BGE TECHNOLOGY GmbH

# Waste Package for Disposal of High-Level Waste (HLW) in Rock Salt

**BGE TEC 2023-02**

**Technical Report**

# Waste Package for Disposal of High-Level Waste (HLW) in Rock Salt

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| Date | 13.04.2023 |
| Client | COVRA N.V |

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## Abbreviations

| | |
|---|---|
| ADR | Accord européen relatif au transport international des marchandises dangereuses par route (French), in English "European Agreement concerning the International Carriage of Dangerous Goods by Road" |
| ANVS | Authority for Nuclear Safety and Radiation Protection |
| BE | Brennelement (German), in English "Spent Fuel Element" |
| BSK | Brennstabkokille (German), in English "Spent Fuel Canister" |
| CSD-C | Conteneur Standard de Déchets – Compactés (French), in English „Standard Waste Packages - Compacted" |
| CSD-V | Conteneur Standard de Déchets – Vitrifiés (French), in English "Standard Waste Packages - Vitrified" |
| COGEMA | Compagnie Générale des Matières Nucléaires (French), in English "General Nuclear Material Company" |
| COVRA | Centrale Organisatie Voor Radioactief Afval (Dutch), in English "Central Organisation for Radioactive Waste" |
| EBS | Engineered Barrier System |
| ECN | Energieonderzoek Centrum Nederland (Dutch), in English "Energy Research Centre of the Netherlands" |
| EDZ | Excavation Damaged Zone |
| ENCON | Endlagercontainer (German), in English "Waste Container" |
| ENTRIA | Entsorgungsoptionen für radioaktive Reststoffe: Interdisziplinäre Analysen und Entwicklung von Bewertungsgrundlagen (German), Acronym of an R&D project, in English "Disposal options for radioactive residues: Interdisciplinary analyses and development of assessment bases" |
| FEA | Finite Element Analysis |
| FKM | Forschungskuratorium Maschinenbau (German), in English „Mechanical Engineering Research Board" |
| DWP | Disposal Waste Package |
| DWR | Druckwasserreadktor (German), in English "Pressurized Water Reactor" |
| HAA | Hochaktiver Abfall (German), in English "High Level Waste" |
| HEU | Highly Enriched Uranium |
| HLW | High Level Waste |
| IAEA | International Atomic Energy Agency |
| ISO | International Organization for Standardization |
| KBS-3 | kärnbränslesäkerhet-3 (Swedish), in English "nuclear fuel safety-3", name of a technology for disposal of high-level radioactive waste |
| KoBrA | Anforderungen und Konzepte für Behälter zur Endlagerung von Wärme entwickelnden radioaktiven Abfällen und ausgedienten Brennelementen in Steinsalz, Tonstein und Kristallingestein (German), Acronym of an R&D project, in English "Requirements and Concepts for Disposal Containers for Heat-Generating Radioactive Waste and Spent Fuel in Salt, Clay and Crystalline Rock |
| LEU | Low Enriched Uranium |
| LILW | Low and Intermediate Level Waste |
| Nagra | Nationale Genossenschaft für die Lagerung radioaktiver Abfälle (German), in English "National Cooperative for the Disposal of Radioactive Waste" |

| | |
|---|---|
| RESUS | Grundlagenentwicklung für repräsentative vorläufige Sicherheitsuntersuchungen und sicherheitsgerichtete Abwägung von Teilgebieten mit besonders günstigen geologischen Voraussetzungen für die sichere Endlagerung wärmeentwickelnder radioaktiver Abfälle (German), Acronym of an R&D project, in English "Development of the basis for representative preliminary safety studies and safety-oriented consideration of subareas with particularly favourable geological conditions for the safe disposal of heat-generating radioactive waste“ |
| R&D | Research and Development |
| RWM | Radioactive Waste Management |
| SC | Supercontainer |
| SSR | Specific Safety Requirements |
| SSG | Specific Safety Guides |
| SKB | Svensk Kärnbränslehantering (Swedish), in English „Swedish Nuclear Fuel and Waste Management Organisation” |
| SWR | Siedewasserreaktor (German), in English “Boiling water reactor“ |
| (TE)NORM | (Technologically Enhanced) Naturally Occurring Radioactive Material |
| UCW | Uranium collection filter waste |
| UNS | Unified Numbering System for Metals and Alloys |
| UTS | Ultimate Tensile Strength |
| WWER | Water-Water Energetic Reactor |

**Formula symbols**

| | | |
|---|---|---|
| A | Elongation at break | [MPa] |
| a | Degree of utalisation | [-] |
| $j_{UTS}$ | Ultimate tensile strength safety factor | [-] |
| $j_E$ | Corrosion safety factor | [-] |
| $j_{YS}$ | Yield strength safety factor | [-] |
| $j_{total}$ | Total safety facor | [-] |
| $K_p$ | Plastification factor | [-] |
| $K_{T,UTS}$ | Temperature factor tensile strength | [-] |
| $K_{T, Rp}$ | Temperature factor yield strength | [-] |
| $n_{pl}$ | Section factor of plastification | [-] |
| $p_a$ | Pressure, outside | [MPa] |
| $p_i$ | Pressure, inside | [MPa] |
| $r_a$ | Radius, outside | [m] |
| $r_i$ | Radius, inside | [m] |
| $r_x$ | Radius for calculation | [m] |
| $R_p$ | Yield strength | [MPa] |
| T | Temperature | [°C] |
| | | |
| $\sigma_a$ | Axial stress | [MPa] |
| $\sigma_{CS}$ | Compressive strength | [MPa] |
| $\sigma_r$ | Radial stress | [MPa] |
| $\sigma_t$ | Tangential stress | [MPa] |
| $\sigma_v$ | Vertical stress | [MPa] |

# 1 Introduction

Today, the Netherlands follow the strategy of long-term interim storage of radioactive waste and an eventual disposal in a repository. A multinational repository and two national repository concepts are considered by COVRA N.V. (hereafter COVRA). The two national repository concepts include the disposal of the radioactive waste in poorly indurated clay and disposal in rock salt, either in embedded salt or in a salt dome. The present inventory includes non-heat-generating and heat-generating waste. The waste covers HLW as well as LILW. Today's preference is to dispose of all categories of waste in one facility. To limit the chemical interactions between the different waste categories, the waste is to be sorted by category as defined by COVRA. Currently, the waste is stored in a long-term aboveground storage facility. It is expected that a disposal facility will go into operation around 2130 as stated by ANVS (2016).

For disposal in clay, a waste container for heat-generating waste has already been developed – a so-called Supercontainer. The Supercontainer concept is characterised by a combination of primary waste package, an overpack, a concrete shell as geochemical buffer, and a steel envelope. In addition to the shielding function, the concrete shell is to provide suitable geochemical conditions for the inner steel-based components.

For disposal in rock salt, no waste container specific to the current disposal concept has been developed yet. COVRA just specifies the use of a self-shielding waste canister, which means that the canister design allows the adherence to a predefined radiation level at the surface of the canister. For the Dutch case, a dose rate limit of maximal 10 mSv/h is defined. A direct transfer of the Supercontainer concept from clay to salt seems not suitable because of the different geochemical conditions between the poorly indurated clay and the rock salt and will be discussed in detail in this report.

In order to further develop the conceptual design of the waste containers, COVRA commissioned BGE TECHONOLOGY GmbH with defining the requirements for a HLW container for heat-generating and non-heat-generating waste and with developing a conceptual design of a container adapted to the Dutch conditions. This is the framework of the study in hand. The overall objectives of the project are to:

- Perform a review of existing international concepts for self-shielding HLW waste packages
- Identify advantages and disadvantages of a standardised self-shielding HLW waste package
- Compare the self-shielding waste package with the other proposed waste package options in the Netherlands
- Produce a conceptual design for a self-shielding standardised HLW package, either using an existing concept or using a new approach
- Quantify the geometric and mechanical key properties of the conceptual design
- Give a first order estimate of key processes that could lead to containment failure
- Give an estimate of the expected containment period for the self-shielding waste package and the resulting amount of gas production due to degradation
- Estimate the shielding effect

## 1.1 Repository Concept

In the repository concept for rock salt, the waste will either be emplaced in a salt dome or in bedded salt. According to the current reference concept (Bartol et al., in progress), emplacement in a Zechstein salt dome is considered for this study. The underground facility consists of two separate disposal levels. The lower disposal level, at a depth of approximately 850 m, is for HLW. The upper disposal level, at a depth of approximately 750 m, is for LIL – (TE)NORM waste. The two levels are connected by shafts and an internal ramp. A central main area at the upper level can be used as infrastructure area. The use of two levels instead of two lateral extended emplacement areas is envisaged to make an optimal use of the vertical extent of a salt dome and to dispose the different waste types in separate sections of the underground facility.

The layout of the HLW disposal area is shown in Figure 1-1. From a central transport gallery, the emplacement galleries will branch off to both sides. All emplacement galleries are connected to a ventilation gallery opposite to the transport gallery. The ventilation galleries are connected to two (exhaust) shafts. The transfer from the surface to underground is performed through the waste shaft, next to the central area. The waste packages will be transported through the transport gallery to the emplacement galleries and will be emplaced there by a transport and emplacement vehicle.

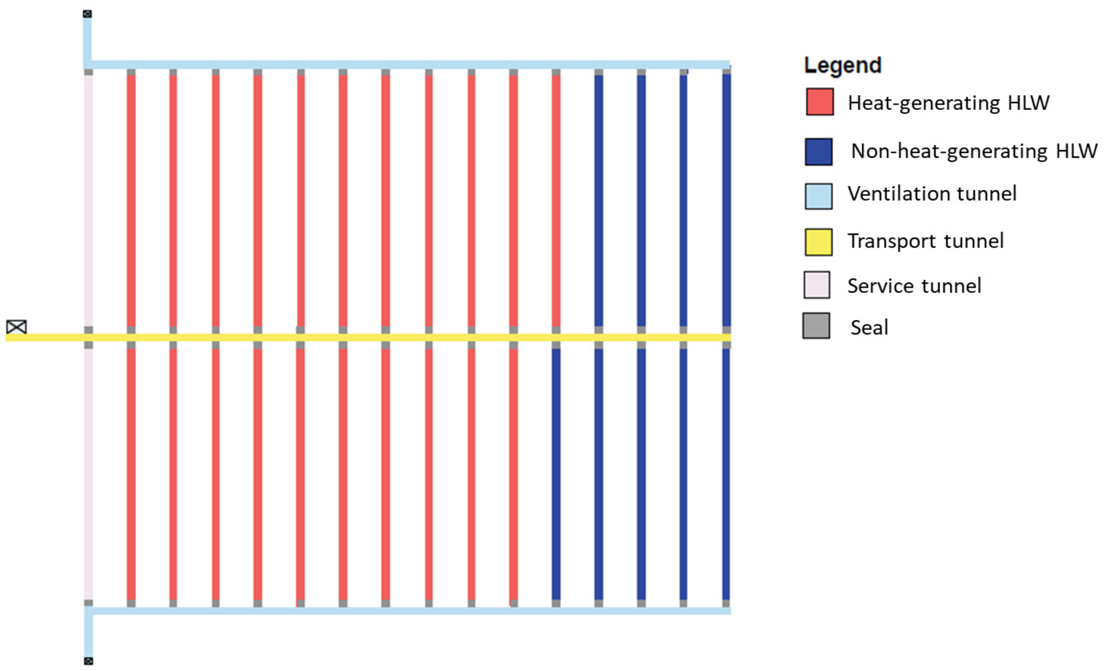



*Figure 1-1: Layout of the HLW disposal level in the salt dome (Bartol et al., in progress)*

The emplacement galleries for the HLW are expected to be 100 m long, have a height of 4 m, and a width of 5 m. The cross section is characterised by slightly rounded edges. The ventilation galleries have the same dimensions but differ in length. The transport gallery has an increased width of 10 m and the same height (4 m).

The waste packages will be disposed inside the emplacement galleries standing in shallow boreholes. The boreholes are to prevent the waste packages from falling or rolling. The galleries will be backfilled with dry crushed rock salt. Figure 1-2 illustrates the disposal concept.

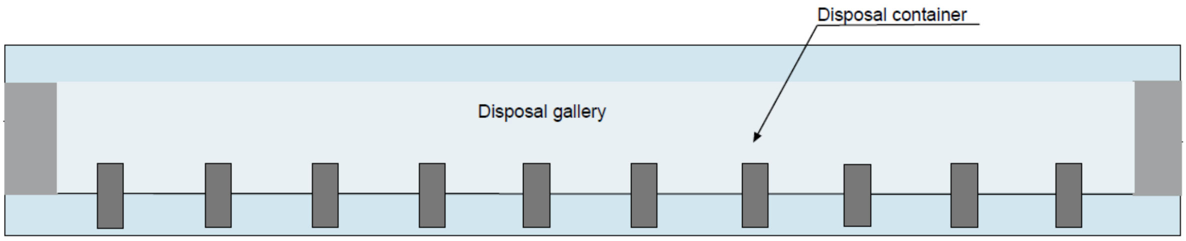



*Figure 1-2: Illustration of the disposal in shallow boreholes (Bartol et al., in progress)*

The Tweede Kamer (1992) has specified a reversibility in principle of the waste disposal process in the deep underground. As a result, a retrieval of the waste has to be possible at least during the operational phase of the disposal facility. Based on NEA (2011), it is assumed that *"Retrievability is the ability in principle to recover waste or entire waste packages once they have been emplaced in a repository. Retrievability implies making provisions in order to allow retrieval should it be required."* The actual process of removing the waste packages is called retrieval. However, it has not yet been decided how long retrievability has to be possible after the end of disposal operation or which technical requirements have to be fulfilled in regard to retrievability/retrieval (Ministry of Infrastructure and the Environment, 2016). Thus, for the study in hand, a detailed technical concept how to retrieve or how to ease retrievability is not considered. It is assumed that the current disposal concept does not exclude retrievability/retrieval.

