Regulatory Viewpoints on Risk Assessment of Spent Nuclear Fuel Final Disposal

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Abstract: In Finland, the final disposal of spent nuclear fuel is scheduled to start in 2025 and the unique nuclear facilities for this purpose are in the commissioning phase. The final disposal solution consists of two nuclear waste facilities: the above-ground spent fuel encapsulation facility and the underground repository (ONKALO).

This paper outlines experiences of the use of Probabilistic Risk Assessment (PRA/PSA) methods for the first-of-a-kind nuclear facility in connection with the application for construction and operating licenses. According to the STUK (Radiation and Nuclear Safety Authority in Finland) regulation, the main risk categories to be considered in the final disposal risk assessment are potential environmental releases and radiological risks to workers from the fuel encapsulation process as well as the long-term safety of the repository. This paper focuses on the risks associated with fuel handling at the encapsulation facility. Examples of methods used to determine the probabilities and values of different consequences and risk criteria for the analyses are presented.

The paper provides insights into how regulatory requirements and risk objectives have been applied to first-of-a-kind nuclear facility. According to the Finnish Regulatory Guides (YVL Guides), the methods used in the risk assessment of a spent nuclear fuel encapsulation facility shall be selected and applied commensurate with the risks associated with the different stages of the encapsulation process. Qualitative methods may be applied in the PRA of the encapsulation facility, supplemented by quantitative analyses as necessary.

Keywords: PRA, Risk Assessment, Spent Nuclear Fuel Final Disposal, Regulatory Requirements

1. INTRODUCTION

According to the Finnish Nuclear Energy Act (990/1987), spent nuclear fuel resulting from the operation of nuclear power plants in Finland, must be stored, processed, and disposed of on Finnish territory. The producer of nuclear waste is responsible for all nuclear waste management activities and their costs. For this purpose, Posiva Oy was established in 1995 in cooperation with the Finnish nuclear power companies Teollisuuden Voima Oyj and Fortum Power and Heat Oy for the mission to manage the disposal of its owner's spent nuclear waste.

Currently, Posiva's spent nuclear fuel encapsulation and final disposal facility is in the process of applying for an operating license in accordance with the procedures and provisions of the Finnish Nuclear Energy Act for nuclear installations. The facility is currently undergoing commissioning tests and is scheduled to start operations in 2025. According to the Finnish Nuclear Energy Degree (161/1988), when applying for a license to operate a nuclear installation, the applicant must submit a Probabilistic Risk Assessment (PRA) to the Radiation and Nuclear Safety Authority. The methods used in the risk analysis of a spent nuclear fuel processing facility shall be selected and applied in accordance with the risks associated with the different stages of the nuclear facility, supplemented by quantitative analyses where necessary. (STUK, 2020)

The objective of this paper is to introduce the risk assessment of the spent nuclear fuel encapsulation process for final disposal in Finland from a regulatory perspective. The long-term safety of geological disposal is assessed and demonstrated with a safety case that is not the focus of this paper. The Posiva solution for final disposal is briefly presented in Chapter 2. Chapter 3 presents the main objectives of risk analyses and Chapter 4 describes some risk assessment methods used. The main consequences and risks are described in Chapter 5 and conclusions in Chapter 6. The paper underlines the Finnish regulatory requirements for risk assessments of final disposal facility.

2. FINAL DISPOSAL SOLUTION IN FINLAND

Posiva's final disposal solution is a complex of two nuclear waste facilities. It consists of an above-ground spent nuclear fuel encapsulation plant and a deep underground repository (ONKALO) at Eurajoki, near the

Olkiluoto nuclear power plant site. The spent fuel is first encapsulated in a copper canister, 4 or 12 fuel assemblies in a one canister, depending on the fuel type. The canisters are then buried deep in the bedrock. Engineered barriers around the canister isolate the fuel from the environment for a period of time long enough to reduce its activity to negligible levels.

