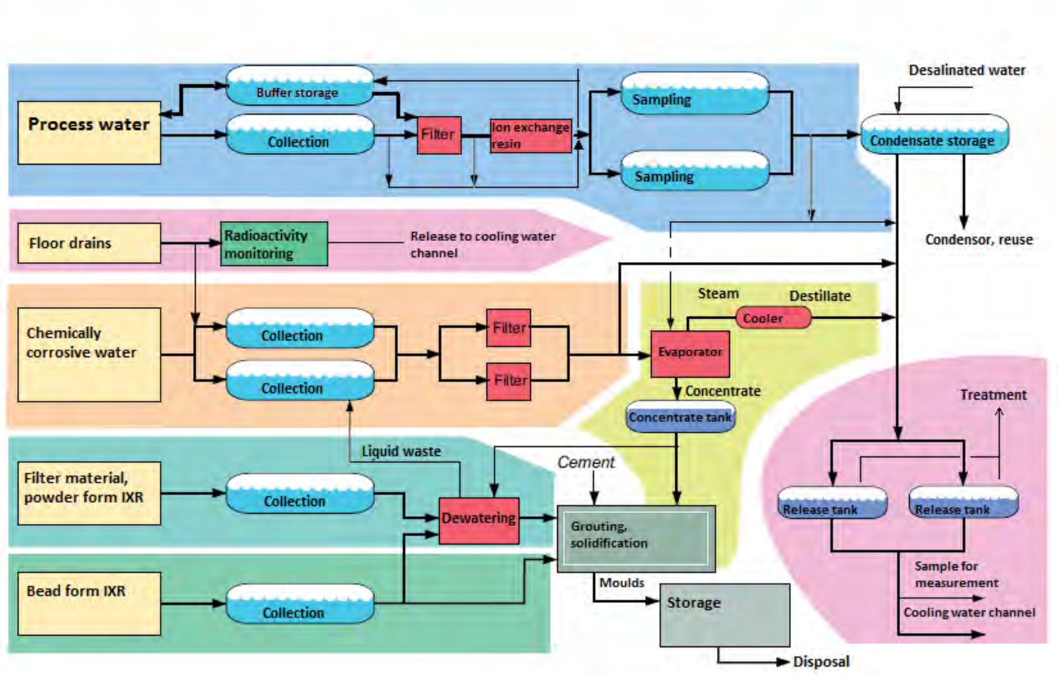

Figure 2-3. Liquid waste management system, R1 (Source KSU Online, modified from original).

System for processing liquid waste

In order to render the treatment of waste water more effective, the treatment method is selected with respect to the origin of the water. Management of liquid radioactive waste is therefore divided into seven lines:

Line 1 receives water from the process systems in the plant. This includes leakage water or water drained from turbine systems, the reactor drainage system or spent fuel pools. The treatment is carried out by filtering and ion exchange before the water is collected in a storage tank. Treated water is reused and returned to the condenser.

Line 2 receives radioactive sludge from the decontamination facility, floor drainage and from waste from cleaning of tanks and pools. The sludge is collected in a tank before it is dispatched for solidification in the waste management building.

Line 3 receives water from the liquid waste collection system (see below), decontamination water from chemical laboratories, water from the decontamination facility, the waste handling building and workshops in controlled areas. This water is treated using filters and an evaporator (in line 5), before release to the cooling water channel.

Line 4 receives and treats ion exchange resins used in the liquid waste management system. The resins are mixed with water and collected in tanks. After decanting, water is passed to line 3 while the resin mix is pumped to the waste management building for solidification with cement in concrete moulds.

Line 5 consists of the evaporator, which treats the water through evaporation leaving a residual sludge containing the radioactive particles. Line 5 receives water from lines 1 and 3. The distillate proceeds to line 6, before release to the cooling water channel. The radioactive sludge is sent to the waste management building.

Line 6 receives treated water from the other lines. The water is collected in tanks, before release to the cooling water channel. The release from the tanks is carried out via the liquid waste discharging system.

Line 7 consists of a buffer storage tank that receives water from the reactor pool, internal parts pool and the condensation pool. The system can store and treat water for reuse in the pools. The line is also designed to redistribute water between line 7 (i.e. buffer tank) and line 1.

System for processing solid waste
Residual radioactive liquid waste from the system for processing liquid waste is transported to the waste management building. The waste is mixed with cement so that the liquid waste solidifies in concrete moulds.

This system handles waste from all Ringhals units. The system is situated in the waste management building located outside Ringhals 1.

Liquid waste collection system
The system collects waste water such as drain water, floor drainage etc. from controlled areas. The system collects the waste water in local tanks. Water from areas where contamination is not normally expected is pumped to the liquid waste discharge system, while water where contamination in expected is pumped to the system for processing liquid waste (line 3).

Liquid waste discharging system
The liquid waste discharging system is used to discharge water to the cooling channels.

For untreated water (from areas where contamination is not expected) there is an online measurement system that can stop the release in case of activity levels above specific criteria (see section 4.3).

Before release of treated water, a representative measurement is taken by mixing the liquid in the release tank before retrieving a sample for analysis. If the radioactivity levels exceed the discharge limits, the liquid is pumped back to the treatment lines. When it has been confirmed that the discharge limits are met, the release is performed manually.

During release of the treated water from controlled areas, a representative sample is taken from the released liquid for the purpose of release reporting. These samples are stored and analyzed with subsequent reporting to SSM at specified intervals.

Tank capacity
The storage capacity of the collection tanks in the process water treatment line (line 1) is 265 m3 and in the chemically corrosive treatment line (line 3) 140 m3 and 105 m3 respectively.

The process water buffer tank has a capacity of 3200 m3.

The storage capacity of the tanks covers the volumes required for the decommissioning.

2.3.2. Ringhals 2

The waste disposal system at Ringhals 2 consists of several subsystems. The liquid waste is treated by the liquid waste processing system, liquid waste collection system, liquid waste discharging system and vent and drain system.

The liquid waste processing system at R2 is schematically presented in Figure 2-4.

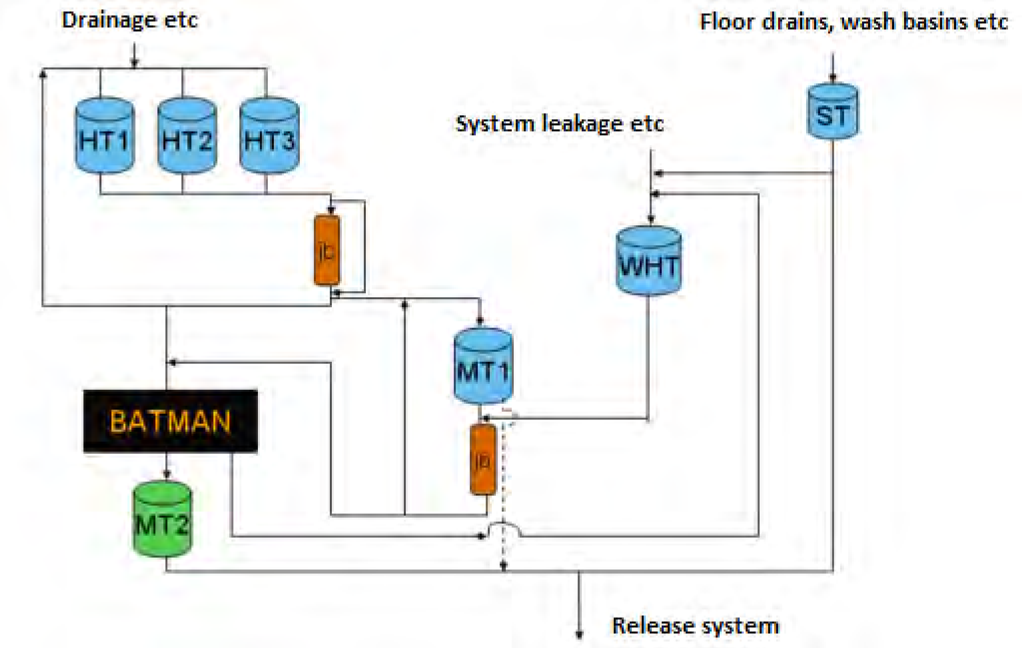

Figure 2-4. Liquid waste processing system, Ringhals 2 (HT = hold up tank, WHT = waste hold-up tank, MT = monitoring tank, ST = sump tank, JB = ion exchange resin, BATMAN is a treatment system, see below) (Source KSU Online))

Liquid waste processing system
The liquid waste processing system collects and processes potentially contaminated liquids for recycling or discharge.

Reactor coolant drainage is collected inside the containment and is normally pumped to three hold-up tanks (HT). After processing, the water is normally directed to two monitoring tanks (MT).

System leakage and drainage from other controlled areas are directed to the waste hold-up tank (WHT).

The waste water is treated through filters and ion exchange resins. The main part of the treatment system is a subsystem called BATMAN (Best Available Technique for Minimizing All Nuclides) which treats radioactive waste in several steps:

1. The liquid passes through a set of filters, both macro and microfilters.
2. After the filters the liquid passes through a gas transfer membrane to separate noble gases and hydrogen gas.
3. The liquid then passes through reverse osmosis membranes and nanofilters for further filtration.
4. Finally, the liquid is treated either using an ion exchange resin or electrochemical ion exchange.

After treatment the liquid is pumped to the liquid waste discharging system. Spent filters and ion exchange resins are transported to the waste management building for handling or for solidification.

Liquid waste collecting system
The function of the liquid waste collecting system is to collect the floor drainage water from the containment, auxiliary, fuel and turbine building and store it in local sump tanks (ST).

Water from areas where contamination is not normally expected is pumped to the liquid waste discharging system, while water where contamination in expected is pumped to the system for processing liquid waste.

Liquid waste discharging system
The liquid waste discharging system is used to discharge water to the cooling channels.

Untreated water (from areas where no contamination is expected) is measured on-line. It is possible to stop the release in case of activity levels above specific criteria (see section 4.3).

Before the water is released to the environment it is controlled by representative sampling and analysis. If the radioactivity levels exceed the discharge limits the liquid is pumped back to the treatment lines. When it has been confirmed that the discharge limits are met, the release is performed manually. The release path is equipped with online measurement systems installed on the outside of the release pipes that automatically terminate the release if the activity exceeds specific criteria (see section 4.3).

During release, a representative sample is taken from the released liquid for the purpose of release reporting. These samples are stored and analysed with subsequent reporting to SSM at specified intervals.

Vent and drain system
The system collects and transports water from drains and venting connected to radioactive system parts. The system consists of two separate parts: one part collects drain water and venting from containment, and the other from the auxiliary and fuel building. The water is collected in sump tanks and transported to the liquid waste processing system.

Tank capacity
The storage capacity of the waste water collection tank (WHT) is 91 m3, and 190 m3 each for the three hold-up tanks HT1-3.

2.4. Solid waste treatment

There are different options available for treating radioactive solid waste. In this section, a short description is given of different methods that are considered to be used during decommissioning.

The methods are chosen based on the most optimized alternative for each waste property. The chosen method must always meet the current requirements and must be approved.

2.4.1. Solid waste treatment methods

Dismantling and mechanical reduction
An important part of solid waste treatment is the dismantling and mechanical processing of material. The treatment is used to optimize the size of the component or to reduce the activity of contaminated or potentially contaminated surfaces. Several different types of methods are used for mechanical segmentation, e.g. sawing and shearing.

Compaction
A large proportion of radioactive solid waste is bulky. By performing compaction prior to disposal, the volume of organic waste and other low-density materials can be significantly decreased. Compaction can also be used on metallic materials to obtain an optimized volume and a good use of the container volume.

Vacuuming, wiping and manual processing
Decontamination of waste items by vacuuming, wiping and manual processing is a proven method in order to avoid spreading loose radioactive contamination and to remove local accumulations of activity.

Hot water washing machine
A large amount of material from the decommissioning of a nuclear power plant is only slightly contaminated to a degree where water-based washing solutions can be sufficient to remove the majority of the contamination. A hot water washing machine used for decontaminating material surfaces yields a good effect at a low cost. The water and filters will be treated and handled in the waste handling building.

High-pressure cleaning
High-pressure cleaning of contaminated items is an efficient waste treatment method when regular water washing is not enough. Pressure washing should be conducted in a closed cabinet to avoid the spread of contamination.

Chemical decontamination
It is advantageous to implement chemical decontamination to reduce contamination levels of components and materials in a nuclear power plant. Chemical decontamination can be performed to reduce the dose rate, to reclassify the waste from intermediate to low level category, or from low level to levels that allow clearance.

Decontamination of concrete and buildings
Surfaces in NPP buildings could be contaminated to such an extent that they need to be decontaminated to meet clearance requirements. There are several different techniques to apply, ranging from simple cleaning with dry or wet methods to removing the contaminated surface with some mechanical or thermal method. Water blasting or hydro demolition are good methods to use when fire and explosion hazards prevail or when dust from traditional blasting could cause problems. The advantage of these methods is that the risk of airborne activity is reduced, but the disadvantage is that the water must be treated and disposed of. The advantage of dry methods in comparison to wet methods is that the volume of waste will be reduced because there is no need to treat the active liquid waste. Traditional methods such as planing/scabbling/shaving as well as methods including removable coverings, nitro-jetting, laser and plasma may be considered in the decontamination of buildings and concrete.

Melting
Melting of radioactive waste has three main advantage: separation of some nuclides from the metal to a slag/filter, homogenization of the contamination within the metal (which enables more precise characterization) and volume reduction. If melting is combined with mechanical decontamination it is a very powerful method for clearance of metals. Materials that can be treated by melting are steel, aluminium, copper, brass and lead.

Incineration of organic material
Incineration of combustible radioactive waste is a proven waste treatment method used for several decades. The advantages of incineration are the volume reduction and the destruction of organic substances (which might affect the disposal facilities' safety functions negatively). Combustible wastes include rags, clothing, wood and paper.

2.4.2. Solid waste management infrastructure

2.4.2.1. Waste management stations at the units

The majority of the waste management of the decommissioning waste will take place in new waste management stations that will be prepared at the units undergoing decommissioning. Preliminary logistics studies have shown that there are suitable locations with adequate space available to manage the expected generated waste amounts.

It is foreseen that a large part of the waste management will take place in the immediate vicinity of the work area and/or at these new waste management stations.

At the Ringhals site there is a waste management infrastructure already in place that is adequate to manage the operational waste from the four (currently) operational units. This includes equipment for decontamination, compaction, clearance measurements etc.

Due to the availability of this infrastructure it is foreseen that it will be used to complement the waste management capacity provided by the new waste management stations. This could for example be necessary in order to manage more advanced waste forms that may need special segmentation capabilities offered in the mechanical workshops.

2.4.2.2. Other waste management areas

In Figure 2-5 the waste management infrastructure[6] at the Ringhals site is shown. A short description of the various buildings and areas is given below. The decommissioning project will likely use this infrastructure as a complement to the waste management stations at the units.