The Dutch repository concept is designed as multi barrier system consisting of the geological barrier and an engineered barrier system (EBS). The main barrier for the long-term safety is the geological barrier supported by geotechnical barriers (shaft or gallery seals) and the backfill (compacted crushed salt). The waste container is expected to be the main barrier for the operational phase and a not yet defined timeframe after closure. This yet undefined time covers the potential period of retrieval, which is not yet determined, and the compaction period of the backfill. Once the backfill is compacted, it can provide its long-term sealing function in combination with the geotechnical barriers and the host rock as geological barrier. See Figure 1-3 for an illustration of the different sealing effects of important barriers over time.

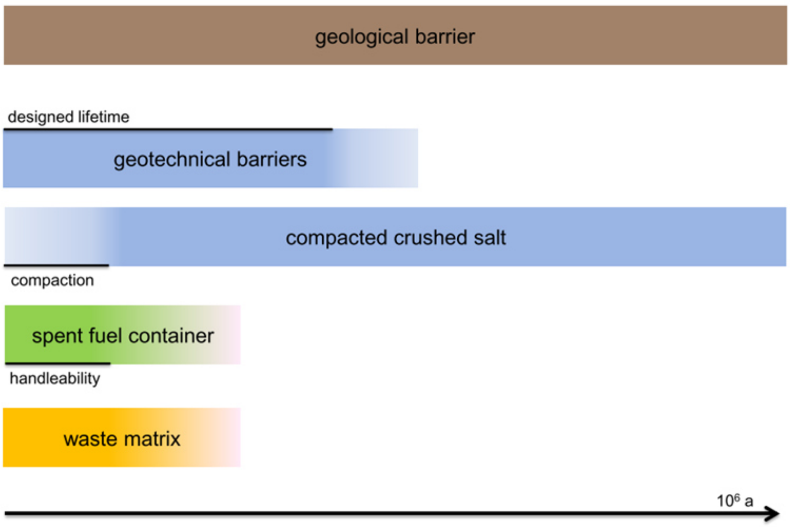



*Figure 1-3: Evolution over time of the sealing effects of important barriers in the post closure phase of a repository system. Note that the colour intensity representing the degree of the respective sealing effect is not to scale. (Bollingerfehr et al., 2013)*

## 1.2 Geological Model

The salt deposits of the Netherlands sedimented approximately 250 Ma ago and can be assigned to the Zechstein group (late Permian). They were formed in the Zechstein Basin, which stretches from the east coast of England through Germany to northern Poland. Lithologically, the Zechstein succession contains clastic sediments, carbonate and anhydrite rocks, rock salt, and potash seams, representing evaporitic cycles. In total, the group can be divided into four, locally five, evaporitic cycles (e.g. Van Adrichem Boogaert & Burgers 1982). In the further course of geological history, the rock salt subsided to greater depths and was overlaid by other rock types.

The mineralogical-geochemical conditions of the Zechstein salts in the Netherlands and in northern Germany are comparable. In both countries, the salts often occur in the form of salt domes or salt diapirs. They are a result of halokinesis, a process that covers the movement of salt under the influence of gravity. Halokinesis causes a folding of the rock strata in a salt dome. Layers of carbonate and anhydrite rocks were fractured during folding of the salt rock and now exist as discrete blocks within the salt dome. Due to the comparable framework conditions, this project used a generic model of a salt dome from the German research project RESUS (Bertrams et al., 2020). Figure 1-4 shows the generic model. The colours are to clarify the differences between the rocks and their positions. An overview of cross-sections through salt diapirs can be found, for example, in Kockel (2000). The second evaporite cycle, also known as the Staßfurt cycle (Z2), often forms the core of the salt domes and is therefore of particular importance for the disposal of radioactive waste. Geluk (1995) describes the structure of the Staßfurt cycle in the Netherlands.

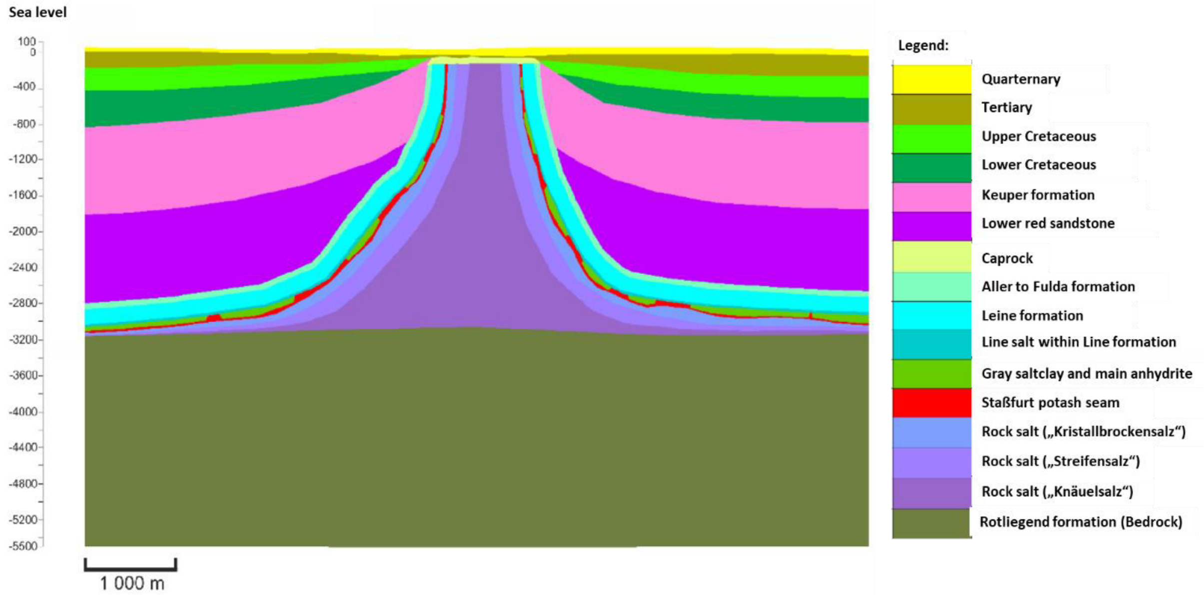



*Figure 1-4: Simple generic model of a salt dome taken from the German research project RESUS (Bertrams et al., 2020)*

In the model, the ground level is 50 m above sea level. The salt dome begins at 120 m below sea level. The base of the salt dome is at 3,150 m below sea level. In the model, the salt dome is approximately 10 km long, normal to the cutting plane. Overall, the salt dome provides enough space to host the repository. The inner structure of the salt dome is expected to be of rock salt from the Staßfurt formation. Other evaporitic minerals such as potash or anhydrite are located at the flanks of the domal formation. The top of the dome is covered by cap rock. The disposal facility will be located around 850 m below sea level. Based on Bonté et al. (2012), an initial rock temperature of 35 °C is expected for the disposal level.

The disposal galleries will be placed in central rock salt layers. In addition to the main component halite ($NaCl$), the sulphates anhydrite ($CaSO_4$) and polyhalite ($K_2Ca_2Mg[SO_4]_4 \cdot 2H_2O$) can occur as secondary components (Biehl et al., 2014; Engelhardt et al., 2000).

### 1.3 Waste Inventory

Figure 1-5 shows the waste families that have to be disposed of. The given numbers of containers and canisters base on (Bartol et al., in progress). For the design of the waste package in this report, the heat-generating HLW is relevant.

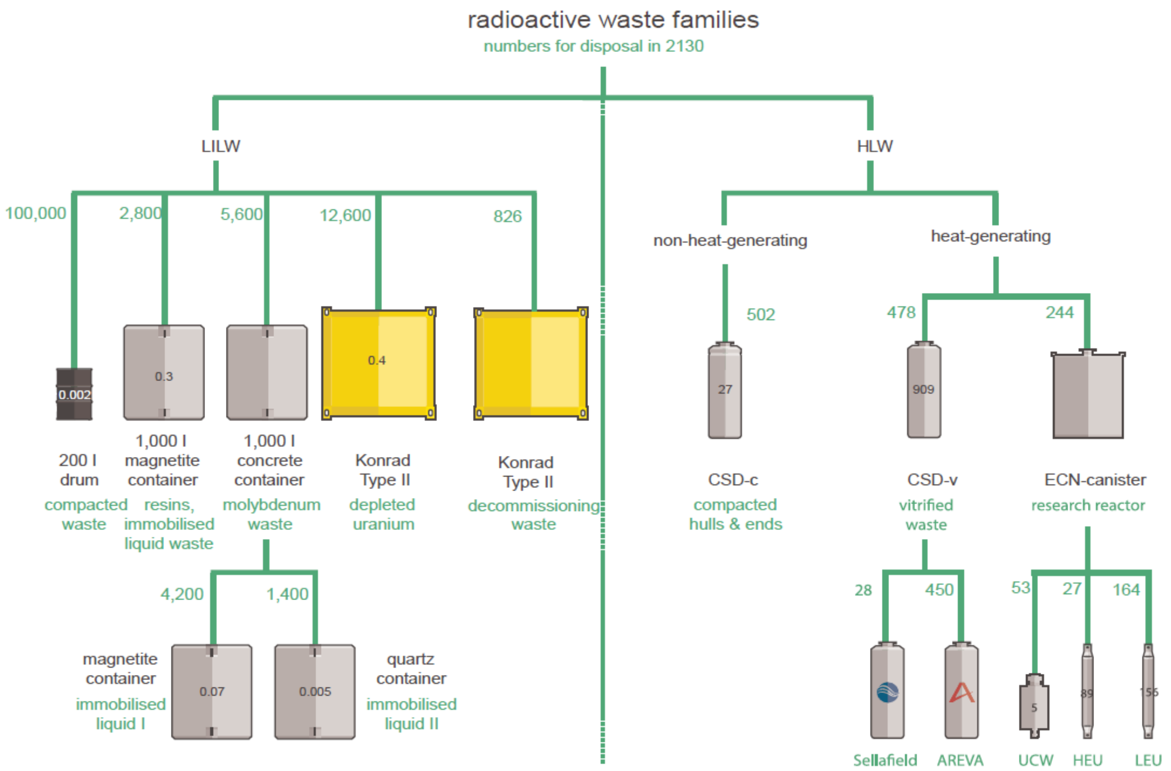



*Figure 1-5: Waste families in the Dutch inventory and their relevant containers and canisters in which the waste is stored and/or conditioned. Expected numbers of each container type are indicated in green. (Bartol et al., in progress)*

The Dutch heat-generating HLW consists of two waste types: CSD-V (Colis Standard de Déchets-vitrifié) canisters filled with vitrified waste and ECN canisters filled with spent fuel from research reactors.

The spent fuel from commercial nuclear power plants is reprocessed in France and in the United Kingdom. 478 CSD-V canisters with heat-generating vitrified waste (CSD-V) have to be disposed. Figure 1-6 shows the standard 170 litre internal volume container for vitrified HLW produced by COGEMA, France.

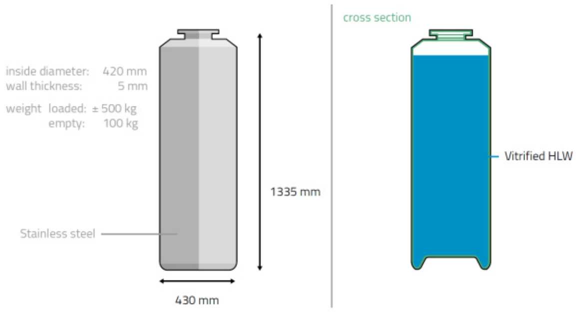



*Figure 1-6: CSD-V canister filled with vitrified HLW from reprocessing of spent-fuel (Verhoef et al., 2017)*

The second canister type that has to be considered are ECN canisters. These are filled with either conditioned spent fuel from research reactors or spent uranium targets from molybdenum production. 244 ECN canisters have to be considered, 27 of which contain Highly Enriched Uranium (HEU), 164 contain Low Enriched Uranium (LEU), and 53 contain filter residuals (UCW). Figure 1-7 shows the ECN canister with its dimenions.