The same solution is applied to the three different types of spent fuel, both from pressurised water reactors (PWR) and boiling water reactors (BWR): Loviisa 1&2 (VVER-440), Olkiluoto 1&2 (BWR) and Olkiluoto 3 (EPR). The repositories are designed for 6500 tonnes of uranium, which covers the planned lifetime amount of uranium to be used in the above plants.

2.1 Encapsulation process

The spent nuclear fuel arrives at the reception area (figure 1: (1)) of the encapsulation plant in transport cask from the spent fuel interim storage facilities at the Olkiluoto and Loviisa power plants. Empty deposition canister components are received at the opposite end of the plant (5). The empty canister is lowered from the floor hatch into the transfer corridor and the canister is driven under the fuel handling chamber (2). The empty deposition canister is lifted into the docking station of the handling chamber and the penetration is sealed. The transport cask with the fuel is transported in the same way from the reception area to the transfer corridor, where it is lifted into the fuel handling chamber preparation station. A key principle is to ensure that there are multiple protective structures around the spent fuel and that contamination shall not spread around the premises or into the environment.

If necessary, the fuel assemblies are transferred from the cask to a drying chamber embedded in the floor of the fuel handling chamber. After drying the fuel assemblies are transferred one by one to the deposition canister. Once the canister is full of fuel, it is filled with argon gas and sealed with an inner steel cap. The deposition canister can then be unsealed from the handling chamber’s docking station. Procedures in the handling chamber are performed remotely from a radiation-shielded room, protecting personnel from radiation exposure.

The canister is lowered back into the transfer corridor and moved further under the welding station (3). The canister is lifted to the friction stir welding station, where a copper lid is fitted and welded to the canister. After welding, the canister is lowered back into the transfer corridor and transported to the machining station and on to the weld inspection station (4). After the approved inspections, the canister is transferred to the interim canister storage area. From the canister storage area, the canisters are further transferred to a canister lift (6) for descent to the repository facility.

The plant is designed to minimize the risk of fuel or fuel containers falling during transfers. The encapsulation process can be interrupted in the event of fault and in the event of malfunction, the process remains in a controlled state. The design criterion for the facility is that, even in the event of a fuel handling accident, no harmful quantities of radioactive substances will be released from the facility into the environment. The air circulation of the plant is designed to direct the air towards the handling chamber to prevent the spread of radioactive material. (Posiva, 2021)

Figure 1. Encapsulation plant. Reception of fuel and storage of the transport cask (1), fuel handling chamber (2), copper cover welding station (3), weld inspection station (4), reception and storage of empty deposition canisters (5) and canister lift (6) (Ahlbom, 2021)

2.2 Repository process

The final repository facility, “ONKALO”, is located below the encapsulation plant on one level, approximately 400-450 metres below ground level. The facility consists of underground repository tunnels including deposition holes, central tunnels connecting them, an access tunnel, four vertical shafts (personnel shaft, canister shaft and two ventilation shafts), an underground temporary canister storage area and technical facilities. There is also provision for the disposal of low and intermediate level waste. An illustrative image of ONKALO is presented in figure 2.

The actual disposal is carried out by transferring the canisters one by one to the deposition hole and covering the hole with bentonite buffer blocks. When all the deposition holes in one deposition tunnel have been filled with canisters and bentonite buffers, the tunnel is finally filled with granular bentonite. Once filled, the tunnel is sealed with a stopper to keep the material in the tunnel during plant operation. In addition to the placement of the deposition canisters, the drilling of deposition holes and the extraction works at the central tunnel and other deposition tunnels will continue during operation. (Ahlbom, 2021)

The PRA of the encapsulation and disposal facility is limited to the handling of the fuel from the moment it arrives at the encapsulation plant until it is placed in a bedrock and is no longer handled. Long-term safety is justified by a safety case and it is excluded from the PRA.