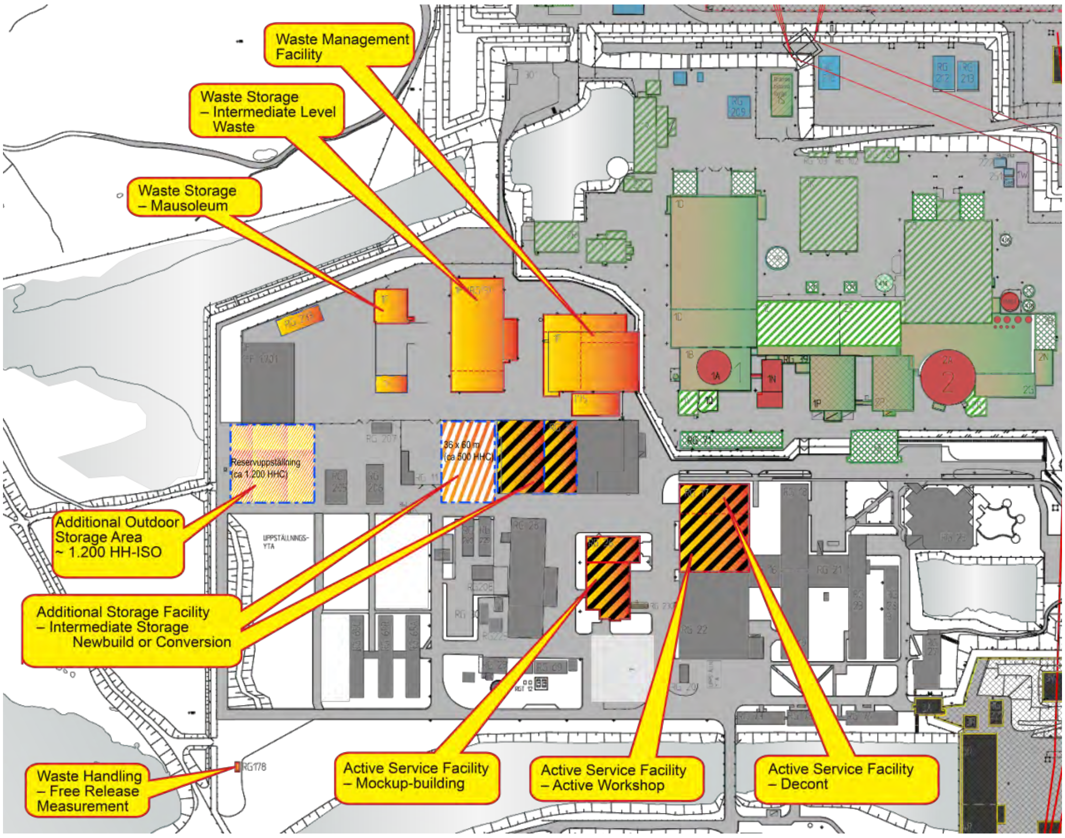

Figure 2-5. Waste management infrastructure at the Ringhals site.

Waste management facilities

Waste management facility

At the site there is a main waste management building comprising parts of the operational waste management systems of all units, e.g. liquid waste management equipment, dry scrap management areas, etc.

[6] Note that some areas such as the mock-up facility are not currently part of the waste management infrastructure, but are considered to be so for decommissioning and therefore included here.

The so-called tank hall area is used for management of liquid waste, sludge, ion exchange resins etc. It also serves as a docking station for mobile liquid waste management equipment. These functions will be used for liquid waste management during the decommissioning. The docking station for mobile liquid waste management equipment might be used when the main liquid waste management system is being decommissioned, for example.

There is also an area for solidification of liquid waste and for grouting and/or capping of waste moulds. The area also has space for curing of grouted waste moulds and for nuclide-specific measurements. This area will be used in management of ILW waste.

Also part of the building is the area for management of general dry active operational waste from all units. The decommissioning project does not foresee that it will use this area for this purpose since this function is handled by the new waste management areas at the units being decommissioned. It is however likely to be used for management of ILW as the area is connected to the grouting area which is used for conditioning ILW.

Clearance measurement area
This area can be used for clearance measurements.

Note that at the time of writing this report it has not been decided where the main areas for clearance management will be located.

Active service facilities

Active workshop
The active workshop contains areas suitable for management of large and/or complex components. It is likely that this area will be used as a complement to the decontamination area with tools for mechanical decontamination.

Mock-up building
This building is currently mainly used for fuel management equipment but is a potential area for management of large components.

Decontamination area
This area contains equipment for decontamination and may be used either as a main area for decontamination of both tools and waste (when advantageous) or as a complement if the new waste management stations will be equipped with decontamination equipment.

Storage facilities

Intermediate level waste store
The intermediate level waste store is for storage of non-incinerable low and intermediate level waste, mainly in concrete and steel moulds[7]. It is also used to store some other types of waste, e.g. large components.

The storage facility consists of a large space with concrete radiation shielding walls to allow for storage of waste with up to 500 mSv/h surface dose rates.

[7] A cubical container with an outer side length of 1.2 m.

The capacity is on the order of 6000 moulds. This storage is deemed to be sufficient for storage of the intermediate level waste in concrete and steel moulds as well as steel tanks from the decommissioning project.

Mausoleum
The so-called mausoleum is a storage facility with concrete radiation shielding walls used to store large components. It is likely that this facility will be used for temporary storage of waste from the decommissioning. One potential component type that might be stored here is the R2 steam generators.

Low level waste storage areas
New storage areas for ISO containers will be built or converted (denoted Additional storage facility in Figure 2-5). There is also an additional outdoor hard stand area for storage of ISO containers. This results in a storage capacity of approximately 500 and 1200 half-height ISO containers respectively.

This storage is deemed to be sufficient for storage of the low-level waste from the decommissioning project.

Other facilities

Harbour
South of the site, Ringhals has its harbour capable of receiving ships for transportation of waste and components. This harbour is used to receive, for example, m/s Sigrid, the main radioactive waste transport ship in Sweden capable of transporting both fuel and waste packages. It can also be used to receive other ships capable of transporting for example large components such as steam generators away from the site.

Surface repository
A surface repository for very low-level operational waste is situated on the site today. The capacity of existing surface repository is based on estimated volume of operational waste production on the Ringhals site. A surface repository for very low-level decommissioning waste disposal is planned to be established on the site. The capacity will be of the same order as the existing surface repository. A separate permit and approval from the Swedish authorities is required for such a surface repository.

2.5. Containment

Radioactivity in decommissioning waste is to a large extent not volatile or easily spread. Induced activity in construction material is bound to the material and only released to some extent during physical processes such as during sawing. Oxide coating that normally occurs on the inside of systems is similarly protected until the physical process of dismantling is initiated. Before decommissioning a large part of primary systems will undergo chemical decontamination, which means that the majority of the oxide layer is removed [8]. During sawing and segmentation of potentially dusty objects it is likely that local vacuum and filtration systems will be used[9].

Radioactive components will be treated within the existing buildings or in temporary buildings designed to contain radioactivity and minimize the risk of cross contamination

[8] The contamination reduced by a factor of approximately 10-1000.
[9] The specific safety measures are described in the package descriptions reported to SSM (see chapter 0.2.1).

outside the building. Dismantled components and building waste will be free released after clearance, or sorted according to activity level and then packed according to clearly defined routines.

Waste is packaged in suitable containers depending on the type of waste. As part of the safety assessment of a facility, the waste producer is required to produce a so-called Waste Type Description (WTD), containing a description of the waste handling process from the time of production until it has been finally disposed of in a final repository. This includes a description of packaging criteria, interim storage criteria and transportation criteria. These WTD presents how the waste fulfils containment criteria, meets the waste acceptance criteria for transport, the repository, and any other facility it is handled in, and are approved by SSM. A WTD is waste type specific, i.e. one WTD is produced for a given waste type, and therefore applies to all packages of that type.

SKB manages a transport and disposal system in Sweden, see further Appendix 1. To a large extent, waste is transported using a purpose-built ship, m/s Sigrid. This ship is classified according to the category INF 3 in the International Maritime Organization scheme, and has been constructed with redundant safety systems in order to ensure safe transport of radioactive materials.

Other modes of transport are also possible, but not used on a regular basis. This includes transport by road and by sea, e.g. of large components. Such transports need approval from SSM, which thereby evaluates the containment measures implemented.

For external treatment, acceptance criteria of the treatment facility are established and the requirements have to be met before the waste is treated. After treatment the waste is either returned to the waste owner for further treatment, or the external treatment supplier packages the residual radioactive waste in accordance with the associated waste type description[10] (WTD). As long as the waste acceptance criteria (WAC) are met, the containment procedures during external treatment are the responsibility of the treatment facility operator, which are described in the facility's safety analysis report.

In accordance with the above, one of the most important documents, ensuring well-planned and safe waste management is the waste handling plan. This plan form part of the safety analysis report (SAR). Also the different WTD are part of the SAR, both at Ringhals (as being the waste producer) and at waste treatment facilities as well as the disposal facilities. Before a new kind of waste is produced, the WTD should be submitted to SSM for review and approval.

[10] Note that the owner of the waste sent for treatment is still the owner of any secondary waste generated during the treatment process.

3. Release from the installation of airborne radio-active effluents in normal conditions

3.1. Authorization procedure in force

3.1.1. Legislation on nuclear activities

The Act on Nuclear Activities 1984:3 and the Ordinance 2008:452 with instructions for SSM have been translated into English and can be found on the SSM website (www.ssm.se). The Radiation Protection Act 2018:396 is not yet published in English. These acts and ordinance, together with the Ordinance on nuclear activities 1984:14 and the Radiation Protection Ordinance 2018:506, stipulate the boundaries for all nuclear activities in Sweden. SSM has developed regulations (SSMFS) to give a more detailed framework for NPPs and other nuclear facilities. Some of the regulations are available in English.

SSM oversees that nuclear operations are conducted safely by issuing regulations as well as carrying out inspections, follow-ups and checks of activities related to nuclear safety and radiation protection. SSM´s aim is to protect people and environment from ionizing radiation through the application of best available technology and the optimisation of radiation protection.

The most important regulations with respect to this aim are:

- SSMFS 2008:1: The SSM’s Regulations and General Advice concerning Safety in Nuclear Facilities
- SSMFS 2008:23: The SSM’s Regulations on Protection of Human Health and the Environment in connection with Discharges of Radioactive Substances from certain Nuclear Facilities. The regulations set some of the definitions and parameters on how to perform dose calculations, which has been used in this report.
- SSMFS 2018:1 The SSM´s regulations concerning basic provisions for practices involving ionizing radiation subject to mandatory licensing.
- SSMFS 2018:3: The SSM’s regulations and general advice concerning clearance of materials, rooms, buildings and land in practices involving ionizing radiation

In SSM’s license conditions for decommissioning [2], the requirements above are complemented and specified for the dismantling and demolition phase. This state, among other conditions (not verbatim):

- Measures for limitation of release to air and water shall be adapted to the activities performed and to the conditions present.
- The effective dose to the public from release from all facilities within a geographic area shall not exceed 0.1 mSv/year.
    - Note that release here refers to the sum of release to air and release to water from all facilities, including R3 and R4.

  - Note also that 0.1 mSv/year is the upper limit for planning and that optimisation of radiation protection and the use of best available technique will result in much lower doses to the public.
- Releases should be monitored to the degree it is possible and reasonable. The monitoring should be adapted to the activities performed, the expected radionuclides and the conditions present.
- A plan for how releases is limited and monitored and how plant systems are adapted for this purpose shall be produced and submitted together with the safety analysis report. The plan should be kept up to date.
- Monitoring of radioactive isotopes in the environment around the facility shall be performed and described in a plan.
- Release of radioactive isotopes shall be reported to SSM periodically.

3.1.2. Discharge limits and associated requirements for decommissioning

In Sweden there are no discharge limits specified in becquerel (Bq) for the time before the envisaged dismantling operations or the dismantling operations themselves.

Instead, the annual dose to the public estimated in accordance with chapter 5, sections 2-3 SSMFS 2018:1 from all nuclear facilities situated in the same area should not exceed 0.1 mSv[11]. The use of best available technique and optimization of radiation protection is supposed to ensure much lower doses to the public.

3.1.3. Environmental impact assessment

The Environmental assessment procedure follows the required steps stipulated in the Swedish Environmental Code.

Early in the Environmental assessment process a public hearing is held involving the authorities, neighbours and other stakeholders. An environmental impact assessment (EIA) must be submitted together with the application. The EIA describes the direct and indirect impact of the planned activities. The EIA includes a site description of the plant or activity as well as descriptions of the technology that will be used, considering the best available technique (BAT). The EIA also describes the impact on people, animals, plants, land, water, air, climate, landscape and the cultural environment. Furthermore, it describes the impact on the management of land, water and the physical environment in general, as well as on the management of materials, natural resources and energy.

The Swedish Environmental Protection Agency, the County Administrative Board, the local Environmental and Public Health Committee and SSM are also consulted in the licensing procedure and are given the opportunity to propose conditions. The license is issued by the Land and Environment Court.

[11] The dose constraints refers to the total dose received from all facilities within a geographical area, i.e. it is not specified how much may be received from either a specific facility or how this dose is distributed between discharges to air and water.

3.2. Technical aspects

3.2.1. Origin of the radioactive effluents

The radionuclides that are of most importance in terms of emissions during power operation (mainly C-14 and H-3, but also noble gases and iodine) will no longer be produced when decommissioning commences, although some activity may remain from the operational phase[12]. The release of radioactive substances that may occur during decommissioning will not be due to volatility, but due to cutting- and segmentation activities, pool work, etc., that potentially could release radioactivity in the form of aerosols.

When the fuel has been removed from a reactor, the vast majority of the remaining activity is found in the neutron activated waste. With activation of components, the material structure becomes radioactive, rather than having a loose layer of contamination. Discharge of activated material therefore occurs mainly as a result of physically mobilizing the material itself, which occurs e.g. when cutting the material.

During the operational phase, fission products (and small amounts of uranium and transuranics) are released from the nuclear fuel. In case of a cladding failure, these products can reach the reactor primary system. In addition to this, wear and tear of reactor system materials causes microscopic fragments to be transported with the primary cooling water and potentially to be deposited in the core where they are activated before being rereleased. Together these activation and fission products eventually get deposited on surfaces in the various reactor systems. The deposition can be relatively hard, e.g. as an oxide layer on a material surface, or relatively loose and more dust like[13].

3.2.2. Annual discharges expected during decommissioning

This section will estimate the expected annual discharges during decommissioning. It is based on [9], which also contains further detail.

For Ringhals 1, the emissions of aerosols during power operation have been dominated by Co-60. Ringhals 2 has generally demonstrated zero emissions (i.e. below detection limits) of aerosols because the plant uses HEPA-type absolute filters under normal operating conditions that effectively hinder release of aerosols. The activity inventory of the systems with highest activity (reactor pressure vessel and internal parts) will contain roughly equal activity (Bq) of Fe-55, Ni-63 and Co-60. However, since Fe-55 and Ni-63 nuclides have significantly lower dose factors (at least by an order of magnitude) compared to Co-60, they are less important from a dose perspective. Therefore, the source term applied in this report only includes Co-60 from contamination and induced activity.

There is some experience in release fractions associated with segmentation of reactor internal components at the Swedish BWR NPPs. Both the Oskarshamn and the Forsmark

[12] For example, when a reactor is shut down the production of C-14 will stop. For the pressurized water reactor Ringhals 2, a considerable amount of C-14 can remain in waste gas decay tanks (and some of the ion exchange resins) that could not be ventilated immediately after reactor shutdown. In [9] it is assumed that all C-14 is vented during the first year after shutdown for both R1 and R2.

[13] Note that the full system decontamination to a very large degree removes this kind of contamination and instead transfers it to ion exchange resins. Contamination levels on system surfaces undergoing full system decontamination can often be reduced by a factor of 10 - 1000.

plants have carried out evaluations of the discharges from previous projects to replace reactor internal components. Calculations indicate a Co-60 air release fraction of 1E-07 [10]. As the segmentation of reactor internals is performed under water, releases of airborne aerosols are mainly correlated to the concentration of radioactive aerosols in the pool where the segmentation is performed. The main release paths for airborne aerosols correlate to evaporation of contaminated water from the pools or other surfaces.

While R2 is a PWR, the same release fraction is used. This is based on the similar techniques and conditions applied in the decommissioning, and therefore similar production mechanisms and release paths of aerosols. The highest contributing activity (segmentation of internals) is performed with similar techniques which therefore give a similar concentration and size distribution of aerosols in the pool water.