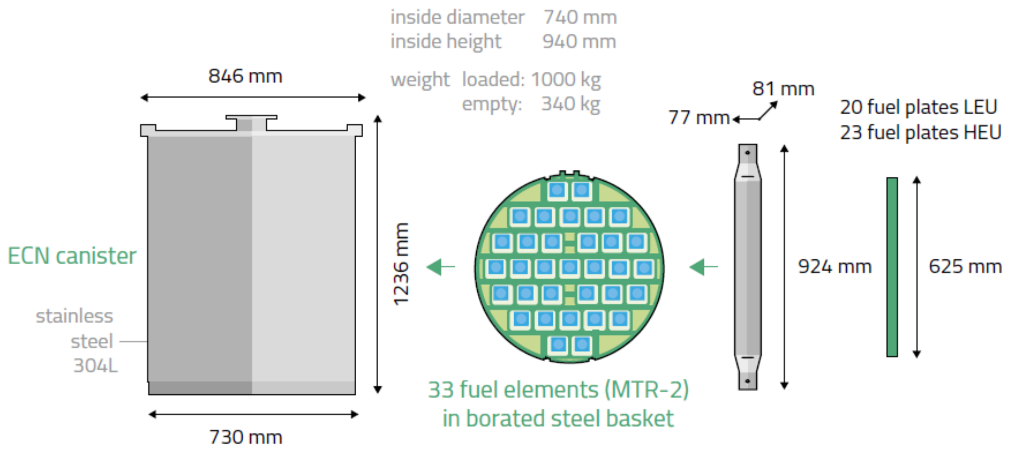



*Figure 1-7: ECN canister filled with spent fuel from research reactors or uranium targets from molybdenum production (Verhoef et al., 2017)*

The Dutch non-heat-generating HLW is packed in CSD-C canisters, which contain the compacted hulls and ends of the fuel rods. A total amount of 502 CSD-C is considered.

## 2 Waste Package Requirements and Safety Functions

According to the IAEA Safety Standard No. SSR 5 *"a disposal facility is designed to contain the radionuclides associated with the radioactive waste and to isolate them from the accessible biosphere"*. Furthermore, the disposal facility is *"also designed to retard the dispersion of radionuclides in the geosphere and biosphere and to provide isolation of the waste from aggressive phenomena that could degrade the integrity of the facility. The various elements of the disposal system, including physical components and control procedures, contribute to performing safety functions in different ways over different timescales."*

It is international consensus that an important design principle to provide containment, isolation, and retardation is the use of a multi-barrier system, consisting of the host rock together with the engineered barrier system. The waste package itself is part of the engineered barrier system. The IAEA safety standard No.SSR-5 further states:

*"The physical elements and their safety functions can be complementary and can work in combination. The performance of a disposal system is thus dependent on different physical elements and on other elements that perform safety functions, which act over different time periods. For example, the roles of the waste package and the host geological formation for a geological disposal facility may vary in different time periods."*

In the German research project "KoBrA" (Bollingerfehr et al., 2020), a method for determining the safety requirements and functions of disposal waste packages (DWP) was developed:

1. Step: Consideration of all regulatory requirements – Result: phase-depending requirements and functions
2. Step: Generic description of repository concept and verification concept for respective host rock – Result: more concrete requirements and functions
3. Step: Site-specific requirements / quantified requirements

The method was developed using a top-down approach. The first two steps of the method were used in generic German R&D studies and will be applied in this study as well.

In the first step, the relevant national and international laws and regulations serve as a basis. From these, generic, i.e. abstract or general, canister requirements are developed for the respective phases of use of the waste packages in the repository, and are derived from the regulatory requirements. As a result, use- or phase-dependent generic waste package requirements can be formulated. The waste package functions necessary to fulfil the phase-dependent requirements can be derived thereof. These first step of the method has already been applied in generic German R&D studies.

In the second step of the top-down approach, generic repository systems and associated safety and demonstration concepts were included in the considerations. The focus of the project was on the German HLW/SNF repository and included the three host rocks salt, claystone, and crystalline formations. In this way, both the waste package functions for fulfilling the requirements and the impacts on the respective host rock could be specified more precisely.

These results were intended to be used as basis for an eventual development of specific canister concepts for individual host rocks.

Finally, in the third and last step of the top-down approach, the repository site determined in the site selection procedure would be considered with its specific properties and characteristics. Based on (i) the specific safety and demonstration concept to be developed for this site, (ii) the quantified site-related impacts, and (iii) the necessary canister functions to fulfil the quantified site-related requirements, the specific canister design for the repository site could be developed. This last step (3.step) is not relevant for the current planning stage of the Dutch repository concept, because a site selection has not happened yet.

Following the top-down approach, six major waste package safety functions were identified in KoBrA (Bollingerfehr et al., 2020):

1. Containment of the radioactive waste, e.g. as defined in IAEA SSR-5 (IAEA, 2011a)
2. Shielding of the radiation, e.g. as defined in IAEA SSR-5 (IAEA, 2011a)
3. Absence of criticality, e.g. as defined in IAEA SSG-14 (IAEA, 2011b)
4. Limiting temperature of the radioactive waste, e.g. as described in IAEA SSG-14 (IAEA, 2011b)
5. Limiting corrosion and gas production, e.g. as described in IAEA SSR-5 (IAEA, 2011a)
6. Operability, e.g. as described in IAEA SSR-5 (IAEA, 2011a)

Furthermore, COVRA required that the waste package behaviour and its degradation over time be predictable. In addition, it was required that the waste package can be produced with proven manufacturing methods and from well-known materials. Thus, for this project, a seventh safety function “manufacturability and predictability” has been introduced.

The seven safety functions have to be fulfilled by the waste package during its required lifetime. The general canister lifetime for the waste package in the Dutch case can be divided into three phases as shown in Table 2-1.

*Table 2-1: Phases of a repository and waste package lifetime*

| **Utilisation Phase** | **Start** | **End** |
|---|---|---|
| Operational and emplacement phase | Supply of the waste package for emplacement | Completion of emplacement and backfilling of the waste package |
| Retrievability phase | Completion of the disposal operation | Not defined |
| Post-closure phase | After closure of the repository (assumption) | End of the reference period of the safety case |

The waste package lifetime considered in this report starts with the encapsulation of the waste at the beginning of the operational phase. The operational period runs until the emplacement of the waste and the backfilling of the disposal galleries have been completed. In the Dutch disposal concept, retrievability of the waste packages has to be considered. So far, the Dutch government has not decided how long the waste packages should be retrievable (Ministry of

Infrastructure and the Environment, 2016). Therefore, the timeframe relevant for the containment function of the waste package has not been adapted to a pre-defined retrievability phase, but is assumed as maximum lifetime, see section 2.1. The later post-closure phase runs until the end of the reference period of the safety case. COVRA currently assumes a reference period of one million years (Bartol et al., in progress).

In the following, each waste package safety function is described and, if possible, also quantified. In addition, it is stated during which phase of the repository lifetime fulfilment of the safety function is necessary.

## 2.1 Containment

The containment function of the waste package must be ensured to prevent the release of radionuclides into the biosphere to the extent that the radiological limits specified in national laws are met. For this purpose, the waste package must be sufficiently sealed and withstand the expected mechanical impacts. The waste package is not considered as long-term barrier for the full reference period. The long-term barrier is the host rock together with the backfill and the geotechnical barriers. Therefore, the containment function of the waste package is only necessary during the phases, in which the repository itself is not yet sealed, and as long as the geotechnical barriers do not yet provide their full sealing capacity. During these phases, the waste package has to prevent the release of radionuclides to protect workers and the biosphere. This includes the period of pre-closure with the actual operational phase, an observation period, a not yet defined period of retrievability, and a part of the post closure phase.

Based on (Verhoef et al., 2017), the disposal of all types of waste will be done in a period of approximately 30 years. An observation and closure phase follows directly and extends over a period of 20 years. A period during which retrievability has to be guaranteed has not yet been defined by Dutch law. During the early post-closure phase, especially the time needed for compaction of backfill defines the required containment period of the DWP. Calculations carried out within the scope of the preliminary safety assessment for the Gorleben site (Müller-Hoeppe et al. 2012) showed that the compaction of crushed rock salt will be finished within several hundreds of years. After 1,000 years, the compaction will be nearly completely finished and the backfill can provide its long-term sealing function. This includes a permeability of the backfill close to the permeability of the undisturbed rock salt (Czaikowski et al., 2020). For the design process of this study, a minimum containment function of at least 1,000 years has been defined for the DWP.

To guarantee containment during this period, the waste package has to withstand all impacts and loads originating from emplacement or retrieval processes or from the repository itself. Resistance to mechanical loads (static or dynamic) is strongly linked to operability. Providing containment is also strongly linked to the corrosion resistance of the waste package. Usually, the waste package also has to provide containment during incidents that may occur in the course of emplacement or retrieval operations. As respective loads rarely have a significant impact on the mechanical requirements, such scenarios have not been considered in this basic study.

## 2.2 Shielding

To protect workers and the environment from ionising radiation, the waste package must have sufficient shielding capabilities. Essential parameters here are the choice of material and the wall thickness of the canister. Performance targets, such as dose rate during operation, will be defined by national law or license conditions. Performance targets during the post-closure phase depend on the design of the other barriers and their resistance against radiolysis or other damage.

Waste packages can be divided into self-shielding and unshielded waste packages. Unshielded waste packages would need an additional overpack during all processes where people could have access to the waste package. COVRA's current disposal concept envisages self-shielding waste packages. The performance target for the dose rate at the waste packages' surface was set to 10 mSv/h in the Opera Safety Case (Verhoef et al., 2017). This performance target is also used in this project for the development of a new waste package for rock salt. It is expected that these dose criteria will also successfully prevent radiolysis damage of other barriers as long as the DWP provides its containment function.

The safety function shielding is most important during the phases of emplacement and retrievability. Looking at present-day technology, it does not seem possible to emplace or retrieve waste packages without the possibility of letting humans interact with the waste package or the disposal machinery. Remotely controlled or partly automated devices are state of the art and can be applied for the operational processes. However, operation also includes dealing with abnormal operational situations or incidents, which require human interaction. Therefore, a shielding safety function is required with respect to the minimisation of radiological impacts on the workers.

## 2.3 Absence of Criticality

Criticality of fissile material inside the repository has to be prevented. Exclusion of criticality is a fundamental protection goal and has to be demonstrated for the most reactive arrangement of the fissile inventory, with optimum moderation and close reflection. Usually, as most reactive arrangement, a breached and water filled waste package is considered to assess potential criticality.

According to the IAEA Specific Safety Guide No. SSG-27 (IAEA, 2014), safety limits for sub-criticality should be derived on the basis of one of two types of criteria:

- Safety criteria based on $k_{eff}$ for the system under analysis, or
- Safety criteria based on the value of one or more control parameters, such as mass, isotopic composition, moderation, reflection, etc.

In nuclear waste disposal, safety criteria based on $k_{eff}$ are commonly used. Sub-criticality can be expected if the effective multiplication factor $k_{eff}$ is below 1. It is international standard (e.g. Sweden, Finland, France, Switzerland) to limit $k_{eff}$ to below 0.95 to ensure sub-criticality with adequate safety (Bollingerfehr et al., 2020).

## 2.4 Limiting Temperature

Temperature has to be limited to protect the multiple barriers in the disposal system. These are the waste package itself, the inventory (e.g. fuel cladding, glass matrix), the near-field engineered barrier system, and the host rock. Additionally, the temperature has to be limited for the operational phase to protect the workers. For operational safety, the surface temperature of the canister should be limited to 85 °C according to ADR (Agreement concerning the International Carriage of Dangerous Goods by Road) (UN ECE, 2021). This must be ensured during the phases of emplacement and retrievability, if the waste package is not covered by backfill or another overpack.

To protect the host rock, the surface temperature of the waste package must be limited. COVRA decided that the salt and backfill shall have a maximum temperature of 100 °C. In the German research project “RESUS”, the maximum heat output for a waste package surface temperature of 100 °C was calculated as 2.4 kW per waste package. For the calculation, the waste packages were expected to be placed 3 m spaced from each other in galleries, which are 35 m apart from each other (Bertrams et al., 2020).

The inventory must be protected from too high temperatures as well. For vitrified waste from reprocessing, the temperature limit is usually set to 500 °C. The temperature limit is based on the segregation temperature of the glass matrix (Bollingerfehr et al., 2020).

The research reactor fuel in the ECN Canisters has aluminium cladding. The temperature of the cladding has to be limited to prevent rupture due to creeping of metals at high temperatures (IAEA, 2003). For aluminium-cladded spent fuel, a maximum temperature of 200 °C is usually accepted (Sindelar et al., 1996).

## 2.5 Limiting Corrosion and Gas Production

It is necessary to limit the waste package corrosion and gas production. Corrosion needs to be limited to ensure that the waste package remains leak-tight for the period in which containment by the waste package is required. Furthermore, high gas pressure should be avoided to protect the geological and geotechnical barriers against damage. Thus, gas production as a result of corrosion has to be limited. For waste packages, two different approaches are possible. The first is the corrosion allowance concept, using materials with known and predictable degradation rates (corrosion-allowable canisters). Containment is then ensured by increasing the wall thickness. The second approach is to use materials that are stable and do not corrode or corrode only to a small extent (corrosion-preventing canisters). Both approaches are explained in more detail when the design process of the waste package is described later in this report.