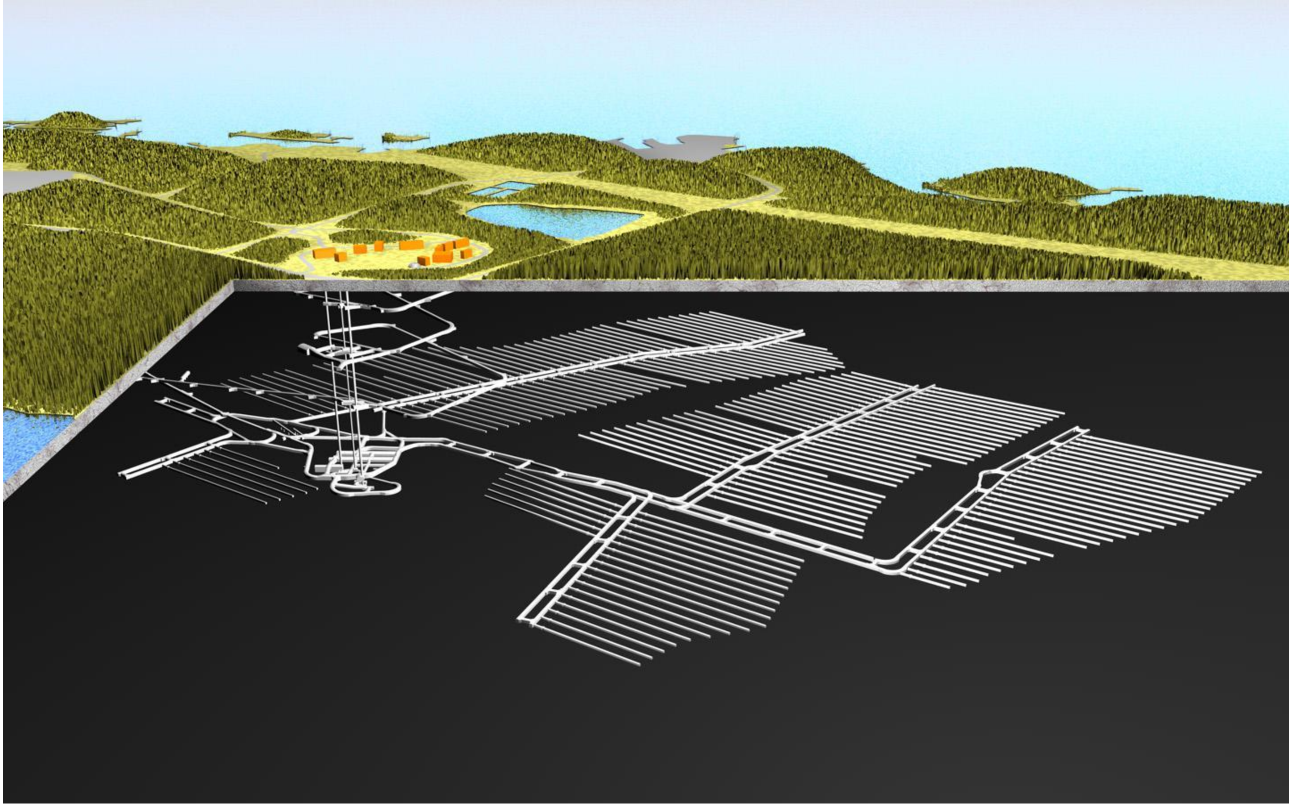

Figure 2. Underground final repository "ONKALO" (Posiva, 2020)

3. THE OBJECTIVES OF THE RISK ANALYSES

In Finland, the purpose of the Radiation and Nuclear Safety Authority's (STUK) operations is to protect people, society, the environment and future generations from the harmful effects of radiation. For this purpose, STUK supervises the safe operation of nuclear installations but also human and personnel exposure to radiation. Various failure and risk analyses are used to ensure the reliability and safe operation of nuclear installation systems. In addition, analyses have been used to identify potential personnel radiation exposures and emission events.