Furthermore it should be noted that R2 will be operated with HEPA filtration which effectively remove approximately 99.95% of aerosols at a minimum. However, no credit is given to this filtration in this report in order to keep the assessment conservative.

In the absence of specific data on common release fractions during dismantling of non-activated systems, the same air release fraction (1E-07) will be used for all systems regardless of whether the release is due to mobilization of neutron-activated material or surface contamination.

As will be shown in section 3.4, the maximum dose will be dominated by air release during R2 internal segmentation. This means that the uncertainty in releases from other sources is of less significance. For the analysis in this report the plan described in section 0.5.1 has been assumed when estimating the release based on the above fraction. This is done by matching the Co-60 activity in the given systems, reducing it by the release fraction, and assuming that the release occurs at the initiation year of the activity. The resulting release to air is given in Table 3-1 and Table 3-2.

Table 3-1. Assumed dismantling sequence and associated Co-60 release to air from R1 [9].

Initiation year	**Activity causing release**	**Release to air Co-60 (Bq)**
2021	Shutdown, Preparation, Chemical decontamination[14]	0
2022	Dismantling of reactor internal components, dismantling of turbine components, Dismantling of cooling systems, parts of the waste systems, chemical systems	$8.9 \cdot 10^{8}$
2024	Dismantling of reactor pressure vessel and primary systems	$6.9 \cdot 10^{5}$
2025	Dismantling of biological shield, Dismantling of remaining systems	$4.7 \cdot 10^{5}$

[14] The release from these activities is already included in the assumed operational release and not as a release from decommissioning activities, and hence is set to zero as to avoid double counting.

Table 3-2. Assumed dismantling sequence and associated Co-60 release to air from R2 [9].

Initiation year	Activity causing release	Release to air Co-60 (Bq)
2020	Shutdown, Preparation, Chemical decontamination, Dismantling of reactor internal components	$1.4 \cdot 10^{10}$
2022	Dismantling of fuel pool and associated cooling systems, dismantling of steam generators, Dismantling of cooling systems and chemical volume control system	$1.9 \cdot 10^{6}$
2023	Dismantling of reactor pressure vessel and primary systems	$1.7 \cdot 10^{7}$
2024	Dismantling of primary shield, Dismantling of remaining systems	$4.7 \cdot 10^{5}$

3.3. Monitoring of discharges

3.3.1. Sampling, measurement and analysis of discharge

At R1 and R2 there are systems for sampling and measurement of the exhaust through the ventilation stacks. The systems samples part of the exhaust at a position where there is isokinetic airflow and runs this air separately through a measurement chamber and through separate equipment intended to collect aerosols, iodine, C-14 and H-3 respectively.

The air that passes through the measurement chamber is measured online, while the activity collected in the measurement equipment for aerosols, iodine, C-14 and H-3 are replaced at regular intervals and analysed with nuclide-specific measurements at an onsite laboratory.

During operation there is also online measurement and nuclide-specific analysis performed of noble gases.

Together with air flow measurements in the stack and in the sampling system it is possible to calculate the total release.

Figure 3-1 and Figure 3-2 present principal diagrams of the stack monitoring for R1 and R2 respectively. The discharge from the filtered ventilation system of the waste handling building is currently also monitored in the ventilation stack of Ringhals 1. During the decommissioning of Ringhals 1, a separate ventilation and monitoring system for the waste handling building will be needed, since the facility will be used to process waste from the operation of Ringhals 3 and 4.

When the plant undergoes decommissioning, aerosols are the main contribution to the release, see section 3.2.1. The aerosol sampling system will be in operation during the decommissioning phase. Other measurement systems will be in operation during the post operation phase but will be removed when they are no longer required.

During the decommissioning, the source term of the plant will gradually decrease. The monitoring systems will be adapted during the decommissioning and, eventually, it will be acceptable to fulfil the requirements for monitoring of discharges using local monitoring and sampling.

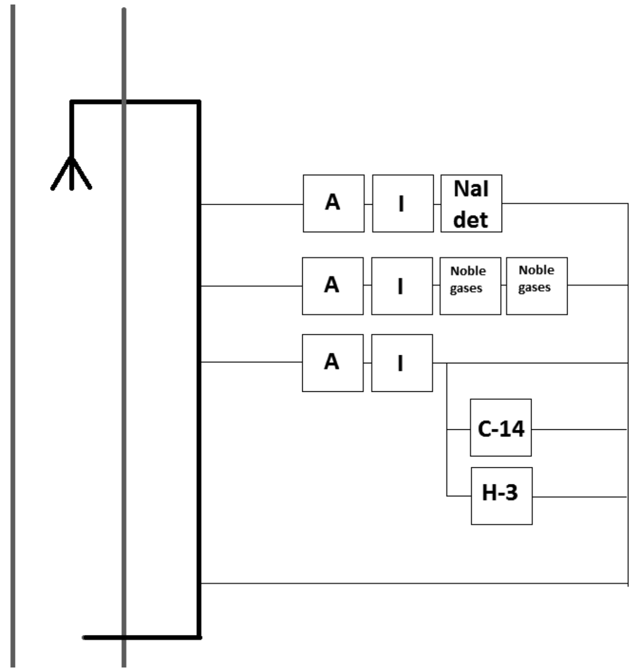

Figure 3-1. Principal diagram of stack monitoring during operation, R1. Boxes show monitoring equipment (A for aerosols, I for iodine).

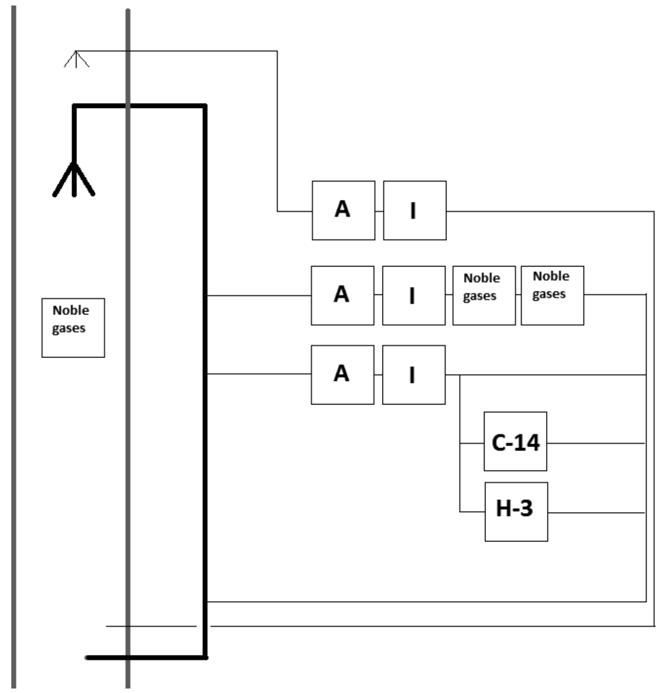

Figure 3-2. Principal diagram of stack monitoring during operation, R2. Boxes show monitoring equipment (A for aerosols, I for iodine).

3.3.2. Alarm levels, intervention actions (manual and automatic)

During operation the main stacks of both R1 and R2 are monitored with automatic alarms for both flow and radiation levels.

In the decommissioning phase, the online stack measuring is planned to be disconnected (note that aerosol measurements will continue to be monitored, but not online), and therefore no activity alarms will be online.

There will, however, be local radiation safety systems in operation whenever deemed necessary to protect personnel, for example during internals segmentation. This can for example include the reactor building radiation safety monitoring system. Such systems, while not measuring discharge directly, give indications if contamination is spread.

3.4. Evaluation of transfer to man

3.4.1. Model description

This section describes the models[15] used for the calculation of dose to a representative person in the vicinity of the plant from release to air.

Note that this discussion is as far as possible kept on a qualitative level with the aim of describing an overview of how the calculations were performed. The models are described in detail in [11], [12], [13], [14], [15], [16] and [17].

Dispersion in air

Dispersion in air is calculated using a Gaussian dispersion model. This kind of model assumes a constant wind direction and wind speed from the release point and allows calculation of the nuclide concentration in any point along the wind direction vector.

It should be noted that R1 has a 110 m high stack, while R2 has a 50 m high stack. However, for R2 a release height of 20 m has been used in the calculations. This is due to R2's reactor building being close to, and of an equal height, as the stack, which causes disturbance of the air flow. Using a lower release height compensates for the building effects.

Atmospheric stability is taken into account by Pasquill class-specific functions used to determine the spread of the Gaussian plume shape (in the height direction the inversion height is used as a limit).

Since the transport time is known by the given distance to the calculation point and the wind speed and direction, this allows the model to take radioactive decay into account.

[15] During the late stages of development of this report a new model (called PREDO) has been developed and is in the process of being implemented for Ringhals. Due to this model being neither reviewed nor approved as this report's calculations were done, the then current model was used. The model has been approved by SSM and has been used for more than a decade to calculate consequences from radioactive releases from the Swedish nuclear facilities. It should, however, be noted that while there are differences in assumptions and definitions between the models, the results are similar. For the conditions used in this report, the results presented are conservative with respect to those that would have resulted if the PREDO model had been used.

Deposition
The nuclides travel with the plume and eventually deposit on the ground by two effects: dry deposition which describes various effects such as physical impact on ground and plants, and wet deposition which describes transport of nuclides from the air to the ground through precipitation.

Calculating annual dispersion and deposition
The dispersion and deposition are both dependent on parameters that vary over the year. This is, for example, the case with the wind speed and direction, atmospheric stability conditions, precipitation etc.

For this reason, the dispersion and deposition need to be calculated for all variations of the above parameters based on meteorological data. If the release rate from the stack is assumed to be constant throughout the year this enables calculation of the annual spread of contamination all around the release point by weighting together the results from individual parameter sets.

Deposition on soil
The nuclides that are deposited over land are deposited either on soil or on vegetation.

A box model describing nuclide migration between vegetation, top soil, bottom soil and ground water is used to describe the transport of deposited nuclides, see Figure 3-3.

Nuclides that are deposited on vegetation (or taken up by vegetation from soil) will potentially be ingested either directly by humans, or by grazing animals. The animals themselves, or animal products such as milk, will then contain radioactivity, which in turn may be ingested by humans when they consume meat and other animal products.

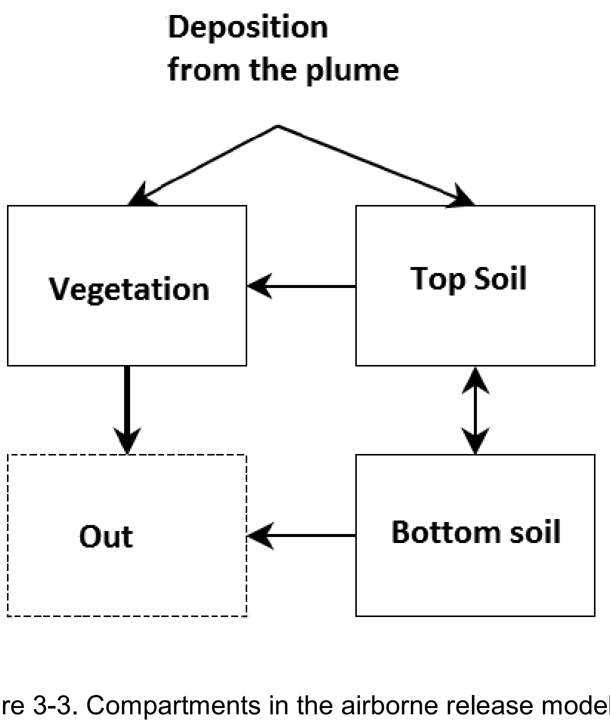

Figure 3-3. Compartments in the airborne release model.

Exposure pathways
Together, the atmospheric dispersion, the deposition, and the soil transport models contribute to identification of the exposure pathways.

The specific pathways that are used in the Ringhals model are:

- External exposure from radionuclides in the plume
- External exposure from radionuclides deposited on soil
- Internal exposure from inhalation of radionuclides in the plume
- Internal exposure from ingestion of:
    - Vegetables
    - Root vegetables
    - Cereals
    - Fruits
    - Berries
    - Mushrooms
    - Milk
    - Meat

All internal exposure is calculated as a committed dose over 50 years.

The exposure is calculated by assuming an annual amount of food ingested and an annual time spent indoors and outdoors. The specific parameter values depend on the age group considered.

Critical group
For all exposure pathways the location for calculation is chosen to yield the highest dose, but in a realistic manner. For example, the milk production is assumed to occur at the location where the highest deposition on grazing land occurs, and the root vegetables are assumed to come from the vegetable field with the highest deposition, etc.

3.4.2. Evaluation of the concentration and exposure levels

According to [9], the highest dose due to air release from R1 and R2 will be received in Börslund which is situated a few hundred meters south west of the NPP, see Figure 3-4.

Figure 3-4. Location of Börslund (critical group) (map source Google, TerraMetrics, Kartdata, 2018)

In Figure 3-5, the black line shows the resulting doses to the reference group (7-12 years old) in Börslund from releases to air from the entire Ringhals NPP (i.e. Ringhals 1-4). The blue, green and red lines shows corresponding doses from releases from Ringhals 1 and Ringhals 2, for children, infants and adults.

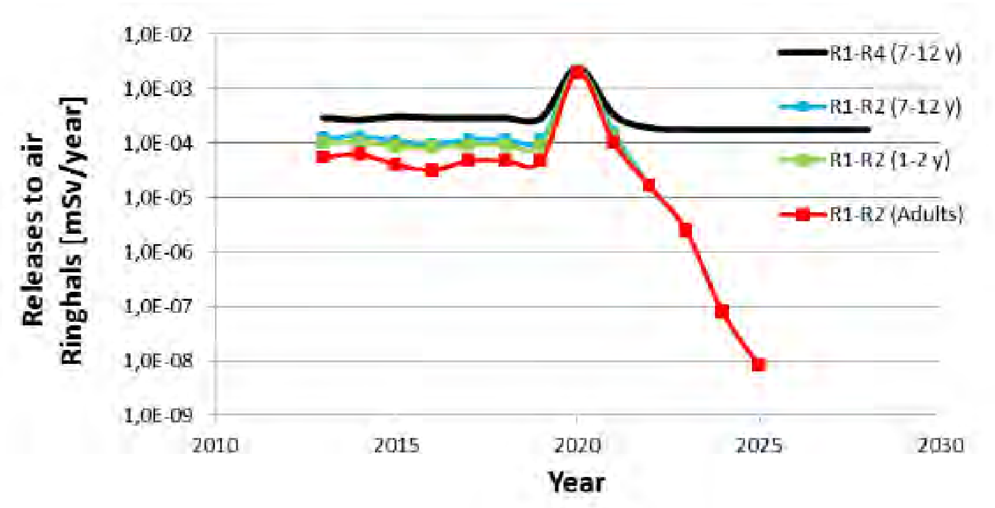

Figure 3-5. Resulting dose from conservative calculations of Co-60 release to air from R1 and R2 decommissioning [9].

The doses for 2013-2016 are extracted from Ringhals annual reports on discharges to air and water. For the years 2017-2020, doses are taken as the average from 2013-2016. The contribution from Ringhals 3 and Ringhals 4 is assumed to remain constant from 2021 onwards and is also based on discharges from 2013-2016. When decommissioning commences, the dose calculations for Ringhals 1 and Ringhals 2 are based on the source terms described above.

As can be seen in the figure, the highest dose occurs in 2020, which is the year when the R2 internal components segmentation is assumed to be initiated[16]. Since the release from the segmentation project is assumed to occur during the first year this causes a significant dose peak. It can also be noted that there is no similar clear peak during R1 internal components segmentation (2022). This is due both to R1 internal components having a lower Co-60 content than those of R2, but also due to the significantly higher release point of R1[17] (110 m in comparison to 20 m for R2).