To limit gas production, metallic corrosion should be limited. However, gas production can also occur through decomposition of organic material. Decomposition can be thermal, radiological, or chemical. Accordingly, there should be minimal use of organic materials in the canister design. Usual organic materials used for canister components are polyethylene or graphite for neutron shielding or paint or surface coatings for corrosion protection.

## 2.6 Operability

Operability of the waste package during the operational phase is strongly linked to the waste package size and weight. Operability is also linked to the safety functions containment, shielding, and limiting temperature as already described before. For operability, the waste package must have attachment points that can bear the static and dynamic mechanical impacts during the emplacement or retrieval operations. For retrieval, the degradation of the materials must also be considered. It must be ensured that the waste package can always be handled and transported in a safe manner.

The technical feasibility of emplacing self-shielding waste packages has already been demonstrated in a gallery disposal mock-up experiment in Germany. Here, POLLUX® type dummies were used for demonstration. Simulating approximately 1,000 emplacement cycles, casks with a diameter of roughly 1.6 m, a length of 5.5 m, and a weight of 65 tons were successfully transported into the mock-up disposal gallery and laid down (Filbert & Engelmann, 1995). The demonstration also included the lifting of the cask to enable the repetition of the emplacement process. Lifting of the canister and transport out of the mock-up disposal gallery are not part of the emplacement process and can be considered as partial technical demonstration of retrievability. Full demonstration of retrievability, however, would also require the backfilling process and the subsequent excavation of the waste package from the host rock, which was not demonstrated.

Further technical examples are given in Scandinavia. The so-called canister installation vehicle developed by Posiva Oy (Finland) has to fulfill similar functions as the not yet developed Dutch transport and emplacement device. Figure 2-1 illustrates the conceptual design of the canister installation vehicle.


*Figure 2-1: left: Canister installation vehicle, right: canister installation vehicle's chassis in a view obliquely from the rear. The radiation shield and the canister are missing from the picture. Both pictures based on Wendelin & Suikki (2008)*

The Finish canister installation vehicle captures the DWP at the shaft landing station, transports it into the actual disposal position and emplaces the DWP. The conceptual design as presented in Figure 2-1 is designed as crawler type vehicle. The latest designs consider a wheeled vehicle as seen in Figure 2-2. For the Dutch canister installation vehicle, a radiation shield similar to the Finnish design would not be required as the DWPs are shielded.

*Figure 2-2: Updated design of the canister installation vehicle, based on NEI (2022)*

In addition to the transport and emplacement equipment, shaft hoisting equipment with a payload of 80 tons has been developed and successfully tested (Filbert & Engelmann, 1994). Thus, it seems reasonable to expect that waste packages up to these dimensions fulfil the requirement of operability. In an industrial study, the German emplacement and shaft hoisting technology was later updated for a payload of up to 160 tons. The concept developed received a positive outlook in a review by technical experts but has not been build and tested yet (Bollingerfehr et al., 2020).

## 2.7 Manufacturability and Predictability

The waste package is to be made with present-day known construction and manufacturing methods. This ensures that the waste package can be manufactured without the need of extensive prior research and that a reliable prediction of the manufacturing process and behaviour of the waste package is guaranteed. The materials used are to be well known, predictable in their behaviour, and easily available on the market. The degradation behaviour of the waste package materials is also to be predictable for the future, so that the waste packages alteration can be predicted in the safety case. The materials used shall be already known from other disposal concepts in rock salt or from salt mining.

# 3 Description of Shielded Waste Packages and Use of Shielded Waste Packages

In the following section, the current Dutch concept for a Supercontainer for HLW as well as canister concepts for other technically advanced disposal concepts are described. The selected examples represent the technically most advanced canister concepts, which partially already meet the requirements of COVRA. These are the shielded POLLUX®-10 and ENCON-S container, the KBS-3 and UOS as well as several Nagra containers.

## 3.1 Description of the Supercontainer Concept

The so called Supercontainer (SC) concept is known from repository concepts for poorly indurated clay in the Netherlands (Verhoef et al., 2017) and Belgium (Areias et al., 2013). The Supercontainer concept is characterised by a combination of primary waste package, an overpack, a concrete shell as geochemical buffer, and a steel envelope. In addition to the shielding function, the concrete shell is to provide suitable geochemical conditions for the inner steel-based components.

The Dutch concept uses a single, uniform design for the disposal of both heat-generating as well as non-heat-generating HLW. The SC should be used for ECN canisters and for waste from the reprocessing of spent fuel.

The conceptual design of the Dutch SC is based on a multiple-barrier system consisting of a hermetically-sealed carbon steel overpack and a surrounding highly-alkaline concrete buffer. The first one is developed to retain the radionuclides. The two main functions of the buffer are

(a) to create a high pH environment around the carbon steel overpack in order to passivate the metal surface and so to slow down the corrosion propagation during the thermal phase and

(b) to provide radiological shielding during the construction and the handling of the SC.

The buffer is surrounded by a steel envelope. The SC is expected to have a diameter of 1,900 mm, a maximum length of 3,000 mm, and a maximum weight of 24 tons (Verhoef et al., 2017). Figure 3-1 shows an illustration of the Dutch Supercontainer concept.

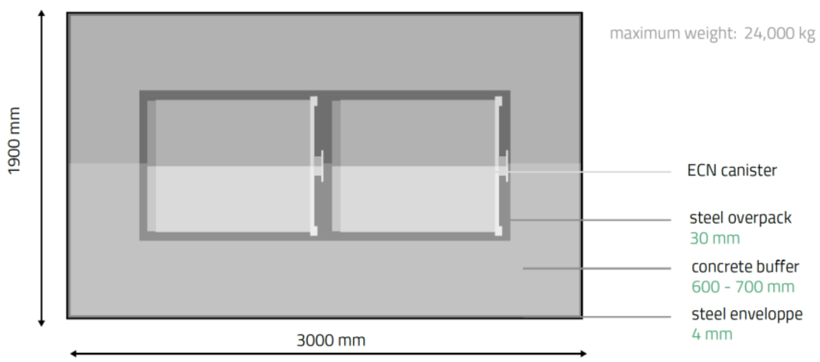



*Figure 3-1: Sketch of the Supercontainer for spent research reactor fuel (SRRF) (Verhoef et al., 2017)*

The Belgian SC consists of a 30-mm-thick carbon steel overpack, which ensures the integrity of the waste package against mechanical and thermal stresses and corrosion. The overpack is placed into a concrete buffer with a thickness of 600 mm to 700 mm. The buffer shields the radiation and is to impede the corrosion of the overpack and the inner stainless steel waste containers by providing a highly alkaline environment (cf. Iliopoules et al., 2015; Poyet, 2006). The steel will be covered by a passive oxide film (Kursten et al., 2011; Sharifi-Asl et al., 2013). The buffer consists of two layers of slightly different compositions. The outer layer (buffer phase 2), also called the ‘filler’, consists of cementitious mortar. The filler has a composition similar to that of the inner buffer layer except for the matrix aggregates, which have a finer grain size. Information on the actual recipe can be found in Areias et al. (2013). To ensure a high pH value, the buffer is made of Portland cement (CEM I) and does not contain puzzolans. Figure 3-2 and Figure 3-3 show details of the composition of the Belgian SC and the Belgian disposal concept related to the SC.

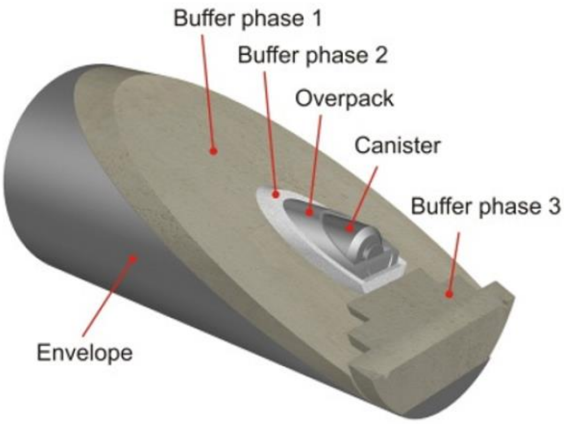



*Figure 3-2: 3D representation of a section through the Supercontainer according to Areias et al. (2013).*

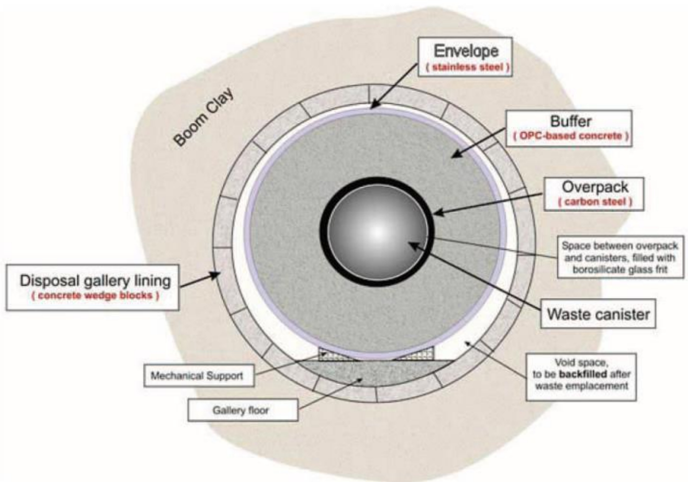



*Figure 3-3: Cross-sections through a disposal gallery and a Supercontainer according to the Belgian concept for HLW and spent fuel disposal (Sharifi-Asl et al., 2013). The SC comprises the stainless steel envelope, the concrete buffer, and the carbon steel overpack.*

## 3.2 German Waste packages: POLLUX® and ENCON

The POLLUX®-10 cask was developed for German repository concepts in a salt dome. The POLLUX® concept is a double-cover cask. It can serve as multi-purpose cask for transport, handling, interim storage, and disposal. Thus, the DWP is safe for public transport and shielded. Herold et al. (2020) summarises the relevant characteristics of the POLLUX®. A radiation dose below 0.2 mSv/h is expected on the surface of POLLUX®-10. The maximum temperature is limited to 200 °C. In addition, the cask is stable against isostatic pressure of 30 MPa after inclusion by the rock salt. The cask is corrosion-resistant in the event of access of brine for at least 500 years (Herold et al., 2020).

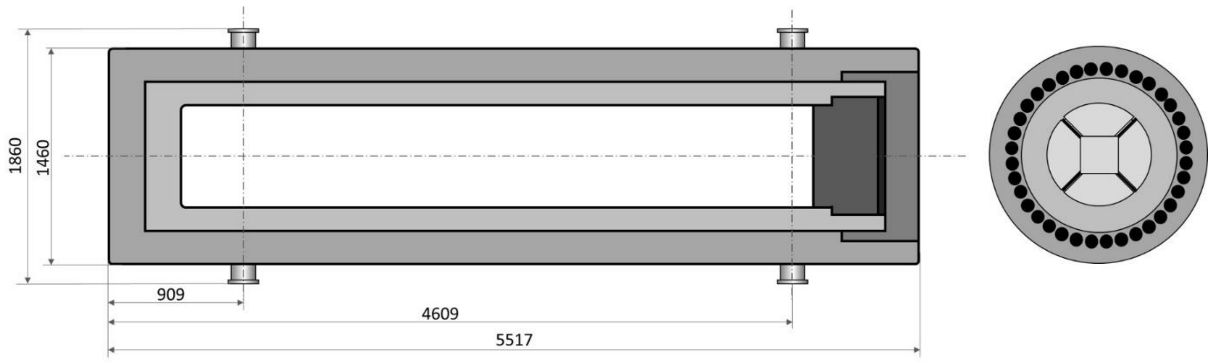
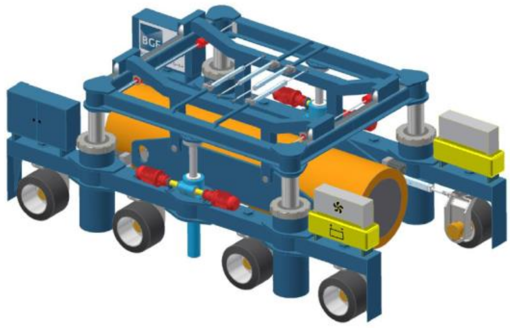



*Figure 3-4: Waste package POLLUX®-10 for PWR spent fuel. The inner cask is shown in lighter grey. The outer cask provides shielding and is shown in a darker grey shade (Bertrams et al., 2021).*

The diverse functions mentioned above are fulfilled through the double-layer cask. Therefore, the inner cask is the disposal cask and the outer container is the shielding cask. The material of the inner cask is steel (15 MnNi 6 3) with a thickness of around 330 mm. In former POLLUX® versions, a corrosion protection layer of the Nickle based alloy Hastelloy-C4 was included. After corrosion experiments showed that Hastelloy-C4 is sensitive to pitting, it was removed from the concept because there was no gain in safety. Furthermore, the inner cask also protects against mechanical and thermal stresses in a repository, while its main function is the containment of the radioactive inventory (Herold et al., 2020; Bertrams et al., 2021).