3.1 The objectives of the encapsulation process risk assessment

According to the Finnish Nuclear Energy Act (990/1987), STUK shall specify detailed safety requirements for the implementation of the required safety level in accordance with the Act. These detailed requirements are determined in Finnish Regulatory Guides (YVL Guides) and they are applied to nuclear power plants as well as other nuclear facilities in Finland.

For nuclear power plants, Level 1 PRA focuses on accident sequences leading to reactor core damage and their probabilities. In Finland, the design of a nuclear power plant unit shall be such that the mean value of the frequency of reactor core damage is less than 1E-5 per year. Level 2 PRA for nuclear power plants shall assess the quantity, probability and timing of the release of radioactive substances from the nuclear power plant in the event of severe accidents. The mean value of the frequency of a release of radioactive substances from the plant during an accident with a release of cesium-137 (Cs-137) to the atmosphere exceeding 100 TBq shall be less than 5E-7 per year. According to the Finnish Nuclear Energy Degree (161/1988), the possibility of release in the early stages of an accident requiring measures to protect the population shall be extremely small.

The amount of fuel in the encapsulation plant is significantly less than in a reactor core, approximately 36 fuel assemblies to be handled at one time. (Ahlbom, 2021) An emission of more than 100 TBq is therefore not possible. Thus, instead of a core damage frequency (CDF) or a large release frequency (LRF), the PRA for an encapsulation and disposal facility considers events and event chains related to personnel doses and environmental releases.

As the encapsulation plant mainly consists of fuel hoisting and transfer operations, the risk assessment of the hoisting equipment plays an important role in demonstrating the safety of the plant. The regulatory requirements for the design and assessment of hoisting and transfer equipment in nuclear installations are specified in Regulatory Guide YVL E.11 (STUK, 2019b). The design of lifting functions and hoisting device units shall ensure adequate radiation protection and that the probability of nuclear fuel damage is minimised. It shall be based on both deterministic and probabilistic methods.

3.2 Long-term safety assessment objectives

Regulatory Guide YVL D.5 (STUK, 2018) defines safety limitations for radiation doses and radioactive releases for long-term safety of final disposal. The dose criterion is applied for the period of first millennia post-closure (dose assessment period) and the release constraints limit criterion for the period after first millennia (release constraint limits period). The disposal of nuclear waste shall be so designed that the radiation impacts arising as a consequence of expected evolution remain insignificantly low or below value 0.1 mSv per annum during dose assessment period. The disposal of nuclear waste shall be also designed so that, as a consequence of expected evolution, the average long-term quantities of radioactive materials released into the living environment from disposed nuclear waste remain below the constraints specified separately for each nuclide by STUK during release constraints limit period.

For this purpose, the probabilities of rare events impairing long-term safety and their impacts on the disposal system and the long-term safety of disposal shall be assessed. The plausibility, probability and potential consequences of each event shall be assessed as far as possible. The probability of events leading to significant radiation exposure shall be very low, and the widespread effects of the release of radioactive substances caused by them shall be limited. Rare events consist of natural events, like earthquakes, as well as future human actions considering the next 10 000 years. (STUK, 2015) The assessment of long-term safety is not part of PRA and not in the scope of this paper any further.

4. RISK ASSESSMENT METHODS

To achieve the objectives presented in Chapter 3, the collection of different deterministic and probabilistic analysis types is presented briefly in Table 1.

Table 1. Collection of used analyses and their targets

Analysis	Target
Failure mode and effect analyses (FMEA)	Safety of equipment and systems
Failure tolerance analyses	Safety of hoisting equipment and operations
Common cause failure analysis	Reliability of I&C and ventilation functions
Hazardous scenario analysis (HAZSCAN)	Identification of process activities
Initiating event analyses	Identification of internal and external events
Quantitative calculations	Amount and probability of radiological release and human dose

The applied risk analysis methodology is based on the IAEA procedures for conducting probabilistic safety assessment for non-reactor nuclear facilities. (IAEA, 2002) According to the logic presented in Figure 3, the analysis examines the initiating events from perspective of both causes and consequences. Consequences are described more in Chapter 5, consequence class C1 is not quantified.