Note also that HEPA-filtration in R2 has not been credited in the results, which means that the results are deemed to be conservative.

Reference [1] states that

If the assessed maximum exposure levels from releases in normal conditions to adults, children and infants in the vicinity of the plant are below 10 µSv per annum and there are no exceptional pathways of exposure, e.g. involving the export of foodstuffs, no data on effective doses in other affected Member States are required if doses to the reference groups in the vicinity of the plant are provided.

From [17] it can be concluded that the foodstuff contribution from Co-60 is in the order of 10% (depending on the age group) of the total dose from release to air. Therefore, the local foodstuff contribution is, using the conservative assumptions used here, in the order

[16] As mentioned previously this activity will be performed in 2022 and therefore the peak dose will in reality occur that year. Assuming an earlier date is conservative as radioactive decay reduces the dose (roughly halving every 5 years.)
[17] This can qualitatively be understood to occur since a high release point requires a longer distance for the center of the plume (which has the highest radioactivity concentration) to reach ground level. Therefore, in the vicinity of the plant a lower release height tends to result in a higher dose.

of 0.1 µSv/year (this can be compared to the average background radiation dose received by a person during one year, 1 mSv, i.e. 10 000 times higher). From this result it can be qualitatively understood that any radiation dose received by an individual in another member state due to the export of foodstuff from the vicinity of the Ringhals NPP would be significantly below this value, and therefore not of any relevance.

The total results are on the order of 1 µSv/year, i.e. below 10 µSv/year (equal to 1.0E-2 mSv/year in Figure 3-5) for all age groups, therefore not requiring further analysis of radiation dose to individuals in other member states.

4. Release from the installation of liquid radioactive effluents in normal conditions

4.1. Authorization procedure in force

See section 3.1 which deals with release to both air and water.

4.2. Technical aspects

4.2.1. Origin of the radioactive effluents

As long as there are still water filled systems, and hence liquids present in the facility, radioactive effluents can be expected to be roughly similar to those during operation, although likely with a lower volume. The full system decontamination will also give rise to liquid effluents.

When the full system decontamination has been done and systems have been emptied the main sources of liquid effluents are expected to be from uses of liquid decontamination techniques and other minor uses of liquids during the dismantling works.

4.2.2. Annual discharges expected during decommissioning

Dismantling and demolition activities are not expected to lead to significant amounts of liquid waste, and hence no significant release of effluents. Some continued releases can be expected for example before all systems have been emptied of water, during decontamination work etc.

It should be noted that release of liquid waste is monitored and requires manual action before discharge, and hence any significantly contaminated water can be rerouted to waste management before release.

For the purpose of this report it is assumed that release of liquid effluents will continue as during operation. This is likely to be conservative for the majority of the years of decommissioning. In Table 4-1 estimated annual discharges of Co-60 and tritium to water from Ringhals 1 and Ringhals 2 are presented. For releases to water, Co-60 and tritium are the main contributors to the dose at Ringhals. The tritium releases from Ringhals 1 and 2 are conservatively assumed to be constant and the values originate from average annual releases from Ringhals 1 and 2 during the time period 2013 to 2016. The assumed release of Co-60 also originates from historical annual releases (2013-2016) but is adjusted for radioactive decay after year 2021.

Table 4-1. Estimated annual discharge of Co-60 and H-3 to water from R1 and R2.

Initiation year	Co-60 Bq/y	H-3 Bq/y
2020	1.6E+08	5.6E+12
2021	1.6E+08	5.6E+12
2022	1.4E+08	5.6E+12
2023	1.2E+08	5.6E+12
2024	1.1E+08	5.6E+12
2025	9.4E+07	5.6E+12

Some previous experience at R1 though, indicates that the release to water during a year when a full system decontamination is performed can increase[18] [9]. The dose from the release to water, including increased activity due to system decontamination activities, is however insignificant compared to the dose from release to air. For this reason, this potential underestimate has no overall effect on the conclusion in this report.

4.3. Monitoring of discharges

Liquid from floor drains and other sources where contamination is not normally expected is released through pipes which are monitored online and automatically rerouted if the activity level reaches a specified threshold.

Other liquid waste that has been treated is collected in release tanks. Before discharge can take place, a representative sample is taken after mixing the contents of the release tank. This sample is first screened using a total gamma measurement. If the activity is below a set screening value it can be released. If activity exceeds the value then the water can either be sent back for treatment, or a nuclide-specific measurement can be performed to determine if it is possible to release.

If the radioactivity levels are below the release limits[19] the water is released to the coolant water channel. During the release a sample is taken and stored as part of the official regular discharge analysis reported to SSM.

If the water in the release tank were to exceed the release limit it is pumped back to the liquid waste treatment line for further processing.

4.3.1. Alarm levels and actions

The release system for R2 consists of three separate paths, each monitored with online measurement during release. Each of these is set to give an alarm if a certain cps-value[20] outside the release pipe is reached. This alarm automatically triggers the release valves to

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[18] Co-60 is one of the major contributors to the dose from releases to water at Ringhals. During a full system decontamination 1997 at Ringhals 1 the annual release of Co-60 increased by a factor of three compared to adjacent years. The dose due to releases of radioactive materials to water is several orders of magnitude lower compared to the releases to air during a normal year at Ringhals.

[19] During operation the release limits have been set in relation to Ringhals operational release target values. Such target values do not apply to decommissioning and therefore corresponding target values will be determined separately.

[20] During operation, the value for the release monitoring from the sump tank and monitoring tank is set at 1.8E4 counts per second (cps) outside the release pipe. During decommissioning, if necessary, the value will be revised to account for changes in the nuclide profile between operation and decommissioning.

close and thereby prevent the release and the water is instead redirected for further treatment.

At R1 the liquid from areas where contamination is normally not expected is monitored online during release. Like the system for R2 the specified alarm level will trigger a closing of the release valves and redirect the water for treatment.

4.4. Evaluation of transfer to man

4.4.1. Model description

This section describes the models used for the calculation of dose to a representative person in the vicinity of the plant from release of liquid effluents.

Note that this discussion is as far as possible kept on a qualitative level with the aim of describing an overview of how the calculations have been made. The models are described in detail in [11], [12], [13][14], [15], [16] and [17].

Release to recipient
Liquid effluent is assumed to be released to the cooling water canals which lead out to Kattegat, see chapter 1. While release actually occurs batch wise, the models are based on a continuous and constant release rate.

Dispersion
Dispersion in water is assumed to occur in a rectangular area with side lengths of 2.5 and 5 km. This is based on an assessment of seawater temperatures, where it has been shown that outside this area no increased temperature can any longer be observed. This in turn indicates that at this distance the effluent is well mixed with the surrounding waters.

In the dispersion model, the radionuclides are transported between different model compartments describing water, marine life, suspended material, upper sediment and lower sediment, see Figure 4-1.

Water, and suspended material in the water, leave the model area through the water turnover. I.e., as radiologically clean water flows into the model area an equal amount of contaminated water flows out taking radionuclides with it out of the model.

Transport of nuclides between the compartments is described by transfer functions which are described in detail in the above references.

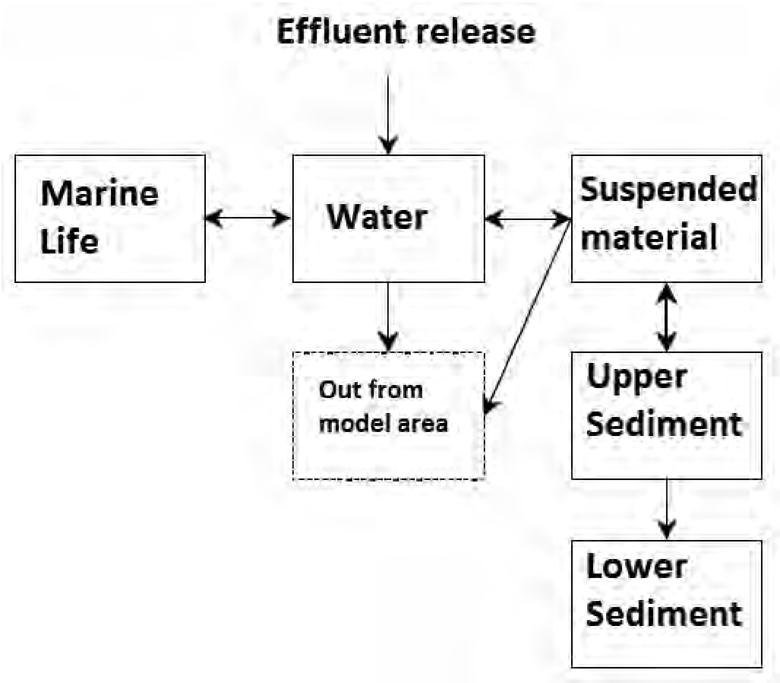

Figure 4-1. Compartments in the liquid effluents release model.

Exposure pathways
There are two exposure pathways included in the model:

- Ingestion of marine life (fish and crustaceans)
- External exposure from radiation at the beach.

In order to calculate the external exposure from a visit to the beach it has been assumed that one percent of the upper sediment constitutes the beach area.

The exposure is calculated by assuming an annual amount of marine life ingested and an annual time spent at the beach. The specific parameter values depend on the age group considered.

All internal exposure is calculated as a committed dose over 50 years.

4.4.2. Evaluation of the concentration and exposure levels

In [9] the resulting dose from release to water has been calculated to infants (1-2 y), children (7-12 y) and adults in Börslund. The result is presented in Figure 4-2.

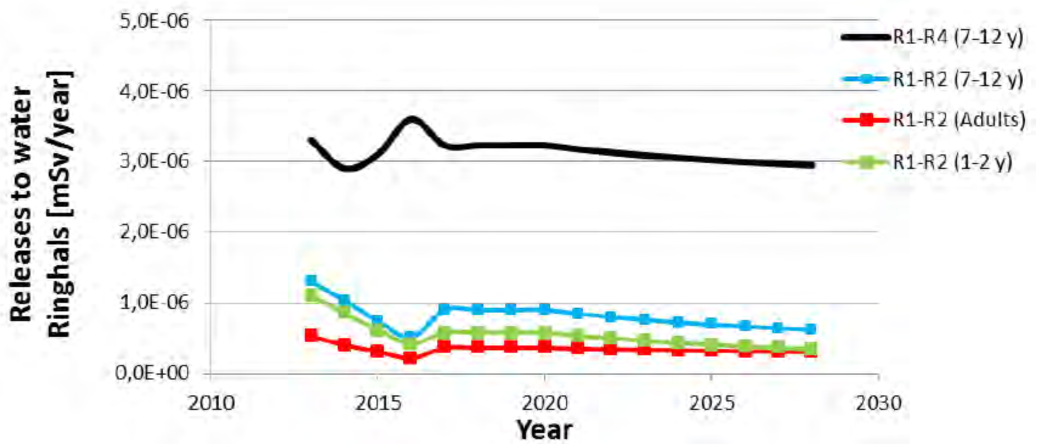

Figure 4-2. Resulting dose from calculation of Co-60 release to water from R1 and R2 decommissioning [9].

Reference [1] states that

If the assessed maximum exposure levels from releases in normal conditions to adults, children and infants in the vicinity of the plant are below 10 µSv per annum and there are no exceptional pathways of exposure, e.g. involving the export of foodstuffs, no data on effective doses in other affected Member States are required if doses to the reference groups in the vicinity of the plant are provided.

From [17] it can be concluded that the foodstuff contribution from Co-60 dominates the total dose from release to water. Therefore, the local foodstuff contribution is on the order of 1 nSv/year (this can be compared to the average background radiation dose received by a person during one year, 1 mSv, i.e., 1 000 000 times higher). From this result it can be qualitatively understood that any dose to another member state through the export of foodstuff would be significantly below this value, and therefore not of any relevance.

The result is approximately four orders of magnitude below the 10 µSv/year limit, and therefore does not require any further assessment of dose in other member states.

5. Disposal of solid radioactive waste from the installation

5.1. Solid radioactive waste

This chapter describes the Swedish requirements governing the management of radioactive solid waste and how Ringhals plans to manage the waste arising from the decommissioning of R1 and R2.

The licensees are responsible for the nuclear waste arising during operation and decommissioning of a nuclear facility. This responsibility ceases once the waste has been placed in a final repository that has been finally sealed.

The decommissioning of R1 and R2 will to a large extent produce waste forms similar to those produced during normal operation, although in significantly greater volumes. This means that the main challenge in adapting the waste management system to the decommissioning phase is primarily a logistical one, rather than a radiological one.

Sweden has an established radioactive waste management system for operational waste. This means that there is a vast amount of experience in producing packages containing operational waste suitable for disposal in approved and operating disposal facilities. The corresponding system for managing decommissioning waste is expected to be very similar.

The general sequence in all solid waste management performed in the decommissioning phase is summarized below:

1. Waste production. For example, by dismantling a section of a system or segmenting an internal reactor component.
2. Collection & Segregation, e.g. sorting the waste generated based on the intended waste type, contamination level, etc.
3. Treatment, e.g. decontamination, stabilisation, compaction etc.
4. Conditioning, e.g. packaging the waste in its final container. For some waste forms this includes solidification/encapsulation in e.g. a grouting material such as concrete.
5. Measurement of the radiological parameters, such as nuclide-specific contents, dose rates and surface contamination levels.
6. Storage until the waste package is ready to be accepted at a repository.
7. Transport to the repository (to external sites mainly through the SKB transport system).
8. Disposal.

Reference [2] requires the waste generator to produce a waste handling plan as well as a specific description of each waste type. Both the waste handling plan and the waste type descriptions (WTD) undergo safety reviews and form part of the safety analysis report for each facility where the waste is handled. This includes the facility of origin, in this case R1 and R2, as well as any offsite treatment facilities and the final repository. The WTD describes the handling sequence, from the time the waste is produced until it has been finally deposited in SFR (Final repository for short-lived radioactive waste) or SFL (Planned final repository for long-lived radioactive waste). The WTD lists the relevant waste acceptance criteria (WAC) together with a description of how compliance with the WAC is verified. Before a particular waste type can be produced and disposed of at the SFR-facility, the WTD must be approved by SSM.

The system for managing radioactive waste in Sweden is described in Appendix 1.

5.1.1. Radiological levels of solid radioactive waste and estimated amounts

The initial sorting of materials, buildings and site areas from a radiological and hazard perspective, as well as the estimated waste quantities, is derived from previous studies. Taking into account the information compiled in previous studies, operational records and other relevant information, the waste expected to arise during the decommissioning phase has been defined and quantified. The estimates of waste quantities will be subject to revision when results are obtained from the radiological survey activities that will continue while the decommissioning is carried out.

The waste is further divided into different waste streams: Metal, Combustible and Other radioactive solid waste (concrete, sand, biological shield, etc.).

Levels of radioactive waste are defined based on the content and half-life of the respective radionuclides in the waste.

The waste is generally divided in four levels of low and intermediate level radioactive waste. These levels dictate how radioactive waste is disposed of.

Short-lived very low-level waste (VLLW): Contains short-lived radionuclides with half-life less than 31 years. The dose rate on the waste container (and unshielded material) < 0.5 mSv/h. This waste is often secondary waste (such as gloves, protective clothing and equipment). This waste meets the acceptance criteria for final disposal in BLA (Rock vault for low level waste in SFR) or in a surface repository.

Short-lived low-level waste (LLW): Contains short-lived radionuclides with half-life less than 31 years. The dose rate on the waste container (and unshielded material) < 2 mSv/h. Long-lived radionuclides with half-life greater than 31 years are allowed in limited quantities. The waste meets the acceptance criteria for final disposal in BLA (Rock vault for low level waste in SFR).