The outer shielding cask does not fulfil a sealing function but mitigates mechanical influences. Additionally, 36 moderator rods are integrated into this layer to further shield neutron radiation. The moderator rods are made of Polyethylene. The thickness of the outer cask is around 450 mm, and it is made of nodular cast iron (GGG40.3) (Herold et al., 2020; Bertrams et al., 2021).

The POLLUX®-10 cask was developed for drift disposal in salt. A respective transfer and handling device was also developed (see Figure *3-5*). Figure *3-5* also illustrates drift disposal of POLLUX® casks in clay. For rock salt, the bentonite bedding would be replaced by crushed rock salt (Bertrams et al., 2021).



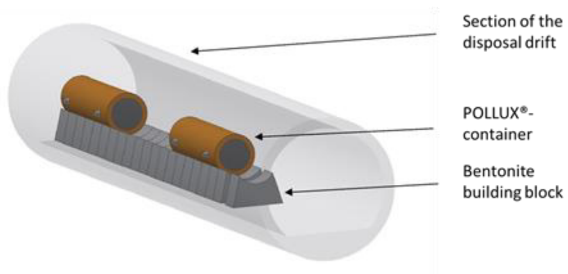



Figure 3-5: Mobile gantry crane for handling POLLUX®-10 (left) and drift disposal of POLLUX®-10 (right) (Bertrams et al., 2021).

In the German research project ENTRIA (Hassel et al., 2019), new waste package concepts for different host rocks were developed. The ENCON-S container, developed in the course of this project, is also a double-layer container (see Figure 3-6). It was developed for drift disposal in salt, too. In contrast to the POLLUX® (expected interim storage for 50 years), it was assumed that the waste would be stored for 75 years before disposal. Therefore, the assumed thermal output of the inventory mass is lower, and the ENCON-S could be designed with different wall thicknesses. The thermal output is limited to 100 °C for easier retrieval for 500 years. In addition, the thermal output of one canister is limited to 3 kW, because the temperature resistance of the moderator from polyethylene is limited (Hassel et al., 2019).

*Figure 3-6: ENCON-S container with PWR spent fuel in a canister (Hassel et al., 2019)*

In the ENCON concept, the fuel elements are disassembled and put into an inventory canister. The separated fuel rods of 10 disassembled PWR fuel elements are put into one inventory canister. The fuel elements are disassembled to enable a more compact design of the container. The inner container is made of stainless steel or zircaloy. It has a wall thickness of 10 mm and an outer diameter of 532 mm (Hassel et al., 2019).

The inventory is encapsulated through the inner container of low-alloyed steel (e.g. P235 GH / 1.3045). The inner container has the task of safely containing the radionuclides. Thus, a material that has a low tendency to local corrosion was chosen, and the container lid is welded gas-tight. The container has a wall thickness of 160 mm (Hassel et al., 2019).

The outer container made of nodular cast iron (GJS 400-18) has the tasks of shielding and of protecting the inner elements against mechanical impacts. There are shielding plates and moderator rods of polyethylene for additional shielding of neutrons and ionizing radiation. The outer container has a wall thickness of 265 mm. The lid is bolted because there is no requirement for permanent tightness. The outer container also has support rings at both ends. This distributes the forces evenly over the outer container during handling (Hassel et al., 2019).

A variant of the ENCON container, the ENCON-S/HAW can be loaded with 3 canisters of heat-generating waste from reprocessing (CSD-V), see Figure 3-7. This type has a slightly different design and dimensions (see Figure 3-7 and Table 3-1) (Hassel et al., 2019).

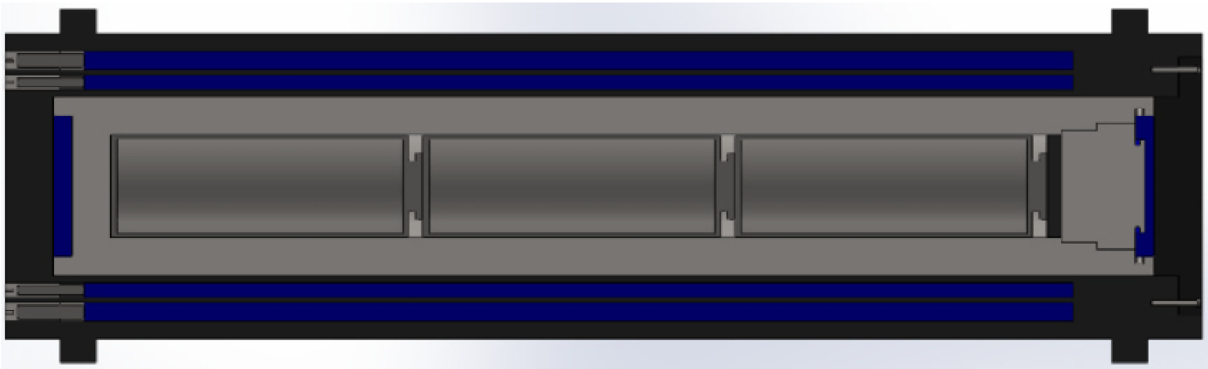


*Figure 3-7: ENCON-S/HAW container with 3 CSD-V canisters (Hassel et al., 2019)*

Finally, a handling device for all ENCON containers has been developed. A support frame encloses the support rings of the container. At the frame, there are two trunnions for container handling in the horizontal and vertical plane, similar to the POLLUX® cask (Figure *3-8*) (Hassel et al., 2019).

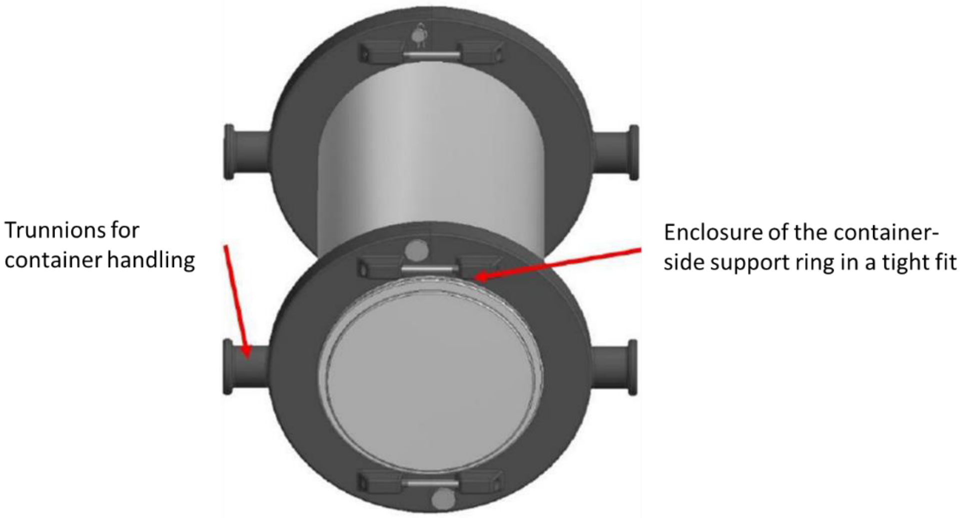



Figure 3-8: ENCON-S with support frame and trunnions (Hassel et al., 2019*)*

*Table 3-1: Technical data of POLLUX®-10 and ENCON (Herold et al., 2020; Bertrams et al., 2021; Hassel et al., 2019), ([B] – calculated from specified size; [C] – estimated from other data)*

| Description | POLLUX®-10 | ENCON-S | ENCON-S/HAW |
|---|---|---|---|
| Inventory | 10 DWR-BE | 10 DWR-BE (scattered) | 3 canister |
| Inventory mass [tons] | 4.65[B] | 6.3 | 1.5[B] |
| Total length [mm] | 5517 | 5600 | 4955 |
| Outer diameter [mm] | 1460 | 1380 (1580) | 1280 (1480) |
| Wall thickness internal cover [mm] | 322.6[B] | 160 | 160 |
| Wall material internal cover | Steel (15 MnNi 6 3) | P235 GH | P235 GH |
| Wall thickness external cover [mm] | 447[B] | 150 | 265 |
| Wall material external cover | nodular cast iron GGG40.3 | nodular cast iron GJS 400-18 | nodular cast iron GJS 400-18 |
| Empty mass [tons] | 54.95[C] | 42.9[B] | 33.7[B] |
| Mass loaded [tons] | 59.6 | 49.2 | 35.2 |

## 3.3 Corrosion-resistant Waste Packages: KBS-3 and UOS

The Swedish-Finnish KBS-3 canister is designed as a double-layer canister, see Figure 3-9. It has an inner canister made of cast iron, which provides mechanical stability. An outer hull made of copper ensures safe containment. It has a wall thickness of 50 mm. This outer hull ensures that the inventory is safely enclosed for at least 100.000 years in the granitic environment (Werme 1998).

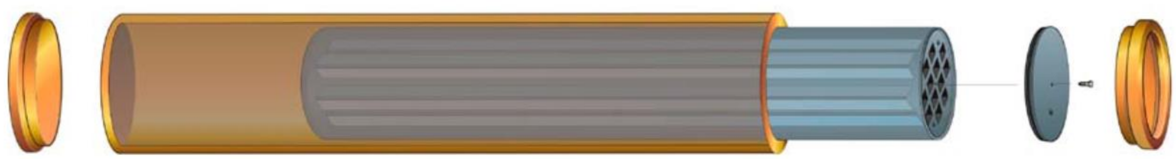


*Figure 3-9: Exploded view of the KBS-3 canister and its components (from the left: copper base, copper tube, insert, steel lid for insert, copper lid) (SKB 2010a)*

The KBS-3 is dimensioned for an isostatic pressure in the repository of 45 MPa. This includes a future ice layer with a thickness of 3 km (30 MPa) and a swelling pressure of the bentonite buffer of 7 MPa (Nilsson et al., 2005; Kuutti et al., 2012; Nilsson, 1992; Dillström, 2005; Takase et al., 1998).

In the repository, bentonite blocks surround the KBS-3 canister (Widén & Sellin 1994). The design of the bentonite blocks guarantees that loads, introduced by the canister into the bentonite blocks, will not result in a displacement or deformation of the blocks (Werme 1998). For different types of spent fuel elements, different sizes of KBS-3 canisters are designed, see Table 3-2.

*Table 3-2: Variants of KBS-3 packages and corrosion-allowable UOS package for 7 WWER-440-BE (Pospísková, 2016) ([B] – calculated from specified size; [C] – estimated from other data; [D] – no data)*

| **Description** | **KBS-3 12 BWR - SKB** | **KBS-3 4 PWR - SKB** | **KBS-3 12 WWER - Posiva** | **KBS-3 12 BWR - Posiva** | **KBS-3 4 PWR – Posiva** | **UOS 7 WWER – igd tp** |
|---|---|---|---|---|---|---|
| Inventory (fuel elements) | 12 BWR | 4 PWR | 12 WWER | 12 BWR | 4 PWR | 7 WWER |
| Inventory mass [tons] | 2.1[B] | 1.86[B] | 1.4 | 2.2 | 2.1 | 0.82[C] |
| Total length [mm] | 4835 | 4835 | 3552 | 4752 | 5223 | [D] |
| Outer diameter [mm] | 1050 | 1050 | 1050 | 1050 | 1050 | 805 |
| Wall thickness external cover [mm] | 50 | 50 | 50 | 50 | 50 | 120 |
| Wall material external cover | Copper | Copper | Copper | Copper | Copper | Carbon steel CSN 422707.9 |
| Empty mass [tons] | 24.7 | 27.6 | 14.2 | 20.9 | 26.8 | [D] |
| Mass loaded [tons] | 26.8 | 29.46 | 15.6[B] | 23.1[B] | 28.9[B] | [D] |

The radiation dose at the container surface is assumed to be 1 Gy/h, or 1 Sv/h. During handling and disposal, the personnel is protected by a separate transport cask. Additionally emplacement technology that allows automated disposal operation is used. For details of the Finnish disposal technology, see Figure 3-10 (Werme, 1998; Lahti, 2016).

The surface temperature of the canister is calculated to be below 100 °C. In the repository, the canisters are arranged in such a way that the other barriers, mainly the bentonite buffer, are not damaged by the thermal output (SKB, 2010b). The KBS-3 system is designed in a way that retrieval is theoretically possible during a period of 40 years after disposal (Werme, 1998).