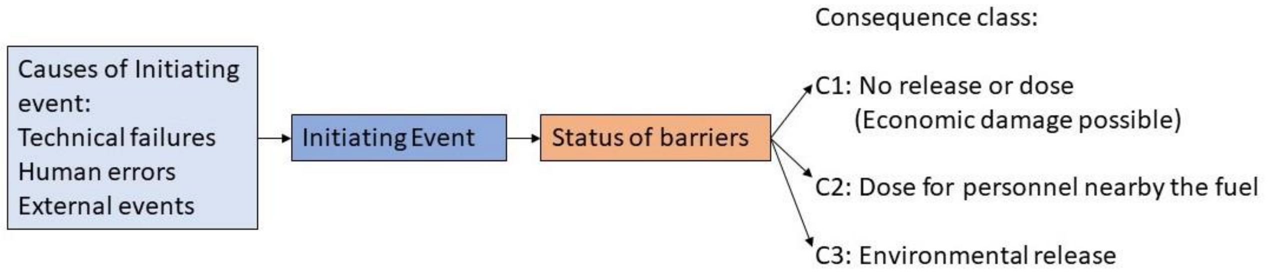

Figure 3 Risk model for Encapsulation plant (Posiva 2021, modified)

Risk analysis was performed in two phases. First, extensive qualitative analyses were done, then quantitative analyses were completed for the most significant event sequences according to radiation safety. Hazardous scenario analysis (HAZSCAN) was performed as a qualification of processes and activities at the encapsulation facility as well as transportation of the canister to the final disposal facility. Initiating events were identified taking into account technical failures, human errors and external events. Failure mode and effect analysis (FMEA) was performed especially for the power supply and ventilation systems. Note that loss of residual heat removal and criticality accidents are not relevant for the risk analysis due to the applied design basis. Mechanical damages of the spent fuel elements during handling (especially load drop) and personnel accidental exposure to radiation from uncovered fuel were found out to be the most important sequences.

Assessment of external hazards contained also possible combinations of events. (Knochenhauer, Louko, 2003) Most important combinations of events are related to extreme weather conditions. Consequences of external hazards are typically affecting operability of the fuel handling devices and ventilation systems. However, the fuel handling process can be stopped in any time and many equipment are locked in a safe state for example in case of loss of power. Considering internal hazards, only some specific fire events inside the facility were needed to assess, for example fire inside the fuel handling chamber and fire within the fuel canister transportation vehicle.

Event trees were developed for initiating events leading to external releases or personnel exposure to radiation. FinPSA code was used to modelling and quantification of the sequences. External releases were quantified only for scenarios where a fuel rod or a fuel element was damaged (damage = loss of tightness). The releases were calculated separately for the three different fuel types to be handled at the facility and taking into account the number of damaged fuel elements in each scenario. The scenarios also cover conditions on the environment of the fuel (either dry or under water), canister tightness (small leakage or total loss of tightness) and filtering system availability.

Analysis of human errors was found preliminary. Probabilities of the most significant human actions will be checked and possibly re-quantified after finalization of the operating procedures of the facility.

5. MAIN RISKS OF THE ENCAPSULATION PROCESS

The reference for emission and dose criteria for operational incidents and anticipated accidents is the Finnish Nuclear Energy Degree (161/1988).

5.1 The risk of radiological release

As an indicator for environmental release (consequence class C3), cesium (Cs-137) is used for particulate nuclides and krypton (Kr-85) for gaseous nuclides. The regulatory dose criteria (mSv/event) are converted to the corresponding Cs-137 emission with conservative assumptions (1E-12 Sv/Bq). The calculation of nuclide emissions is based on fuel type-specific estimated quantities of nuclides, from which the emission per initial event is estimated by considering the number of broken fuel assemblies, nuclide inventory per fuel assembly, release fraction from the rod, retention in water, effect of removal to surfaces i.e. deposition and effect of controlled area filtration.