Short-lived intermediate level waste (ILW Short-lived): Contains significant quantities of short-lived radionuclides with half-life less than 31 years. The dose rate on the waste container < 500 mSv/h for storage in the silo repository in SFR or < 100 mSv/h for storage in BMA (rock vault for intermediate level waste) in SFR. Long-lived radionuclides with half-life greater than 31 years are allowed in limited quantities[21]. The waste meets the acceptance criteria for final disposal in BMA or Silo.

Long-lived low and intermediate level waste: Contains long-lived radionuclides with half-life greater than 31 years and in significant quantities greater than the limits that apply to short-lived waste. No dose rate limit for disposal has been specified. The waste

[21] Limitations are specified in the Safety Analysis Report for SFR for the total quantity of long-lived nuclides stored in the rock vault. There are no specific nuclide limits imposed on individual waste packages.

will be disposed of in SFL (Final repository for long-lived waste), which is planned to open in 2045.

High-level waste refers to spent fuel and is not addressed in this report as the fuel will be removed before decommissioning commences.

Figure 5-1 depicts the quantities of the waste levels projected to arise during decommissioning of R1 and R2.

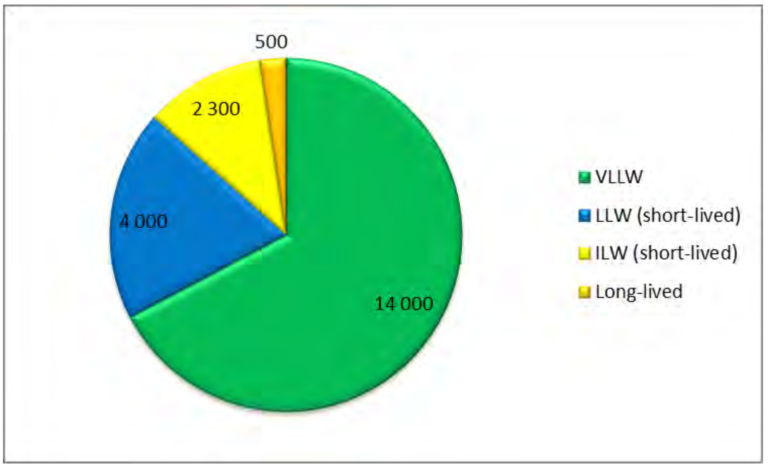

Figure 5-1. Decommissioning waste sorted by waste level (tons).

In Figure 5-2 the waste is divided into streams, and in Table 5-1 the waste is sorted according to stream and radioactivity level. The primary waste streams are described below.

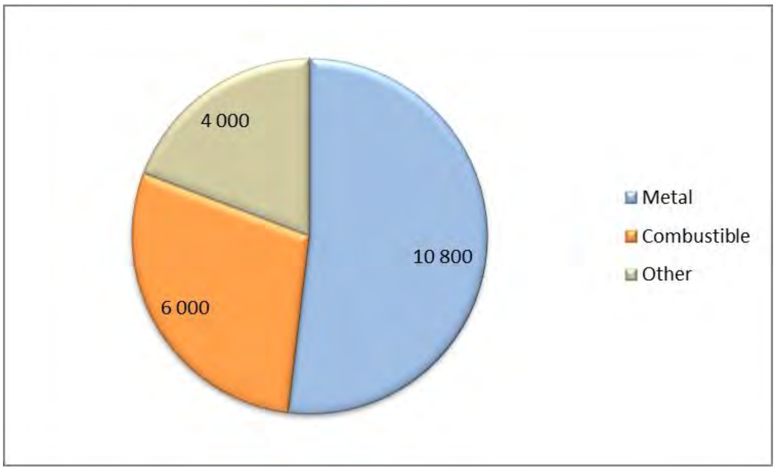

Figure 5-2. Waste sorted by waste stream (tons).

Table 5-1. Decommissioning waste from R1 and R2 (tons)

Waste	Metal	Combustible	Other	Total
VLLW	6 000	5 000	3 000	14 000
LLW	2 100	900	1 000	4 000
ILW (short-lived)	2 200	100	0	2 300
Long-lived	500	0	0	500
Total	10 800	6 000	4 000	20 800

Metallic waste
The metallic waste is comprised of both long-lived and short-lived radioactive waste, where the long-lived waste consists solely of the reactor internal components and parts of the reactor tanks that have been neutron-activated. The metallic waste originates from:

- Process systems
- Steel structures
- Ventilation systems
- Electrical equipment

Combustible Waste
During decommissioning, secondary waste will be generated from many operations that result in contaminated protective clothing, gloves, plastic, etc. The quantities of combustible waste are estimated based on similar work performed during annual outages at the Ringhals plant.

Other Solid Waste
Other radioactive solid waste is comprised primarily of concrete and sand. Concrete is present in conventional building material and as radiation shielding in the biological shield. Concrete ILW is expected to arise solely from the demolition of the biological shield. Sand from the gaseous waste system contributes with 700 tons of waste partially contaminated with noble gas daughter products.

5.1.2. Processing, packaging and disposal

Most requirements for processing and packaging of radioactive solid waste are determined by the intended disposal. If the waste will be disposed of in a repository then the waste acceptance criteria must be met, which may entail specific treatment and packaging. If the waste is intended for clearance then it may be subject to treatment before this can take place.

More than one disposal option may be applicable for several of the waste streams. The appropriate option is selected based on logistics, radiation protection, environmental impact and risks.

Processing
Radioactive solid waste is processed or treated with the objective of ensuring that it can either be cleared or rendered compliant with waste acceptance criteria for disposal. This treatment can also include volume reduction to maximize packing efficiency in the geological repository or surface repository. Processing includes segregation of waste in different streams and levels, segmentation of piping and components, decontamination,

solidification of particulate waste, such as ion exchange resin, fixing of loose surface contamination and compaction. Melting or incineration at an offsite facility is also possible.

Packaging

Packaging criteria for radioactive solid waste are dictated by both transportation and disposal requirements. In general, very low-level waste is disposed of onsite in rudimentary packaging such as plastic-wrapped bales or containers. Low-level waste is transported and disposed of in ISO containers. Short-lived intermediate level waste is packaged in steel or concrete moulds and long-lived intermediate level waste is packaged in steel tanks.

Disposal

In Sweden there are three options available for disposal of radioactive solid waste; disposal in a surface repository, disposal in the geological repository SFR, or disposal in the future geological repository SFL. For the waste that is suitable for clearance, the option is conventional disposal or recycling. Clearance is addressed in chapter 5.4.

For VLLW, the options[22] include disposal in surface repositories (see Figure 5-3), owned and managed on site. The contact dose rate of the waste must be less than 0.5 mSv/h and nuclide-specific activity limits stipulated in the surface repository license must not be exceeded. Both compactable and non-compactable waste can be deposited in these repositories. The repository area is monitored for at least 30 years by which time most of the radioactivity will have decayed.

The existing surface repository at Ringhals has limited capacity remaining and is currently licensed only for the disposal of operational waste. Ringhals intends to build a new surface repository for decommissioning which is currently in the pre-study phase to evaluate different design and location alternatives.

Waste potentially suitable for disposal in a future surface repository consists of mainly metals and combustible waste, with some concrete.

Swedish NPP operations also have access to a geological repository for the final disposal of short-lived low and intermediate level radioactive waste (SFR), built and operated by SKB (see Figure 5-4). This repository is planned to be extended in order to provide sufficient capacity in SFR for future decommissioning and operational waste. An application for this extension was submitted to the SSM in 2014 and is presently being reviewed. In the meantime, until the extension is completed, low level and intermediate level waste are temporarily being stored in an interim storage facility on site. SFR is planned to resume accepting short-lived low and intermediate level waste for disposal in 2030.

For long-lived low level and intermediate level radioactive waste, a geological repository (SFL) is planned to be in operation 2045. Long-lived radioactive waste needs to be stored in the interim, either on or offsite until SFL is available.

An overview of the Swedish system for managing radioactive waste is provided in Appendix 1.

[22] For waste with options the available routes as well as general decision criteria are given through the waste management plan and waste type descriptions.

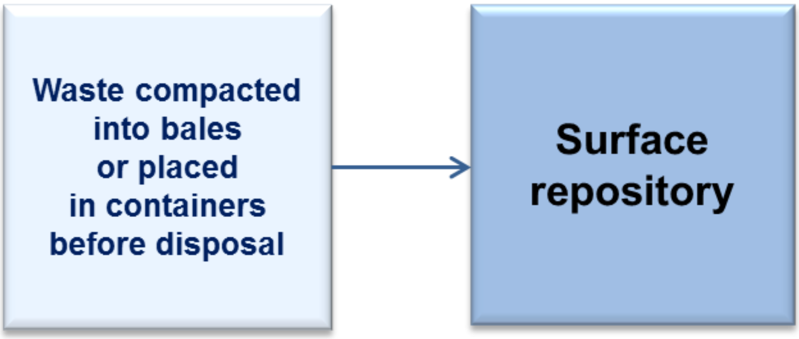

Figure 5-3. Management of VLLW - surface repository waste.

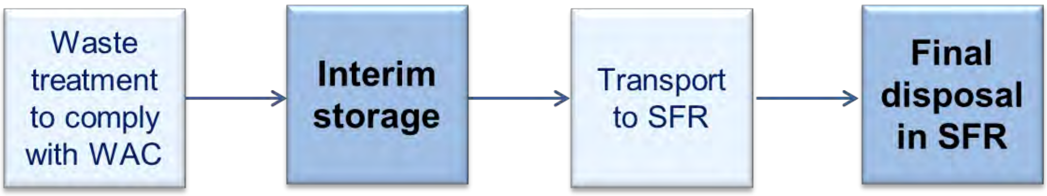

Figure 5.4. Management sequence for waste sent for geological disposal.

5.1.3. Interim storage onsite

Necessary interim storage of all low level and intermediate level radioactive waste produced during decommissioning is planned[23] to take place onsite at Ringhals. The unavailability of the SFR and SFL repositories requires the interim storage of short-lived waste until after 2030 when SFR is planned to open and of long-lived waste until 2045 when SFL is planned to commence operations.

Intermediate level waste produced from e.g. segmentation of reactor internal components, is planned for storage in the existing ILW Storage building, which is equipped with radiation shielding and is planned for further modifications to facilitate waste handling.

There are plans to expand the capacity for interim storage of low-level waste to supplement existing storage capacity at Ringhals. The interim storage facilities will be located within or near the existing waste management area at Ringhals where other waste facilities are situated in order to minimize unnecessary transport.

5.2. Radiological risks to the environment

The waste to be handled during the dismantling process consists of various radioactive components and materials. In order to reduce the activity level within the two units to the extent possible early in the process, early activities will involve segmentation and removal of the reactor internal components, which contain a large proportion (roughly

[23] It should be noted that as time progresses there might be other off-site facilities available for storage. If this is the case off-site storage could be possible in the future. Any such storage requires that the WTD is included in the SAR of that facility ensuring that safety is maintained.

99.9%) of the radionuclides that remain after the spent fuel is removed from the site. This approach of removing the most active systems early in the dismantling and safely store the radioactive waste results in a significant decrease in the radiological risk to the environment from the decommissioning activities.

Radiological risks to the environment during decommissioning consist of releases related to the handling of radioactive components and materials and accidents in the plant. An assessment of radiological consequences is presented in chapters 3, 4 and 6 in this report.

Precautions will be taken to prevent impact to the environment during dismantling and demolition. Examples of equipment that will be used during dismantling to mitigate risk include systems to detect releases (to water or air), controlled ventilation and fire detection equipment. These systems are already in place at the plant or new temporary solutions will be installed while dismantling is underway. Monitoring of onsite interim storage facilities for waste will be applied to the extent necessary depending on the type of waste stored and the associated risk for releases of radionuclides.

5.3. Off-site arrangements for the transfer of waste

In order to transfer waste from the decommissioning site to another location in Sweden, the first step is to verify that the waste meets the acceptance criteria established by the recipient. The recipient may be a final repository, a central storage or treatment facility or an offsite treatment facility, such as Cyclife's facilities in Sweden[24]. This verification is documented by a waste type description for each waste type, as mandated in the SSM regulations. These waste type descriptions form part of the safety analysis report for the receiving facility, thereby ensuring that safety is maintained when handling the waste.

The Swedish nuclear industry has an established transportation system that has been in use since the 1980s. This system is primarily based on transport by sea using SKB's specially designed ship m/s Sigrid, see Appendix 1. Intermediate level waste is normally transported by sea. Low level waste in ISO-containers can be transported by road. Shipments of radioactive waste in Sweden comply with the requirements stipulated in the European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR) and the International Maritime Dangerous Goods Code (IMDG).

SKB owns and maintains casks for the transport of intermediate level waste, each of which can hold multiple waste containers.

The shielded cask for off-site transportation of steel tanks containing SFL waste is under construction and is not yet available as a part of the system. Ringhals intends to store SFL waste at Ringhals until 2045.

The Studsvik site, approximately 100 km south west of Stockholm, provides an option for waste treatment of some components and waste streams. Here Cyclife Sweden provides waste treatment including volume reduction e.g. metal treatment by melting and thermal treatment of organic waste by incineration or pyrolysis. For transportation to and from the Studsvik site, both the sea route system and the roadway system are applicable.

[24] Cyclife Sweden AB is located at the Studsvik Industrial site outside Nyköping on the east coast of Sweden. Cyclife Sweden provides waste treatment including volume reduction e.g. metal treatment by melting and thermal treatment of organic waste by incineration or pyrolysis.

5.4. Release of materials from the requirements of the basic safety standard

National strategy, criteria and procedures for the release of contaminated and activated materials

SSM's regulations concerning the clearance of material, rooms, buildings and land used in activities involving ionizing radiation, SSMFS 2018:3, were published on May 31, 2018. The objective of the regulation is to enable rational processing and use of materials, rooms, buildings and land that could have been contaminated with radioactive substances. This regulation brings Swedish regulations into line with directives from EURATOM pertaining to clearance criteria.

The management of decommissioning waste from R1 and R2 will be based on the recommendations and guidelines for clearance developed jointly by nuclear power suppliers and other nuclear facility owners in Sweden.

Clearance levels established by competent authorities for disposal, recycling and reuse

In Sweden there are two types of clearance – general and conditional. General clearance entails that material and waste can be reused, recycled or deposited without imposing any further radiological control requirements. General clearance as per SSMFS 2018:3 applies to clearance levels for free use. As examples, the clearance limit for Co-60 and for Cs-137 is 0.1 Bq/g. Appendix 2 in the regulation lists the clearance level for radionuclides for systems, materials and components, and is provided in Appendix 2 of this report.

Conditional clearance as per SSMFS 2018:3 applies to some types of hazardous waste. For example, the clearance limit for Co-60 is 1 Bq/g and for Cs-137 10 Bq/g. Appendix 2 in this report also includes tables with the corresponding clearance levels for buildings and contaminated oil and other hazardous waste.

The regulations allow for the granting of exemptions to allow for the clearance of materials provided there is no unacceptable risk of radiation exposure to people or the environment. This type of clearance requires special permission from the SSM.

For buildings and areas, a clearance decision is made by SSM after application by the licensee (areas and buildings). Clearance of materials can be made directly by the licensee, based on a control program that is mandated by SSM.

Envisaged types and amounts of released materials.

It is estimated that the dismantling and demolition of R1 and R2 will require the clearance of 320 000 tons of material. Most of this material will consist of metal and concrete.

Clearance of materials can take place either directly after removal or after treatment. The nuclide-specific activity of the materials is measured to ensure that the levels comply with the requirements for clearance. See Figure 5-5.

Figure 5-5. Management of local clearance.