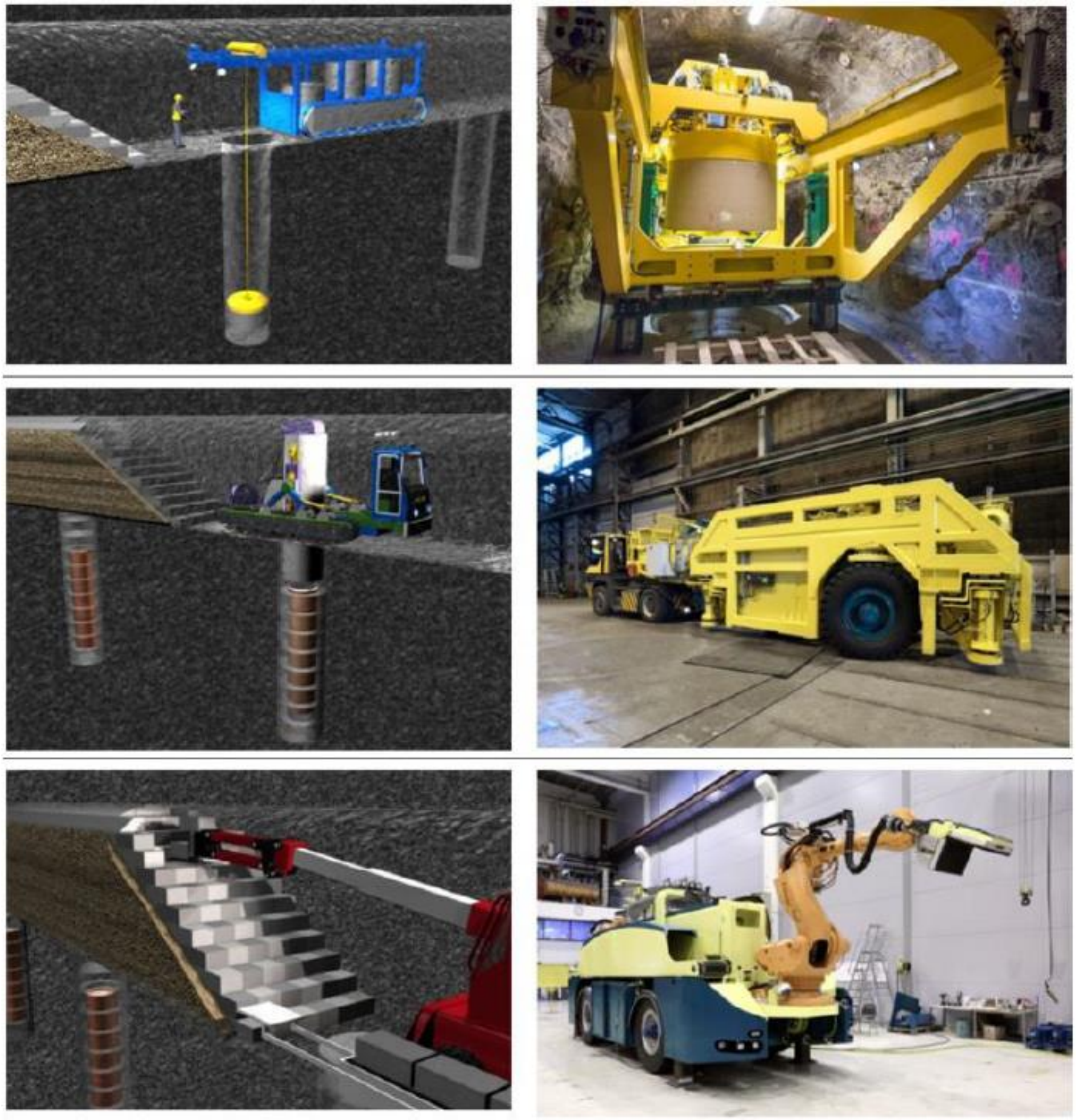


*Figure 3-10: Prototypes of disposal devices for handling KBS-3 canisters in the Finnish repository project ONKALO. Above: Placement vehicle for the bentonite buffer; middle: Transfer and repository vehicle; bottom: Special crane for backfilling the disposal gallery (Lathi, 2016).*

In the Czech Republic, the UOS canister is based on the KBS-3 canister. UOS was developed until 2016 also for granitic environment. Contrary to the KBS-3, the UOS is partly corrosion-allowable. Currently, a corrosion protection layer is not considered. A thick outer shell of carbon steel serves as calculable corrosion barrier. Further examinations will show if this is sufficient. If not, copper or titanium could be used as further barrier. In addition, the inner shell of stainless steel was reinforced to 10 mm and should provide containment for around 10.000 years (Pospísková, 2016). Figure 3-11 shows a view of the UOS-type waste package.

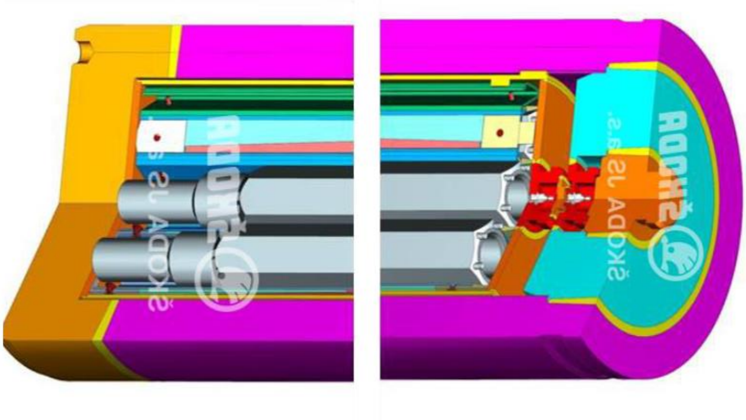


*Figure 3-11: Czech Republic´s waste package UOS, corrosion-allowable, for 7 WWER spent fuel elements (Pospísková, 2016)*

### 3.4 Nagra Canister

In Switzerland, Nagra developed a concept for a waste package for disposal in clay. The Nagra canister has to provide containment for at least 1,000 years (ENSI, 2009).

Contrary to the KBS-3, the Nagra canister does not have an outer shell of copper. It is assumed that the copper cover is not required in clay. Instead, the thickness of the outer shell, which will be made of steel-based materials, has been increased, which ensures the necessary corrosion resistance (Johnson & King 2003). Figure 3-12 shows an illustration of the Nagra canister.

In the United Kingdom, the Nagra canister is also considered for the use in rock salt (RWM, 2016).

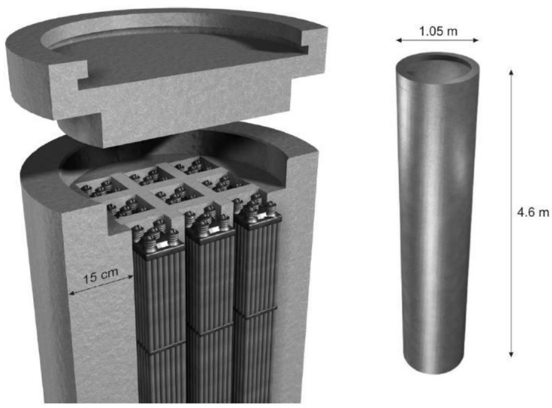
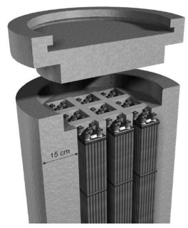



*Figure 3-12: Swiss steel canister for SWR-fuel elements (Johnson & King 2003)*

Nagra developed different versions of the Nagra canister. Table 3-3 shows the technical data of different versions for different waste inventories.

*Table 3-3: Technical data of canister references from NAGRA (Patel et al., 2012; Stein, 2014; Johnson & King 2003; Herold et al., 2020) (B – calculated from specified size; C – estimated from other data; D – no data)*

| **Description** | **HLW-Canister Nagra** | **BE-Canister-Nagra** | **LB HAA NAGRA** | **LB BE-DWR NAGRA** | **LB BE-SWR NAGRA** |
|---|---|---|---|---|---|
| Inventory | 2 CSD-Canisters | 9 BWR-SF | 2 CSD-Canisters | 3 - 4 PWR-SF | 9 BWR-SF |
| Inventory mass [tons] | 0.96 | 4.96 | 0.96[C] | 1.4 … 2.1[C] | 1.6[C] (9 SWR) |
| Figure | [D] |  |  |  |  |
| Total length [mm] | 3233 | 5350 | 3070 | 5000 | 5190 |
| Outer diameter [mm] | 720 | 1050 | 720 | 1050 | 1050 |
| Wall thickness external cover [mm] | 140 | 140 | 140 | 140 | 140 |
| Wall material external cover | Cast steel | Cast steel | steel | steel | steel |
| Empty mass [tons] | 6.4 | 17.1 | 6.276 | 18.042 | 19.38 |
| Mass loaded [tons] | 7.36 | 22.06 | 7.23[B] | 19.44 … 20.14[B] | 20.98[B] |

A wall thickness of 120 mm (for fuel elements) or 140 mm (for HLW) restricts the radiation dose at the surface to 1 Sv/h. This reduces corrosion due to radiation. In addition, a shielding container is used to protect persons and to reduce the radiation dose to 2 mSv/h at the container surface. The handling of the canister using such a shielding container is shown in Figure 3-13 (Patel et al., 2012).

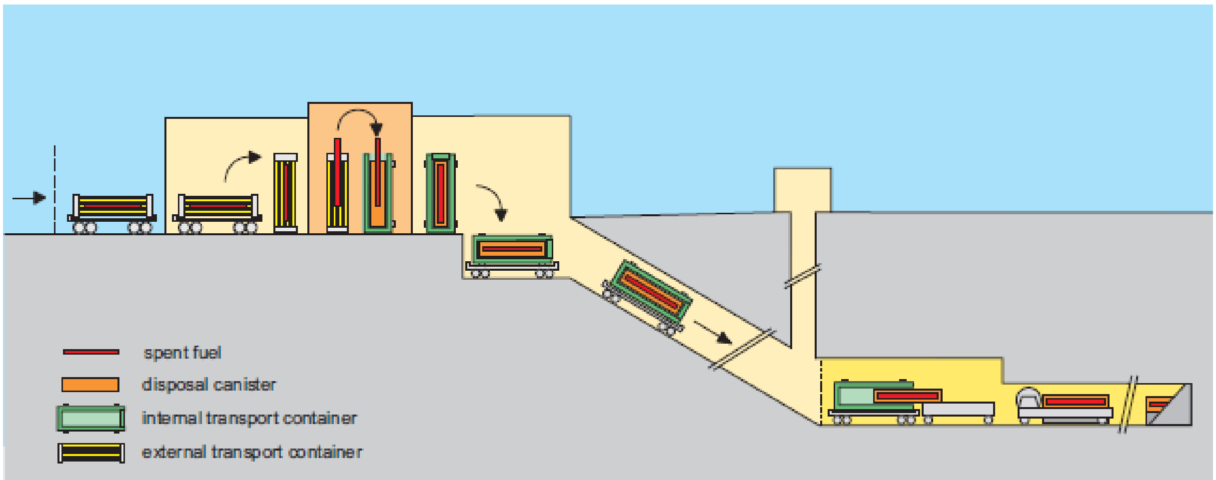



*Figure 3-13: Handling plan of the Swiss waste package (Patel et al., 2012)*

## 3.5 Advantages / Disadvantages of the Different Concepts / Shielded Waste Packages

After the description of the different waste package designs, the general advantages and disadvantages of shielded waste packages will be discussed in this section.

In former Dutch disposal concepts, borehole disposal of unshielded waste packages was considered as disposal option. In the “Eindrapport 1996 METRO I” (Heijdra & Prij, 1997), disadvantages of waste packages or overpacks that are fully shielded and long-term resistant against corrosion and rock pressure were stated:

- Higher costs due to use of material. The reduction in costs due to easier storage and handling of a shielded waste package would not outweigh the production costs.
- Potentially valuable materials are withdrawn from the material flow due to disposal.
- Possible problems with formation of corrosion gases due to anaerobic corrosion and the high amount of disposed material
- In the case of a heavy or bulky overpack, the shaft hoisting, transport equipment, and infrastructure will become more expensive due to higher necessary payloads.

Using shielded waste packages, a lot of material is brought inside the repository together with the nuclear waste. If the waste package material can corrode under the geochemical conditions of the repository, gas generation can start and possibly affect the long-term safety due to the increase in gas pressure. High gas pressures in the repository can lead to cracking of the host rock and can form flow paths. As a result, contamination of the biosphere may be possible. The high mass of waste package materials that is disposed with the waste and could be converted into gas is the strongest argument against a self-shielding waste package. Possible solutions for this problem are the reduction of organic material in the waste package to prevent further gas generation from other sources and to decrease the appearance of water in the disposal system. Within the initial conditions and the expected reference scenario for long-term evolution, a repository in rock salt is considered as dry, see e.g. Bertrams et al. (2020). Thus, corrosion and gas production is limited. Water intake during long-term development represents scenarios with a very low probability. Such scenarios will include a malfunction of several components of the EBS, see e.g. (Müller-Hoeppe et al., 2012). The appearance of water in the repository can be further decreased by:

- drying of the waste
- limiting the design temperature so that water included in the salt minerals is not set free
- using sensitive mining methods, such as mechanical excavation, to limit the excavation damaged zone (EDZ) in the host rock, which could function as flow path for intruding fluids

Another aspect also related to the high amount of material are material costs for shielded waste packages due to the higher material volumes. Partly, this cost increase is reduced by the use of cast iron or steel instead of expensive and corrosion-resistant materials such as titanium or copper. However, it has to be highlighted that the use of iron-based materials for shielded waste packages is not driven by economic considerations. The main driver is the corrosion and the two different concepts how to handle corrosion, see section 2.5.