The most critical issues in terms of emission risk are the frequency of initial events and the reliability of the air conditioning filtration. Initial events are defined as events that lead to damage or malfunction of the plant, which could result in damage to the fuel or deposition canister and compromise the radiation safety of the plant. Significant emissions in excess of 1E9 Bq Cs-137 may only occur in the event of fuel assembly damage combined with a failure of the air conditioning filtration system.

The most common, even annual, fuel damage may occur if the fuel assembly collides with another fuel assembly during the transfer between the transfer cask, the drying chamber and the deposition canister. Another significant event may occur if the lid of the transfer cask falls on the fuel. However, Cs-137 emissions are negligibly low if the air conditioning filter is working as designed.

A significant proportion of the frequency of unfiltered releases from fuel assembly damage occurs when a deposition canister falls into a deposition hole and becomes partially or completely unsealed. The magnitude of the release, approximately 1E8 Bq Cs-137, is comparable to the anticipated operational occurrence in a nuclear power plant. Estimated frequency of the event is 7E-6 per year. (Posiva, 2021)

5.2 The personnel dose risk

The individual personnel dose amount (consequence class C2) is calculated for each initiating event, considering the estimated dose rate. Exposure time is assumed to be conservatively one hour. Personnel dose amounts are not calculated with risk models, but the dose amounts for each initiating event are based on a separate statement considering fuel type and release barriers at the time of event.

Personnel may receive a dose from personal exposure to radiation from a bare fuel assembly or canister, or if a person is exposed to a small amount of radioactive material that has been misplaced. Doses are divided into three categories: low dose (less than 20 mSv per year), dose exceeding the annual dose limit (more than 20 mSv per year) and lethal dose (more than 6 Sv).

The lethal radiation dose is only possible if a person enters the handling chamber or transfer corridor at the wrong time, when the transfer cask or deposition canister is not sealed, and the fuel is exposed. Due to restricted access, the frequency of this event is estimated to be very low, approximately 3E-8 per year. Doses above the annual limit could also result from an hour spent near a deposition canister, or from being in the transfer corridor at the wrong time or from incompletely cleaned fuel pieces in the handling chamber. However, these doses are not expected to cause any radiological disease. (Posiva, 2021)

5.3 Benefits of the risk analyses

Risk analyses are part of the process of demonstrating that the nuclear installation meets the safety requirements established as the design basis. The PRA has been used to assess whether the design of the encapsulation process is balanced from a risk perspective. The licensee shall also use the PRA to improve the safety of nuclear facility and to identify and prioritise plant modifications. (STUK, 2019a)

The regulatory body requires the risk analyses to be used in the planning of personnel training. The PRA shall also be used to support the development of abnormal and emergency operating procedures and Operational Limits and Conditions. (STUK, 2019a) During the commissioning phase of the encapsulation plant, the main risk-related operational activities were considered as verification and validation scenarios for the operating procedures.

6. CONCLUSION

As the encapsulation and disposal facility is the first of its kind, the suitability of the risk analysis requirements for nuclear power plants had to be reviewed. Due to the different plant design, the overall risks are significantly lower than for nuclear power plants and numerical targets for core damage frequency or large release frequency are not meaningful as the CDF is not applicable to encapsulation plant and the limit for LRF cannot be exceeded. However, the same regulatory requirements apply for demonstrating that the required level of safety has been achieved.

From the regulatory perspective, the key to demonstrating the required level of safety is a defined set of analyses, including both deterministic and probabilistic methods. Although the numerical targets are different from those for nuclear power plants, similar analysis methods may be appropriate. Qualitative analysis methods play an important role in identifying new type of risks.

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