6. Unplanned releases of radioactive effluents

6.1. Review of accidents of internal and external origin which could result in unplanned releases of radioactive substances

The licensee of a nuclear facility must verify that the probability of serious accidents or incidents is low and that, should such an event nevertheless occur, the consequences to the environment and personnel are acceptable.

SSM regulation SSMFS 2018:1 stipulates that events and conditions that are significant for the radiation safety shall be identified and evaluated. The requirement also covers the decommissioning of the Nuclear Power Plant.

The safety analysis reports for operation of Ringhals 1 and 2 identifies and evaluates all events that are significant for the radiation safety during all modes of operation, including outage and releases from sources other than the nuclear fuel.

In section 0.5.2 some early decommissioning activities during the post-operation and care and maintenance phases are described. All such early decommissioning activities are subject to approval from SSM and will be performed in compliance with the safety analysis report and within the limits specified in the plant technical specifications for shut down reactor. This means that such activities will not affect the spent fuel or the fuel handling.

When the two units reach the decommissioning phase, it means that all the used nuclear fuel has been transported offsite, and the risk of major radioactive releases is thus minimized. Furthermore, there are no inherent driving mechanisms such as high temperatures or pressures and the amount of oils and chemicals is reduced, which reduces the risk of a fire that can result in the release of radioactivity.

The limiting events for decommissioning have been identified in [18], which is based on a methodology that can be described in the following steps.

1. Identification of sources of radioactive materials.
2. Mapping of events that may occur during decommissioning activities.
3. Identification of limiting events with regard to radioactive release.

The consequences of the identified limiting events are then evaluated. In order to verify that the releases cannot exceed the specified acceptance criteria for any identified event, the release from a bounding postulated hypothetical event is quantified. This postulate is highly conservative and effectively bounds all other identified events, see section 6.1.3.

Sections 6.1.1 and 6.1.2 in this report describe in more detail the most relevant internal and external events during dismantling and demolition activities, including waste management.

6.1.1. External events

The safety analysis reports for the plants in operation present the external events that are included in the design basis for the plants. In this report, information about the geological and hydrological situation at the NPP is described in section 1. The plant buildings are robust and designed to withstand external events and protect the plant safety functions. During decommissioning, the surrounding environment is protected by the integrity of the buildings and structures that contain radioactive material.

The analyses performed for power operation indicate that the effect on buildings from external events is limited and that there are no consequences on the surrounding environment. The following external events are relevant to consider for decommissioning. The results from the power operation analysis are considered to be relevant also for decommissioning. That is, if there are no radiological releases during power operation it is reasonable to assume that there is no consequence during decommissioning.

Extreme water levels
Since the ground level at the plants is higher than the extreme water levels, there is no consequence for the decommissioning project.

Extreme rain, snow or wind
The plant is designed against extreme precipitation and wind.

Tornado and generated missiles
A tornado-generated missile could lead to local damage, but no active equipment is necessary for mitigation.

Airplane crash
For normal power operation this event is classified as an extreme low risk event and therefore not analysed further[25].

Transport mishaps
Transport mishaps against buildings can cause local effects on weak buildings or some doors. The event could affect the ventilation systems but will not have any significant consequence to the environment.

External fire or explosion
An external explosion could lead to a loss of ventilation but will not have any consequence to the environment.

Loss of offsite power, lightning or other electromagnetic disturbances
The events could lead to a shutdown of active equipment such as ventilation and process systems. However, there is no need for active equipment to mitigate the event. The ongoing dismantling work will be stopped in a controlled manner without consequence to the environment.

[25] It is not necessary to analyze an airplane crash for decommissioning, but it could be noted that the consequences of an airplane crash are nevertheless covered by the conservative postulate in section 6.1.3.

Earthquake
Some of the service systems are not verified against earthquake and can therefore be assumed to be damaged by the event. The most limiting consequences identified from an earthquake are:

1. A crack in the drainage tank or the storage/delay tank for active gases at Ringhals 1.
2. Structural failure of the waste handling building and system for radioactive waste water.

If these consequences are combined with a fire, there could be a release to the environment, see sections 6.1.2 and 6.1.3 below.

6.1.2. Internal events

In order to find the most relevant events for decommissioning the relevant type of events from the safety analyses report for normal operation are analyzed further with the precondition that the units no longer contain any spent fuel and that there are no driving mechanisms such as high pressure or high temperatures. Based on the remaining sources of radioactive materials, the remaining systems in operation and expected manual actions during the decommissioning, the following events have been identified as relevant events which can lead to a release to the environment:

Fire
Fire has been evaluated for all parts of the plants that can contain radioactive material. In general, the amount of combustible radioactive material is small and only limited effects on the surrounding environment are expected from a fire. Fire will constitute the main risk of unplanned release to the environment. Precautions will be taken to prevent fire, such as removal of combustible materials, and to limit the consequences of a fire. In section 6.1.3, the bounding postulated fire event is presented.

Flooding
Internal flooding includes all types of water sources. The integrity of the plant is designed to withstand internal flooding. Systems that may contain significant liquid radioactive material are designed against potential leakages using for instance embankments or controlled drainage systems.

A malfunction, operator mishap or pipe break that causes leakage in a system containing radioactive material could lead to contamination of the plant. In general, such a leakage is contained within the plant, and only local consequences are expected. Normally, the radioactive material is contained in the fluid and no atmospheric dispersion is expected. The fluids can be handled as described in chapter 4. If the system contains aerosols there could be a small release to the surrounding environment.

Operator error, load drops and other malfunctions
The main consequence of an operator error can be a load drop or a transport mishap that leads to the loss of containment for radioactive material. Also, the integrity of a building, structures or systems could be affected. The buildings are designed to only sustain local damage in the event of a transport mishap or a load drop. The source with the highest concentration of radioactive material are the reactor internals. A load drop of heavy reactor internals or a load drop of a tank with segmented reactor internals could lead to local contamination and a loss of radiation protection. Even if it is a significant event, the release to the environment will be limited since the radioactive material is in the form of solid waste. However, if there is a malfunction and a subsequent fire involving the equipment for drying internal parts during segmentation, a release to the environment could occur. This type of malfunction is relevant during the period after the reactor internals are dismantled. All other malfunctions are covered by the potential events during earthquake or flooding that are described above.

Many of the above listed accidents are only applicable as long as there is water in the pools. When the fuel has been removed, the primary system will be decontaminated and all filters flushed and drained of radioactive materials before decommissioning commences. In order to further reduce the risk of leakage the plan is to eliminate the water as soon as possible, this is, when all the radioactive material in the spent fuel pools has been packed and transported to the chosen destination.

During interim storage of steel/concrete tanks containing segmented reactor internals no handling of the containers are expected. Thus, there are no unplanned releases of radioactive substances due to handling of the containers during the remaining period before transport to the final repository. During interim storage of other types of contaminated or activated components or material there is no additional handling of the containers/tanks planned before transport to SFR.

6.1.3. Bounding postulated event

Based on the activity content it can be assumed that occurrences with respect to the segmentation of internal parts is one of the worst-case scenarios together with fire in a fire cell containing radioactive materials.

Earthquake in combination with fire is another worst-case scenario with extremely low probability of occurring.

No flooding event has been identified as limiting since there is no flooding event that affects the integrity of a building.

The releases from the identified events above are very small. In order to overestimate the consequences a bounding postulated hypothetical event is designed to envelope all of the identified events. The bounding postulate is:

Fire affecting all non-induced activity[26] in the facility.

The postulate is physically not possible and highly conservative. The postulated fire covers all of the plant except the neutron-induced material in the reactor primary system, including internals, and the biological shield surrounding the reactor vessel. This means that the postulated fire is affecting all radioactive material that is contained in fluid systems, all radioactive material on the surfaces in the primary circuit and the reactor internals, ion exchange resins, the waste handling facility at Ringhals 1 and all contamination and corrosion products. The postulate effectively covers releases from an aggregate of all identified events.

6.2. Reference accident(s) taken into consideration by the competent national authorities for evaluating possible radiological consequences in the case of unplanned releases

The Swedish authorities have not specified any reference accidents during decommissioning, since this is the responsibility of the operator of the NPP, see section 6.1.

6.3. Evaluation of the radiological consequences of the reference accident(s)

6.3.1. Accidents entailing releases to atmosphere

In this section, results from evaluations of the reference accidents are given. Note that the discussion here is brief. For details the reader should consult the corresponding reference.

In [19] the dose consequences of a postulated fire affecting all non-induced activity in the facility is calculated.

The calculation uses a source term where the non-induced activity inventory in each facility is reduced by nuclide-specific fire release fractions. The released fraction is considered to be released at a height of 20 m during 1 hour, with a 2 m/s wind speed in Pasquill class F conditions with an inversion ceiling height of 100 m. Deposition speed is set at 0.002 m/s for particles, and 0.0002 m/s for organic iodine. The atmospheric dispersion is calculated using a Gaussian dispersion model described in [19] and [20].

The inventory is calculated for the start of decommissioning activities for R1 (2022-01-01) while the R2 inventory is calculated immediately at shutdown (2020-01-01).

Note that the height of 20 m is used for both R1 and R2 which is considered the most limiting release height for both plants since the dose is evaluated near the plant (the filtered ventilation system is assumed to fail during unplanned releases).

Exposure is calculated for adults, children and infants, and pathways included are external exposure from the passing plume, internal exposure from inhalation in the

[26] Note that while the induced activity level is much larger than the non-induced, the fact that the induced activity is bound in the crystal structures of the material means that the activity is not easily mobilized unless the material itself is being physically affected (e.g. by sawing). Such mobilizing activities are only performed while segmenting, and are therefore included in the planned activities described in chapters 3 and 4.

plume, and external exposure from deposited radionuclides on the ground. The latter exposure is integrated over a 30 day period.

The resulting effective dose is calculated at a distance of 200 m and is presented in Table 6-1.

Table 6-1. Maximum effective doses from the reference accident to individuals of different age groups 200 m from R1 and R2 respectively [19].

Age group	Ringhals 1 Total dose (mSv)	Ringhals 2 Total dose (mSv)
Adults	0.32	0.22
Children (7-12 y)	0.24	0.17
Infants (1-2 y)	0.21	0.15

In accordance with [1], which states

If the assessed maximum exposure levels from the reference accident to adults, children and infants in the vicinity of the plant are below 1 mSv and there are no exceptional pathways of exposure, e.g. involving the export of foodstuffs, no data on exposure levels in other affected Member States are required if exposure levels in the vicinity of the plant are provided.,

it can be concluded that the acceptance criterion is met even for this bounding postulated event.

6.3.2. Accidents entailing releases into an aquatic environment

No accident has been identified that could give a release to the aquatic environment during the decommissioning period that could affect the surrounding member states.

7. Emergency plans, agreements with other member states

In accordance with [1] the following discussion is valid for the time period when fuel is still located on-site at Ringhals.

7.1. Emergency preparedness

Ringhals has a plan for emergency preparedness [21] in the event of an emergency or a threat of an emergency at the facility. Since Ringhals 3 and 4 will continue to operate there will be no major changes in the emergency preparedness plan. The plan describes the entire scope of the emergency preparedness at Ringhals. It describes e.g. the organization, including available personnel with competence in relevant areas, alarm levels, instructions and routines to activate the alarm, instructions for informing relevant authorities, evacuation plans etc.

Relevant documentation including checklists for all functions in the emergency preparedness organization is also collected in a handbook that is available to the Emergency Preparedness Management. The handbook is well known and training is performed regularly.

Ringhals uses four different emergency levels depending on the situation.

1. Information level – This level is initiated in case of events where external authorities or Ringhals management may require immediate information on the event. Note that this level can be initiated by a wide range of events, e.g. a non-radiological accident, a large demonstration etc.
2. Assembly of Ringhals emergency preparedness organization – This level is initiated by events where a future threat to the reactor safety can be expected. The command center is manned by the emergency preparedness organization. Authorities are notified of the event. Examples include fuel handling accidents, a larger fire in an area containing safety systems, etc.
3. Increased preparedness – This level is initiated after an event that threatens the safety of the surrounding areas. The command center is manned by the emergency preparedness organization. Authorities are alarmed and man their emergency staff. Examples include dose rates, temperatures or pressures above given threshold values in certain areas, power disruptions etc..
4. Failure alarm – This level is initiated when a release is in progress that require protective measures in the vicinity of the plant, or when such a release cannot be ruled out to occur within the next 12 hours. The command center is manned by the emergency preparedness organization. Authorities are alarmed and man their emergency staff. Sound alarms to the surrounding areas are sounded. Examples include dose rates, temperatures or pressures above given threshold values in certain areas, e.g. in the surrounding areas or in the stack.

In accordance with the Swedish Regulation on Civil Protection, the County Administrative Board of Halland has established an emergency plan [22] for emergencies at Ringhals. The plan covers organization, liaison with other authorities and the station operator, where and how to measure radioactivity, how to handle public information, personnel and material resources available in the county, and methods of decontamination.

As established by ordinance 2003:789 there are preparedness zones established around the Swedish nuclear power plants, including Ringhals. An inner preparedness zone of about 12-15 km contain systems for in- and outdoor alarms, preparedness information to inhabitants, iodine tablets in households etc. An outer indication zone reaching about 50 km from the plant contain planned measures for monitoring of radiation for further decision making.

7.2. Agreements with other member states

Sweden has signed international and bilateral agreements on a national and official level concerning the early notification and subsequent information in the event of a nuclear energy accident. The Swedish Meteorological and Hydrological Institute (SMHI), which is manned around the clock, receives notifications of accidents abroad. The Swedish Radiation Safety Authority SSM, also manned around the clock, is responsible for disseminating the information nationally, and also for sending information to other countries in the event of an accident in Sweden. The most important agreements are:

- The binding EU directive on early notification (ECURIE, European Community Urgent Radiological Information Exchange) requires that a warning must be given if measures are adopted for protection of the domestic population. (ECURIE is the interface to the EU early notification and information exchange system for radiological emergencies.)
- Bilateral national agreements with Norway, Denmark, Finland, Germany, Russia, and Ukraine on warnings of accidents.
- The IAEA convention EMERCON on warnings of accidents if another country might be affected by release.

8. Environmental monitoring

When reading this section it should be kept in mind that during decommissioning of R1 and R2, there will still be two operational units (R3 and R4) that continue normal operation at the Ringhals site. The monitoring program is site-specific, and for this reason the monitoring program will not change from a radiation safety perspective.

The purpose of the radiological environmental monitoring program, described from a practical point of view in [23], is to examine the impact on the environment on account of the operation of NPPs as well as activities related to decommissioning. At the Ringhals site there will be operating plants as well as plants undergoing decommissioning.

As a complement to the monitoring of effluent release that is performed at the plant (see chapter 2, 3 and 4), the level of radioactivity in the vicinity of the NPP is also monitored. Through this monitoring program any potential larger unregistered discharge can be detected. Long term monitoring of radionuclides in the environment also produces basic data enabling estimation of potential effects on biological life in the recipient. The results can be used for informing the public and as a basis for international reporting and other collaboration in the environmental area.

The Swedish Radiation Safety Authority (SSM) requires nuclear facilities to monitor the environment in accordance with a program specified by SSM in the regulation on Protection of Human Health and the Environment in connection with Discharges of Radioactive Substances from certain Nuclear Facilities, SSMFS 2008:23. For decommissioning there are similar conditions in [2]. At present the SSI report 2004:15, [24], describes the content of the environmental monitoring program for the four NPPs in Sweden, as well as nuclear activities in Studsvik and the fuel fabrication facility in Västerås. The main focus is on biota, but also water, atmospheric precipitation, digested sludge and sediments are included.