One of the mentioned cost aspects refers to the larger weight of shielded waste packages, which requires higher payloads for transport, disposal, and shaft hoisting equipment. This does not take into account that for the transport and disposal of an unshielded waste package, a shielded overpack would be necessary. It can be assumed that this shielded waste package would have a similar wall thickness and so have a similar weight as the shielded waste package. Therefore, it can be expected that the necessary payload of the equipment must be very similar. For example, both German concepts for disposal in rock salt are compared: The unshielded “Brennstabkokille” (BSK-3) and the shielded POLLUX®-type cask. Filled with the fuel rods of 10 PWR fuel elements, the POLLUX® cask has a mass of slightly less than 60 tons. The shielded overpack together with the BSK-3 has a mass of 52 tons filled with the fuel rods of only three PWR fuel elements. Thus, the proportion of total mass to mass of the inventory is better for the shielded waste package, and the difference in payload of the disposal and hoisting equipment is small. This statement applies in general. Even if no mobile overpack is used, a certain kind of shielding has to be installed in the disposal and hoisting equipment and the total mass of the devices increases, see e.g. Posiva’s canister transfer and installation vehicle.

The advantage of a shielded waste package is the shielding itself. The shielding allows the direct interference of humans with the waste package. Relevant tasks are:

- control after encapsulation
- disposal
- backfilling
- retrieval
- possible repair and work during malfunctions or breakdown of disposal equipment

The shielding further increases the long-term safety due to preventing radiolysis effects in the surrounding host rock and backfill. However, in rock salt, this is not that relevant since the resistance of the host rock against radiation is high (RSK, 2006).

Due to the necessary greater wall thicknesses of shielded waste packages, these waste packages can be seen as more robust during emplacement and especially during excavation for retrieval.

As a result, advantages of shielded waste packages outperform the disadvantages. The shielding improves operational safety, simplifies encapsulation, disposal and retrieval, and makes the whole disposal system more robust against malfunctions. The analysis of disadvantages shows that the difference in weight of shielded and unshielded waste packages together with the overpack is small. In addition, the expected corrosion in the base (expected) scenario is small and therefore, no crucial gas generation is expected.

Table 3-4 summarises the mentioned advantages and disadvantages.

*Table 3-4: Summary table of advantages and disadvanteges of shielded waste packages*

| **Advantages** | **Disadvantages** |
|---|---|
| • Reduction in operational costs due to easier storage and handling<br>• The shielding allows the direct interference of humans with the waste package (during normal and abnormal operation as well as during incidents).<br>• Prevent radiolysis effects in the surrounding host rock and backfill<br>• More robust against impacts (e.g. mechanical) due to wall thickness | • Higher costs due to use of material<br>• Potentially valuable materials are withdrawn from the material flow due to disposal.<br>• Possible problems with formation of corrosion gases due to anaerobic corrosion and the high amount of disposed material<br>• In the case of a heavy or bulky overpack, the shaft hoisting, transport equipment, and infrastructure will become more expensive due to higher necessary payloads. |

# 4 Assessment of Transferability of Dutch Waste Package Concept for Clay to Rock Salt

For the disposal of HLW in the Netherlands, a waste package was developed in the Research Programme Disposal Radioactive Waste (OPERA). This waste package in form of a Supercontainer (SC) is based on the Belgian Supercontainer concept. In the present project, we assessed the suitability of the OPERA Supercontainer concept for disposal in rock salt.

## 4.1 Description of Concrete Degradation in Rock salt

According to Chapter 3.1, the SC contains a concrete buffer in order to shield against radiation (workers protection) and to provide a highly alkaline environment in order to passivate steel surfaces and to reduce corrosion rates. With the aim of ensuring a high pH, the buffer could be made of certified sulphate-resistant Portland cement and without reactive substances that could lower the pH value, such as pozzolan. The decision to use a buffer and the development of the buffer took into account the chemical boundary conditions in poorly indurated clay, as present in the Boom Clay formation. In Belgium, the Boom Clay is studied as the reference formation for geological disposal of high-level radioactive waste and spent fuel. From a mineralogical/geochemical point of view, the Boom Clay consists mainly of clay minerals, quartz, and feldspars. Minor amounts of pyrite and carbonates are also present, see e.g. (Le et al., 2011).

The Boom Clay formation is characterised by a low salt content, and the Boom Clay pore water is a $NaHCO_3$ solution. Sulfate ions are of particular relevance regarding the assessment of buffer stability (Table 4-1). This fact is also illustrated in Figure 4-1, according to which a weak to moderate chemical attack is to be expected from the reaction with dissolved sulfate ions. Sulfate attacks calcium aluminate hydrates, forming calcium sulfoluminates, such as ettringite. With the aim of reducing the extent of sulphate attack, a sulphate-resistant cement with a low aluminate content is used. Calcium sulphate hydrate and portlandite (calcium hydroxide, $Ca(OH)_2$) do not react with sulfate, so that a high pH value of the buffer is guaranteed in the long term.

Govaerts & Weetjens (2010) carried out calculations concerning the transport of aggressive species through the concrete buffer towards the overpack. The calculations showed that some anions (e.g. $Cl^-$ and $I^-$) are strongly retarded by concrete phases, which could also be the case for sulphide, sulphate, and thiosulphate. The processes could delay the arrival of these species at the overpack by many thousands of years. One calculation considered the formation of ettringite.

*Table 4-1: Concrete aggressiveness classes (XA1, XA2, XA3) according to DIN 4030-1 (as well as information on the properties of Boom Clay pore water (De Craen et al., 2004; De Craen et al., 2006). Index*: Due to the presence of calcite.*

| | | **Boom Clay** | **Weak aggressive** | **Moderate aggressive** | **Strong aggressive** |
|---|---|---|---|---|---|
| | | | XA1 | XA2 | XA3 |
| pH value | [–] | > 7.5 | 6.5 – 5.5 | 5.5 – 4.5 | 4.5 – 4.0 |
| Lime solving carbonic acid | [mg/l] | Negligible* | 15 – 40 | 400 – 100 | > 100 |
| Ammonium ($NH_4$) | [mg/l] | < 5.1 | 15 – 30 | 30 – 60 | 60 – 100 |
| Magnesium (Mg) | [mg/l] | < 100 | 300 – 1,000 | 1,000 – 3,000 | > 3,000 |
| Sulfate ($SO_4$) | [mg/l] | < 800 | 200 – 600 | 600 – 3,000 | 3,000 – 6,000 |

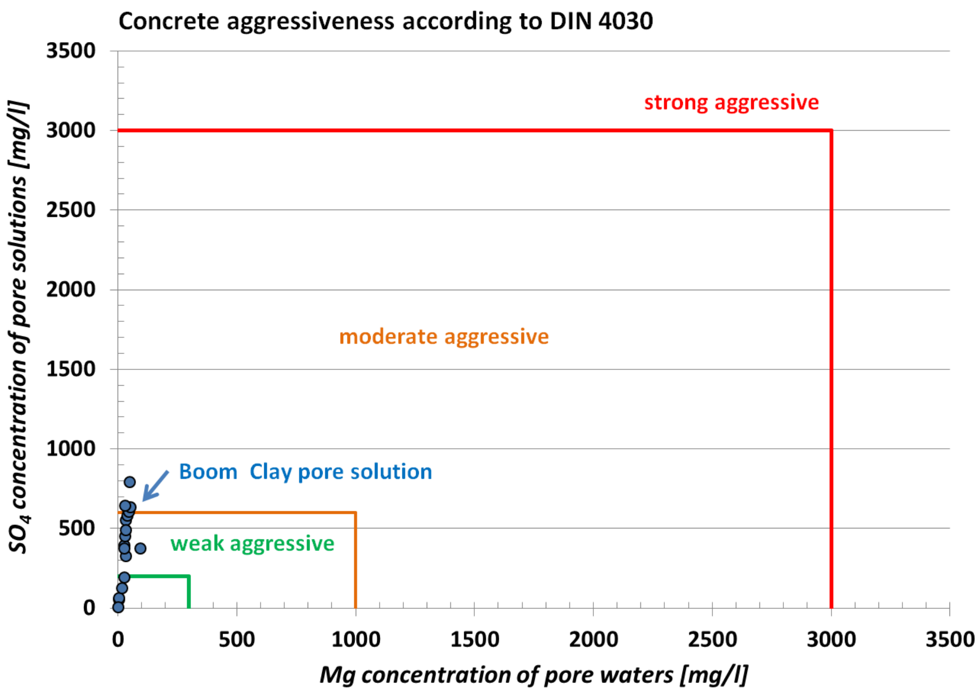



*Figure 4-1: Sulfate versus magnesium content of pore water in Boom Clay. The horizontal and vertical lines mark the limit values of the concrete aggressiveness classes according to DIN 4030-1 and NEN EN 206, Note: the salt solutions IP09 and IP21 are not shown, the concentration is about one order of magnitude outside the domain. IP: Invariant points for the salt solutions of the so-called hexary system of oceanic salts.*

Table 4-2 is intended to give an impression of the range of composition of solutions that can be expected in Zechstein salt domes. An IP09 (IP: invariant point, cf. Usdowski & Dietzel 1998) solution results from the dissolution of rock salt that consists of the minerals halite, anhydrite, and polyhalite. An IP21 solution results from the dissolution of the potash rock carnallitite or some hard salts (cf. Raith et al., 2015).

*Table 4-2: Composition of salt solutions that can be expected in Zechstein formations in the Netherlands (IP09, IP21). Concentrations in milligrams per litre (mg/l). The calculated values have been rounded to an accuracy of 100 mg/l. IP: Invariant points for the salt solutions of the so-called hexary system of oceanic salts.*

| | $Na^+$ | $K^+$ | $Mg^{2+}$ | $Ca^{2+}$ | $Cl^-$ | $SO_4^{2-}$ |
|---|---|---|---|---|---|---|
| IP09 | 71,700 | 26,800 | 18,900 | 200 | 169,100 | 28,500 |
| IP21 | 9,300 | 22,100 | 87,200 | 200 | 271,200 | 23,700 |

A comparison of the numerical values in Table 4-2 with the limit values shown in Figure 4-1 shows that the solutions to be expected attack cementitious building materials very strongly. In contrast to the pore solutions in Boom Clay, in addition to the sulfate attack, the cement stone degrades through a reaction with $Mg^{2+}$ ions.

All phases of the cement stone (the cement matrix) that contain calcium are affected by this attack and thus, the calcium sulphate hydrates and portlandite, which are important for maintaining the pH value, are affected. A so-called decalcification of the cement stone phases occurs. In other words, a calcium leaching occurs.

Moreover, hydroxide ions ($OH^-$) present in the pore solution of the cement stone are precipitated as magnesium hydroxide ($Mg(OH)_2$). The following equations describe the reactions. Calcium silicate hydrates (CSH) react with dissolved magnesium ions ($Mg^{2+}{}_{aq}$) to form magnesium silicate hydrates (MSH).

$$Mg^{2+}{}_{aq} + Ca(OH)_2 \rightarrow Ca^{2+}{}_{aq} + Mg(OH)_2 \quad (1)$$

$$MgSO_4 + Ca(OH)_2 + 2\ H_2O \rightarrow CaSO_4 2H_2O\ (gypsum) + Mg(OH)_2 \quad (2)$$

$$Mg^{2+}{}_{aq} + CSH \rightarrow Ca^{2+}{}_{aq} + MSH \quad (3)$$

The pore solution in a cement stone containing portlandite has a pH of at least 12.5 due to the solubility of portlandite and the presence of NaOH and KOH. The reactions with $Mg^{2+}$ result in a significant decrease in pH. Thus, the solubility of $Mg(OH)_2$ results in a pH value of around 10.5 (cf. Baston et al., 2012).

The ability of solutions to passivate (cover the surface with an oxide layer) and thus protect the surface of steel depends on the pH value and the chloride concentration of the solutions. $Cl^-$ ions change important properties of the solution and influence the corrosion reactions (see Beck & Burkert, 2011):

$$Fe^{2+} + 2Cl^- + H_2O \rightarrow Fe(OH)Cl + HCl \quad (4)$$

The relationship between $Cl^-$ concentration and pH value is shown in Figure 4-2. The diagonal black line shows that with increasing chloride concentration, an increasing pH is required to prevent steel corrosion. A pure NaCl solution contains about 5.4 mol $Cl^-$/l and an IP09 solution about 4.8 mol $Cl^-$/l. The pH of the pore solution in the cement paste will be between about 10.5

and 12.5. According to Figure 4-2, corrosion of the steel is to be expected in the event of contact with salt solution, regardless of the pH value. From this fact, it can be concluded that the buffer in saline solutions has no significant protective function.