Reference [24] defines selection of samples and their locations, preparation, analyses, evaluation and reporting. The samples include e.g. moss, apples, seabed sediment, locally produced milk, fish and meat, and the samples are analyzed by nuclide specific gamma spectroscopy using High Purity Germanium detectors (HPGe). The main radionuclides are Cs-137 and Co-60 as well as naturally occurring radionuclides, mainly K-40. The activity levels and the detection limits of specified radionuclides are reported to SSM. Sampling is performed by the Swedish University of Agricultural Sciences (SLU). Sample preparation and analyses are performed at Ringhals, in accordance with the guidelines developed by SSM for environmental monitoring. Meteorological data are continuously recorded.

The monitoring program consists of two different parts: one annual program and one extended program. The annual program enables detection of changes in the environment in the short term, but also to detect trends on a longer time-scale. The extended program, conducted every four years, covers a wider geographic area. The sampling locations are shown in Figure 8-1. Each sampling location has a specified season for taking the sample, and which species to sample [24].

There is also a network of area TL-dosimeters placed both on and outside the Ringhals site [25]. The outer area dosimeters are placed so that there is one monitoring location in every 30 degree sector on land at about one km distance from the center of the plant. In each location there are four dosimeters, two that are evaluated annually, and two that are evaluated quarterly. Each dosimeter contains 4 LiF-tablets.

The Ringhals site is also monitored by a network of permanent continuous online gamma dose rate detectors (Geiger-Müller detectors). The location of these detectors have been chosen with respect to the flow of personnel and traffic at the site.

SSM also has a regional measuring system that consists of permanent measuring stations around NPPs. The system consists of measuring stations (GM-tubes) at 5 km distance and at 10 km distance from the plants. The purpose of this system is to detect and follow-up major emissions during accident or large failure conditions.

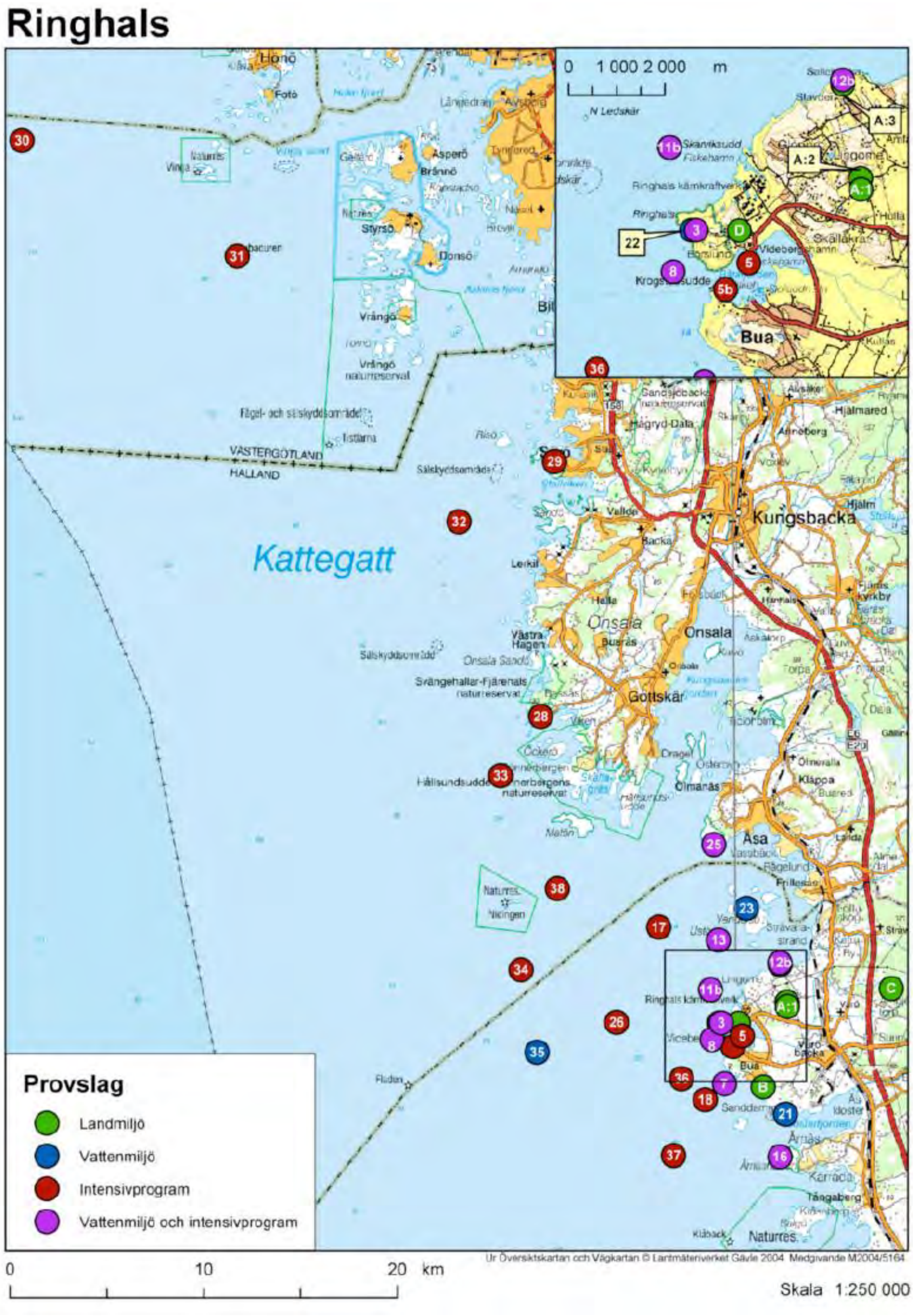

Figure 8-1. Sampling locations for the Ringhals environmental monitoring program

(Green: Land, Blue: Water, Red: Extended program, Purple: Water extended program) [24].

9. References

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Appendices

Appendix 1 The Swedish system for the disposal of spent nuclear fuel and radioactive waste

A law was enacted in Sweden in the 1970s stipulating that anyone who produces electricity using nuclear power must also manage and dispose of the waste. The nuclear power companies in Sweden therefore jointly established the Swedish Nuclear Fuel and Waste Management Company, SKB, with the task of managing and disposing of all radioactive waste from Swedish NPPs.

Figure A1-1 shows the Swedish system for radioactive waste disposal. The system for dealing with Swedish radioactive waste comprises a number of facilities that together provide a safe chain. The first links in this chain were already in place in the early 1980s, others still have to be constructed. The radioactivity level of the waste determines how it is managed. Radioactive waste from sources other than NPPs such as that produced in hospitals and industry must also be safely disposed of.

Transport by sea

Waste from nuclear power stations is transported by SKB's specially built vessel M/S Sigrid.

Central Interim Storage Facility for Spent Nuclear Fuel (Clab)

Today all spent nuclear fuel produced by Swedish nuclear power stations so far, is in interim storage in Clab outside Oskarshamn. It is stored in pools located in rock vaults 25–30 meters underground and is under continuous surveillance and control. Clab has been operating since 1985.

Clab is not a final repository, so after interim storage, the waste will be moved to the Spent Fuel Repository that SKB plans to construct at Forsmark.

Final Repository for Short-lived Radioactive Waste (SFR)

This is where operational waste from NPPs, which includes used protective clothing, replaced components and filtering materials that have been used to decontaminate reactor water, is deposited. Radioactive waste from hospitals, industry and research is also disposed of here. The repository is located at Forsmark in bedrock about 50 meters below sea level.

A license application has been submitted to the authorities in order to extend SFR, primarily to make room for decommissioning waste.

Final Repository for Spent Nuclear Fuel at Forsmark

The planned Spent Fuel Repository forms the last link in the chain when it comes to dealing with spent nuclear fuel. It will be deposited there in sealed copper canisters placed in rock vaults and surrounded by bentonite clay at a depth of 500 meters in the rock.

Encapsulation plant at Oskarshamn

After interim storage the spent nuclear fuel will be sealed into canisters and SKB plans to construct an encapsulation plant at Oskarshamn, which will be designated Clink, a joint facility with the central interim storage facility for spent fuel (Clab). The canisters will be made of copper with inserts of nodular cast iron and will each contain about 2 tons of waste.

Final Repository for Long-lived Radioactive Waste (SFL)

SKB is also planning a final repository for long-lived radioactive waste, SFL. This project has not, however, progressed as far as the others.

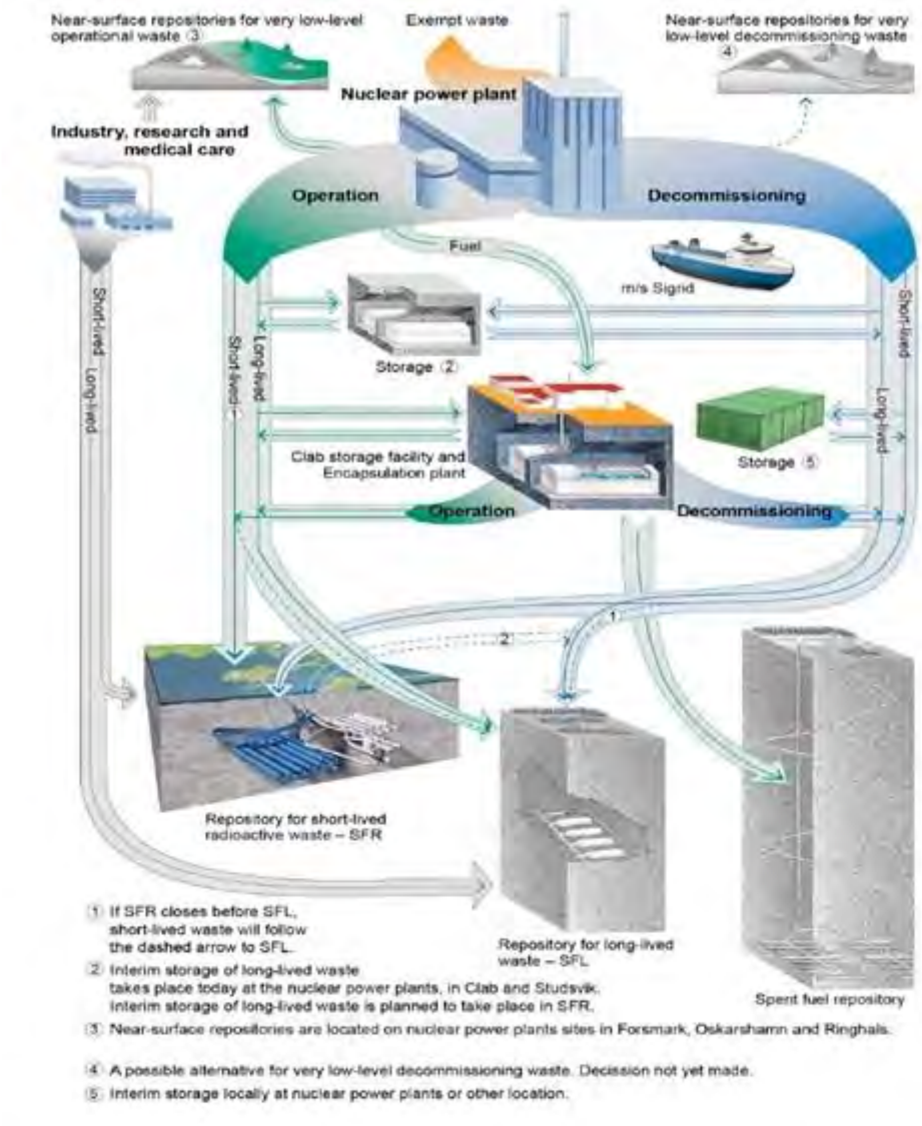

Figure A1-1. The Swedish system for radioactive waste disposal.

Appendix 2 – Clearance levels

The information in this appendix has been excerpted and translated from the Swedish Radiation Authority's regulation concerning clearance of materials, buildings and land, SSMFS 2018:3.

SSMFS 2018:3 Appendix 2 - Values for exemption or clearance levels

Applies to materials other than radioactive contaminated soil or sediment that has been excavated, unless samples are collected. (+) designates parent nuclides. See table below.

Radionuclide	Level (Bq/g)
H-3	100
Be-7	10
C-14	1
F-18	10
Na-22	0.1
Na-24	1
Si-31	1 000
P-32	1 000
P-33	1 000
S-35	100
Cl-36	1
Cl-38	10
K-40	1
K-42	100
K-43	10
Ca-45	100
Ca-47	10
Sc-46	0.1
Sc-47	100
Sc-48	1
V-48	1
Cr-51	100
Mn-51	10
Mn-52	1
Mn-52m	10
Mn-53	100
Mn-54	0.1
Mn-56	10
Fe-52 (+)	10
Fe-55	1 000
Fe-59	1
Co-55	10
Co-56	0.1

Radionuclide	Level (Bq/g)
Co-57	1
Co-58	1
Co-58m	10 000
Co-60	0.1
Co-60m	1 000
Co-61	100
Co-62m	10
Ni-59	100
Ni-63	100
Ni-65	10
Cu-64	100
Zn-65	0.1
Zn-69	1 000
Zn-69m (+)	10
Ga-72	10
Ge-71	10 000
As-73	1 000
As-74	10
As-76	10
As-77	1 000
Se-75	1
Br-82	1
Rb-86	100
Sr-85	1
Sr-85m	100
Sr-87m	100
Sr-89	1 000
Sr-90 (+)	1
Sr-91 (+)	10
Sr-92	10
Y-90	1 000
Y-91	100
Y-91m	100
Y-92	100

Radionuclide	Level (Bq/g)
Y-93	100
Zr-93	10
Zr-95 (+)	1
Zr-97 (+)	10
Nb-93m	10
Nb-94	0.1
Nb-95	1
Nb-97m (+)	10
Nb-98	10
Mo-90	10
Mo-93	10
Mo-99 (+)	10
Mo-101 (+)	10
Tc-96	1
Tc-96m	1 000
Tc-97	10
Tc-97m	100
Tc-99	1
Tc-99m	100
Ru-97	10
Ru-103 (+)	1
Ru-105 (+)	10
Ru-106 (+)	0.1
Rh-103m	10 000
Rh-105	100
Pd-103 (+)	1 000
Pd-109 (+)	100
Ag-105	1
Ag-108m (+)	0.1
Ag-110m (+)	0.1
Ag-111	100
Cd-109 (+)	1
Cd-115 (+)	10
Cd-115m (+)	100
In-111	10
In-113m	100
In-114m (+)	10
In-115m	100
Sn-113 (+)	1
Sn-125	10
Sb-122	10
Sb-124	1
Sb-125 (+)	0.1

Radionuclide	Level (Bq/g)
Te-123m	1
Te-125m	1 000
Te-127	1 000
Te-127m (+)	10
Te-129	100
Te-129m (+)	10
Te-131	100
Te-131m (+)	10
Te-132 (+)	1
Te-133	10
Te-133m	10
Te-134	10
I-123	100
I-125	100
I-126	10
I-129	0.01
I-130	10
I-131	10
I-132	10
I-133	10
I-134	10
I-135	10
Cs-129	10
Cs-131	1 000
Cs-132	10
Cs-134	0.1
Cs-134m	1 000
Cs-135	100
Cs-136	1
Cs-137 (+)	0.1
Cs-138	10
Ba-131	10
Ba-140	1
La-140	1
Ce-139	1
Ce-141	100
Ce-143	10
Ce-144 (+)	10
Pr-142	100
Pr-143	1 000
Nd-147	100
Nd-149	100
Pm-147	1 000

Radionuclide	Level (Bq/g)
Pm-149	1 000
Sm-151	1 000
Sm-153	100
Eu-152	0.1
Eu-152m	100
Eu-154	0.1
Eu-155	1
Gd-153	10
Gd-159	100
Tb-160	1
Dy-165	1 000
Dy-166	100
Ho-166	100
Er-169	1 000
Er-171	100
Tm-170	100
Tm-171	1 000
Yb-175	100
Lu-177	100
Hf-181	1
Ta-182	0.1
W-181	10
W-185	1 000
W-187	10
Re-186	1 000
Re-188	100
Os-185	1
Os-191	100
Os-191m	1 000
Os-193	100
Ir-190	1
Ir-192	1
Ir-194	100
Pt-191	10
Pt-193m	1 000
Pt-197	1 000
Pt-197m	100
Au-198	10
Au-199	100
Hg-197	100
Hg-197m	100
Hg-203	10
Tl-200	10

Radionuclide	Level (Bq/g)
Tl-201	100
Tl-202	10
Tl-204	1
Pb-203	10
Pb-210 (+)	0.01
Bi-206	1
Bi-207	0.1
Bi-210	10
Po-203	10
Po-205	10
Po-207	10
Po-210	0.01
At-211	1 000
Ra-223 (+)	1
Ra-224 (+)	1
Ra-225	10
Ra-226 (+)	0.01
Ra-227	100
Ra-228 (+)	0.01
Ac-227 (+)	0.01
Th-226	1 000
Th-227	1
Th-228 (+)	0.1
Th-229	0.1
Th-230	0.1
Th-231	100
Th-232 (+)	0.01
Th-234 (+)	10
Pa-230	10
Pa-231	0.01
Pa-233	10
U-230	10
U-231	100
U-232 (+)	0.1
U-233	1
U-234	1
U-235 (+)	1
U-236	10
U-237	100
U-238 (+)	1
U-239	100
U-240 (+)	100
Np-237 (+)	1

Radionuclide	Level (Bq/g)
Np-239	100
Np-240	10
Pu-234	100
Pu-235	100
Pu-236	1
Pu-237	100
Pu-238	0.1
Pu-239	0.1
Pu-240	0.1
Pu-241	10
Pu-242	0.1
Pu-243	1 000
Pu-244 (+)	0.1
Am-241	0.1
Am-242	1 000
Am-242m (+)	0.1
Am-243 (+)	0.1
Cm-242	10
Cm-243	1
Cm-244	1

Radionuclide	Level (Bq/g)
Cm-245	0.1
Cm-246	0.1
Cm-247 (+)	0.1
Cm-248	0.1
Bk-249	100
Cf-246	1 000
Cf-248	1
Cf-249	0.1
Cf-250	1
Cf-251	0.1
Cf-252	1
Cf-253	100
Cf-254	1
Es-253	100
Es-254 (+)	0.1
Es-254m (+)	10
Fm-254	10 000
Fm-255	100

Decay products that have been taken into account when determining values for radionuclides marked with (+) are listed in the table below.

Parent nuclide	Decay product(s)
Fe-52	Mn-52m
Zn-69m	Zn-69
Sr-90	Y-90
Sr-91	Y-91m
Zr-95	Nb-95
Zr-97	Nb-97m, Nb-97
Nb-97m	Nb-97
Mo-99	Tc-99m
Mo-101	Tc-101
Ru-103	Rh-103m
Ru-105	Rh-105m
Ru-106	Rh-106
Pd-103	Rh-103m
Pd-109	Ag-109m
Ag-108m	Ag-108
Ag-110m	Ag-110
Cd-109	Ag-109m
Cd-115	In-115m

Parent nuclide	Decay product(s)
Cd-115m	In-115m
In-114m	In-114
Sn-113	In-113m
Sb-125	Te-125m
Te-127m	Te-127
Te-129m	Te-129
Te-131m	Te-131
Te-132	I-132
Cs-137	Ba-137m
Ce-144	Pr-144, Pr-144m
Pb-210	Bi-210, Po-210
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207, Po-211
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214
Ra-228	Ac-228
Ac-227	Th-227, Fr-223, Ra-223, Rn-219, Po-215, Pb-211, Bi-211, Tl-207, Po-211
Th-228	Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Th-232	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Th-234	Pa-234m, Pa-234
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
U-235	Th-231
U-238	Th-234, Pa-234m, Pa-234
U-240	Np-240m, Np-240
Np-237	Pa-233
Pu-244	U-240, Np-240m, Np-240
Am-242m	Np-238
Am-243	Np-239
Cm-247	Pu-243
Es-254	Bk-250
Es-254m	Fm-254

SSMFS 2018:3 Appendix 3 - Clearance levels for oil and other hazardous waste to be sent for incineration or disposal

Radionuclide	Clearance level (Bq/g)
H-3	1 000
Be-7	100
C-14	100
Na-22	1
P-32	1 000
P-33	1 000
S-35	1 000
Cl-36	10
K-40	10
Ca-45	1 000
Ca-47	10
Sc-46	1
Sc-47	100
Sc-48	1
V-48	1
Cr-51	100
Mn-52	1
Mn-53	10 000
Mn-54	1
Fe-55	1 000
Fe-59	1
Co-56	1
Co-57	10
Co-58	1
Co-60	1
Ni-59	1 000
Ni-63	1 000
Zn-65	10
Ge-71	10 000
As-73	1 000
As-74	10
As-76	10
As-77	1 000
Se-75	10
Br-82	1
Rb-86	100
Sr-85	10
Sr-89	100

Radionuclide	Clearance level (Bq/g)
Sr-90 (+)	10
Y-90	1 000
Y-91	100
Zr-93	100
Zr-95 (+)	1
Nb-93m	1 000
Nb-94	1
Nb-95	10
Mo-93	100
Mo-99 (+)	10
Tc-96	1
Tc-97	100
Tc-97m	100
Tc-99	10
Ru-97	10
Ru-103 (+)	10
Ru-106 (+)	10
Rh-105	100
Pd-103 (+)	1 000
Ag-105	10
Ag-108m (+)	1
Ag-110m (+)	1
Ag-111	100
Cd-109 (+)	100
Cd-115 (+)	10
Cd-115m (+)	100
In-111	10
In-114m (+)	10
Sn-113 (+)	10
Sn-125	10
Sb-122	10
Sb-124	1
Sb-125 (+)	10
Te-123m	10
Te-125m	1 000
Te-127m (+)	100
Te-129m (+)	100
Te-131m (+)	10
Te-132 (+)	1

Radionuclide	Clearance level (Bq/g)
I-125	10
I-126	10
I-129	1
I-131 (+)	10
Cs-129	10
Cs-131	1 000
Cs-132	10
Cs-134	1
Cs-135	100
Cs-136	1
Cs-137 (+)	10
Ba-131	10
Ba-140	1
La-140	1
Ce-139	10
Ce-141	100
Ce-143	10
Ce-144 (+)	100
Pr-143	1 000
Nd-147	100
Pm-147	1 000
Pm-149	1 000
Sm-151	1 000
Sm-153	100
Eu-152	1
Eu-154	1
Eu-155	100
Gd-153	100
Tb-160	1
Dy-166	100
Ho-166	100
Er-169	1 000
Tm-170	100
Tm-171	1 000
Yb-175	100
Lu-177	100
Hf-181	10
Ta-182	1
W-181	100
W-185	1 000
Re-186	1 000
Os-185	10

Radionuclide	Clearance level (Bq/g)
Os-191	100
Os-193	100
Ir-190	1
Ir-192	1
Pt-191	10
Pt-193m	1 000
Au-198	10
Au-199	100
Hg-197	100
Hg-203	10
Tl-200	10
Tl-201	100
Tl-202	10
Tl-204	100
Pb-203	10
Pb-210 (+)	0.1
Bi-206	1
Bi-207	1
Bi-210	100
Po-210	0.1
Ra-223 (+)	10
Ra-224 (+)	10
Ra-225	10
Ra-226 (+)	0.1
Ra-228 (+)	0.1
Ac-227 (+)	0.1
Th-227	10
Th-228 (+)	1
Th-229 (+)	1
Th-230	1
Th-231	1 000
Th-232 (+)	0.1
Th-234 (+)	100
Pa-230	10
Pa-231	0.1
Pa-233	10
U-230 (+)	10
U-231	100
U-232 (+)	1
U-233	10
U-234	10
U-235 (+)	10

Radionuclide	Clearance level (Bq/g)
U-236	10
U-237	100
U-238 (+)	10
Np-237 (+)	1
Np-239	100
Pu-236	1
Pu-237	100
Pu-238	1
Pu-239	1
Pu-240	1
Pu-241	10
Pu-242	1
Pu-244 (+)	1
Am-241	1
Am-242m (+)	1
Am-243 (+)	1
Cm-242	10
Cm-243	1

Radionuclide	Clearance level (Bq/g)
Cm-244	1
Cm-245	1
Cm-246	1
Cm-247 (+)	1
Cm-248	1
Bk-249	100
Cf-246	100
Cf-248	10
Cf-249	1
Cf-250	1
Cf-251	1
Cf-252	1
Cf-253 (+)	10
Cf-254	1
Es-253	10
Es-254 (+)	1
Es-254m (+)	10

Decay products that have been taken into account when determining values for radionuclides marked with (+) are listed in the table below.

Parent nuclide	Decay product(s)
Sr-90	Y-90
Zr-95	Nb-95m
Mo-99	Tc-99m
Ru-103	Rh-103m
Ru-106	Rh-106
Pd-103	Rh-103m
Ag-108m	Ag-108
Ag-110m	Ag-110
Cd-109	Ag-109m
Cd-115	In-115m
Cd-115m	In-115m
In-114m	In-114
Sn-113	In-113m
Sb-125	Te-125m
Te-127m	Te-127
Te-129m	Te-129
Te-131m	Te-131
Te-132	I-132
I-131	Xe-131m
Cs-137	Ba-137m
Ce-144	Pr-144, Pr-144m
Pb-210	Bi-210, Po-210
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207, Po-211
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214
Ra-228	Ac-228
Ac-227	Th-227, Fr-223, Ra-223, Rn-219, Po-215, Pb-211, Bi-211, Tl-207, Po-211
Th-228	Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Tl-209, Pb-209
Th-232	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Th-234	Pa-234m, Pa-234
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208

Parent nuclide	Decay product(s)
U-235	Th-231
U-238	Th-234, Pa-234m, Pa-234
Np-237	Pa-233
Pu-244	U-240, Np-240m, Np-240
Am-242m	Np-238
Am-243	Np-239
Cm-247	Pu-243
Cf-253	Cm-249
Es-254	Bk-250
Es-254m	Fm-254

SSMFS 2018:3 Appendix 4 - Clearance levels for buildings.
Clearance levels for use refers to buildings to be cleared for continued use, and Clearance levels for dismantling refers to buildings cleared provided that they are dismantled.

Radionu-clide	Clear-ance level for use (kBq/m²)	Clearance level for dismantling (kBq/m²)
H-3	100 000	100 000
C-14	10 000	100 000
Na-22	10	100
S-35	10 000	1 000 000
Cl-36	1 000	1 000
K-40	100	100
Ca-45	10 000	1 000 000
Sc-46	10	100
Mn-53	100 000	100 000
Mn-54	10	100
Fe-55	100 000	100 000
Co-56	10	100
Co-57	100	1 000
Co-58	100	100
Co-60	10	10
Ni-59	1 000	1 000 000
Ni-63	100 000	1 000 000
Zn-65	10	100
As-73	10 000	100 000
Se-75	100	1 000
Sr-85	100	1 000
Sr-90 (+)	1 000	1 000
Y-91	10 000	1 000 000
Zr-93	10 000	10 000
Zr-95 (+)	10	100

Radionuclide	Clearance level for use (kBq/m²)	Clearance level for dismantling (kBq/m²)
Nb-93m	10 000	1 000 000
Nb-94	10	100
Mo-93	1 000	10 000
Tc-97	1 000	10 000
Tc-97m	1 000	10 000
Tc-99	1 000	1 000
Ru-106 (+)	100	1 000
Ag-108m	10	100
Ag-110m	10	100
Cd-109 (+)	1 000	100 000
Sn-113 (+)	100	1 000
Sb-124	10	100
Sb-125 (+)	10	100
Te-123m	100	1 000
Te-127m	1 000	100 000
I-125	1 000	100 000
I-129	100	100
Cs-134	10	100
Cs-135	10 000	100 000
Cs-137 (+)	10	100
Ce-139	100	1 000
Ce-144 (+)	100	1 000
Pm-147	10 000	100 000
Sm-151	100 000	100 000
Eu-152	10	100
Eu-154	10	100
Eu-155	100	1 000
Gd-153	100	1 000
Tb-160	10	100
Tm-170	10 000	100 000
Tm-171	10 000	1 000 000
Ta-182	10	100
W-181	1 000	10 000
W-185	10 000	10 000 000
Os-185	100	100
Ir-192	100	1 000
Tl-204	10 000	10 000
Pb-210 (+)	10	10
Bi-207	10	100
Po-210	100	1 000
Ra-226 (+)	10	10

Radionuclide	**Clearance level for use (kBq/m²)**	**Clearance level for dismantling (kBq/m²)**
Ra-228 (+)	10	100
Th-228 (+)	1	10
Th-229 (+)	1	10
Th-230	10	10
Th-232	1	10
Pa-231	1	1
U-232	1	10
U-233	10	100
U-234	10	100
U-235 (+)	10	100
U-236	10	100
U-238 (+)	10	100
Np-237 (+)	10	100
Pu-236	10	100
Pu-238	10	10
Pu-239	1	10
Pu-240	1	10
Pu-241	100	1 000
Pu-242	10	10
Pu-244 (+)	10	10
Am-241	10	10
Am-242m	10	10
Am-243 (+)	10	10
Cm-242	10	1 000
Cm-243	10	100
Cm-244	10	100
Cm-245	1	10
Cm-246	10	10
Cm-247 (+)	10	10
Cm-248	1	10
Bk-249	1 000	10 000
Cf-248	10	100
Cf-249	1	10
Cf-250	10	100
Cf-251	1	10
Cf-252	10	100
Cf-254	10	100
Es-254 (+)	10	100

Decay products that have been taken into account when determining values for radionuclides marked with (+) are listed in the table below.

Parent nuclide	**Decay product(s)**
Sr-90	Y-90
Zr-95	Nb-95, Nb-95m
Ru-106	Rh-106
Ag-108m	Ag-108
Ag-110m	Ag-110
Cd-109	Ag-109m
Sn-113	In-113m
Sb-125	Te-125m
Te-127m	Te-127
Cs-137	Ba-137m
Ce-144	Pr-144, Pr-144m
Pb-210	Bi-210
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214
Ra-228	Ac-228
Th-228	Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208, Po-212
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Tl-209, Po-213, Pb-209
U-235	Th-231
U-238	Th-234, Pa-234m, Pa-234
Np-237	Pa-233
Pu-244	U-240, Np-240m, Np-240
Am-242m	Np-238, Am-242
Am-243	Np-239
Cm-247	Pu-243
Es-254	Bk-250

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The Swedish Radiation Safety Authority has a comprehensive responsibility to ensure that society is safe from the effects of radiation. The Authority works to achieve radiation safety in a number of areas: nuclear power, medical care as well as commercial products and services. The Authority also works to achieve protection from natural radiation and to increase the level of radiation safety internationally.

The Swedish Radiation Safety Authority works proactively and preventively to protect people and the environment from the harmful effects of radiation, now and in the future. The Authority issues regulations and supervises compliance, while also supporting research, providing training and information, and issuing advice. Often, activities involving radiation require licences issued by the Authority. The Swedish Radiation Safety Authority maintains emergency preparedness around the clock with the aim of limiting the aftermath of radiation accidents and the unintentional spreading of radioactive substances. The Authority participates in international co-operation in order to promote radiation safety and finances projects aiming to raise the level of radiation safety in certain Eastern European countries.

The Authority reports to the Ministry of the Environment and has around 300 employees with competencies in the fields of engineering, natural and behavioural sciences, law, economics and communications. We have received quality, environmental and working environment certification.

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