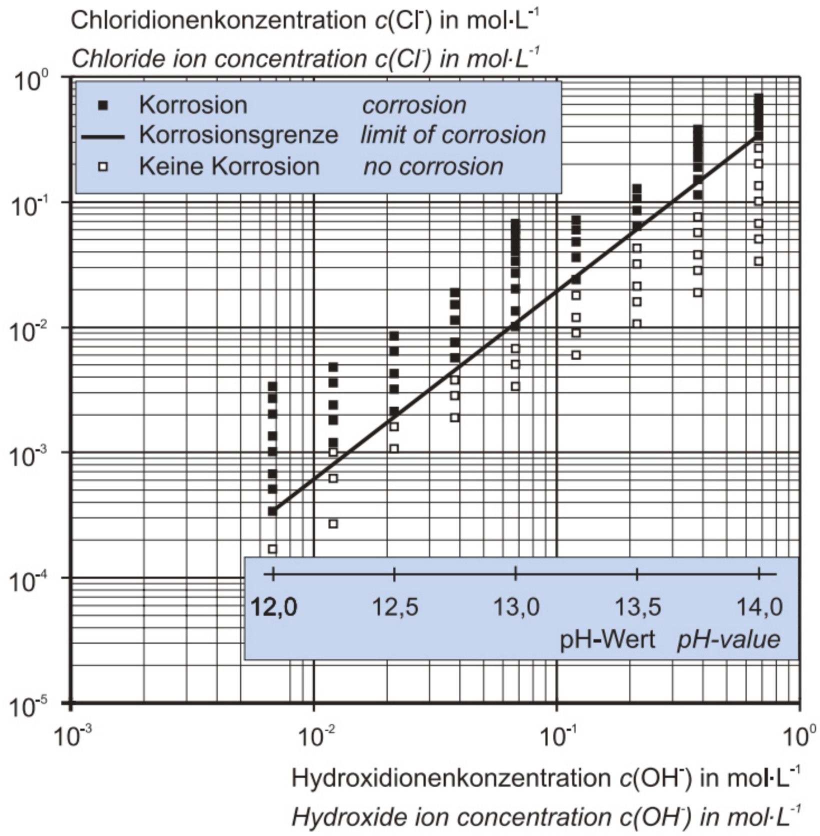



*Figure 4-2: Results of electrochemical investigations of steel in alkaline chloride solutions relative to pH-value (Breit, 2000; cf. Breit 2003; Breit et al., 2011). For example, the IP9 solution given in Table 4-2 has a $Cl^-$ content of 4.8 mol/l.*

Figure 4-3 shows exemplary corrosion rates of steel in oxygen-containing brine. According to this, the corrosion rates first increase with increasing NaCl content and then decrease after a maximum value.

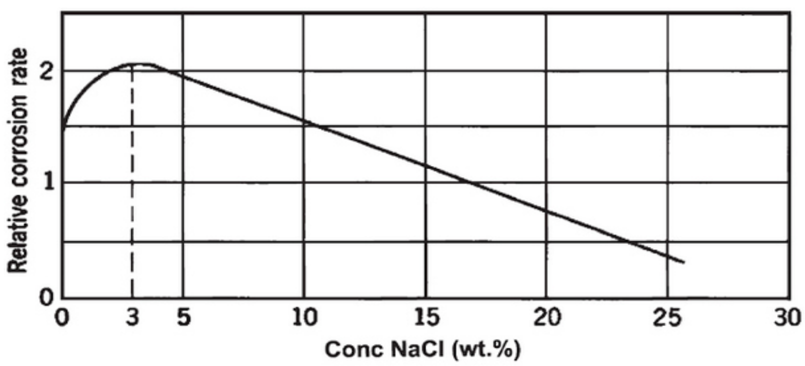



*Figure 4-3: Corrosion rate of steel in oxygen-saturated NaCl solution at 21.1 °C according to Uhlig & Revie (1985) (cf. Pollitt, 1926, Fig. 7; Sisler & Peterson 1949, Appendix 2). A solution saturated with halite contains about 26.6% by weight of NaCl at 30 °C.*

In principle, this relationship also applies to other salt solutions. The reason for the decrease in the corrosion rate is that the solubility of oxygen decreases with increasing salt concentration (Figure 4-4, Figure 4-5). Figure 4-6 illustrates how the amount of oxygen affects corrosion rates. In addition, the electrical conductivity increases with increasing salt concentration and decreases with high salt concentrations.

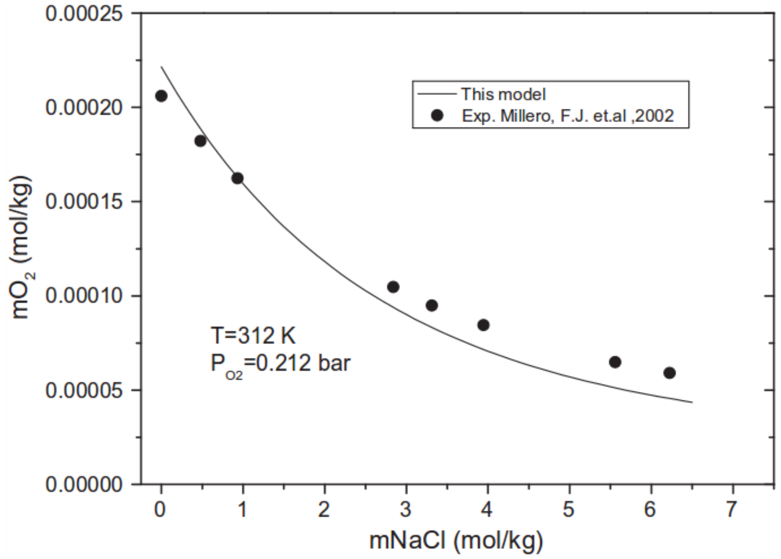



*Figure 4-4: Solubility of $O_2$ as a function of the concentration of NaCl solutions at a temperature of 312 K (39 °C) according to Geng & Duan (2010) (cf. Hasan, 2010 and Millero et al., 2002).*

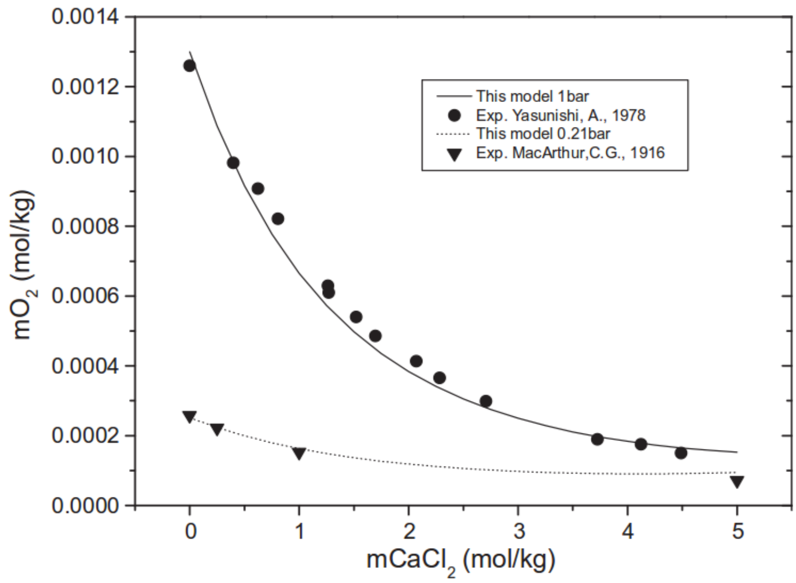



*Figure 4-5: Solubility of $O_2$ in $CaCl_2$ solutions at different pressures according to Geng & Duan (2010) (cf. Yasunishi, 1978 and MacArthur, 1916)*

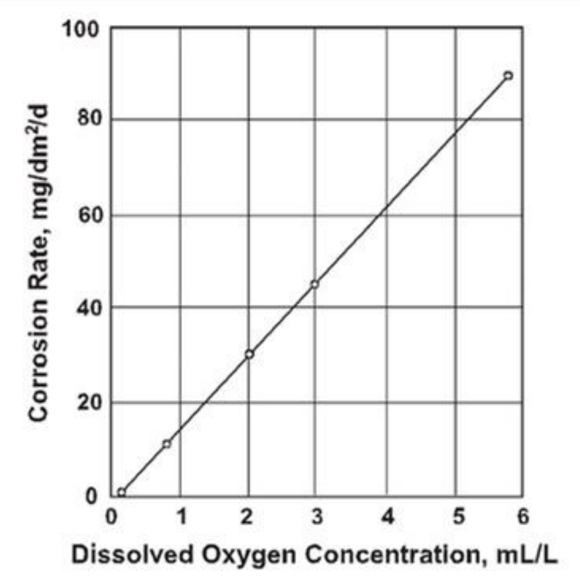



*Figure 4-6: Effect of oxygen concentration of the corrosion of low carbon steel in slowly moving water containing 165 ppm $CaCl_2$ (Tsang & Apps 2005). The 48-hour test was conducted at 25 °C.*

## 4.2 Optimisation of the Waste Package design

The explanations in chapter 4 show that the use of a cement-based buffer in the salt does not result in a significant gain in safety. From this follows the need to optimise the waste package design. One possibility is to exchange the cement-based concrete for an alternative material that ensures a favourable chemical environment. This approach would result in no or no significant changes in the dimensions of the waste package and no significant changes in the production process.

A material system often used in salt formations for backfilling and sealing applications are magnesia binders. This system is based on the reaction of caustic magnesium oxide powder and/or magnesium hydroxide. The binders are mixed with concentrated magnesium chloride solutions. The hardening of the mixtures results from the crystallisation of magnesium oxychlorides (MOC) consisting of magnesium hydroxide, magnesium chloride, and crystal water. The formulations can contain inert and reactive fillers as well as aggregate. Recipes with barite have been developed (Engelhardt, 2019; Engelhardt et al., 2019), which have a shielding effect against radioactive radiation. Such mixtures can be used to provide radiological shielding during handling operations. However, the pH values of suitable mixtures are in a very slightly alkaline range (maximum pH around 8.5 to 9.5, based on Qiao et al. 2014). The pore solutions are very highly concentrated magnesium chloride solutions. Consequently, no positive influence on the stability of the steel is to be expected. A concrete buffer in salt environment cannot provide the same benefits for the steel components as in other host rocks or geochemical environments.

If there is no concrete buffer, a sufficient shielding effect of the container can also be achieved by adapting the steel casing (overpack). An unacceptable decrease in chemical stability can also be ruled out in rock salt. As a result, waste containers designed for the host rock salt do not have a concrete buffer. With regard to the SC, the omission of the buffer results in a simplification of the production process, a shorter production time, and possibly also a reduction in costs.

The geochemical properties and processes described are relevant for all kinds of concrete components within the repository. In the salt formations of the Zechstein, salt solutions with very high levels of magnesium and sulfate are present. In addition, ammonium can occur in dissolved form. The solutions are neutral or slightly acidic. According to the findings of concrete technology, the solutions can be classified as highly aggressive. This statement is supported by many chemical calculations. Due to the fact that several components react with the cement paste, several corrosion processes occur simultaneously. From the known principles, it can be deduced that mixtures with a comparable ratio of water to cement corrode the fastest, when the mixtures were produced with Portland cement (CEM I) and without reactive additives (pozzolan, latently hydraulic substances). This fact is based on the occurrence of Portlandite (calcium hydroxide) in the cement paste. In addition, it is known that the rate of corrosion increases with the ratio of water to cement (or the sum of Portland cement clinker and reactive additives). The number of pores in the cement stone, the permeability of the cement paste, and the rate of diffusion of corrosive substances increase with the water-cement ratio. The statements can be underlined by experimental studies such as Wakeley et al. (1994), Kienzler et al. (2003) or Kienzler et al. (2016). In summary, all experiments show that cementitious materials react very quickly with the corrosive substances. With respect to the long-term safety, this has to be considered in the safety concepts, see section 1.1 and especially Figure 1-3. The containment function of concrete-based waste packages is limited and will have to be covered by other barriers during the long-term evolution.

Corrosion as a result of a Ca-Mg exchange dominates. Experiments show that corrosion cannot be completely prevented. In relation to the very long periods of time that have to be taken into account in the disposal of radioactive waste, the corrosion processes cannot be neglected, see e.g. Herbert et al. (2004), Janschik et al. (2016a) and Janschik et al. (2016b).

# 5 Waste Package Design

For the design of a waste package for COVRA, a general waste package type has to be selected first. Options are the four waste package types described before (see chapter 3) and as fifth option the Supercontainer concept. The Supercontainer concept is considered for the disposal concept in poorly indurated clay. The Supercontainer itself is also described in chapter 3. Why a waste package made of concrete is unusable in rock salt is further described in chapter 4.

*Table 5-1: Comparison of different waste package concepts*

| **Concept** | **Containment** | **Self-Shielding** | **Criticality Prevention** | **Temperature** | **Operability** | **Corrosion resistant** |
|---|---|---|---|---|---|---|
| POLLUX® | 500 a | yes | yes | 200 °C | Trunnions | no |
| ENCON | 500 a | yes | yes | 200 °C | Trunnions | no |
| KBS-3 | 100,000 a | no | yes | 100 °C | Notch | yes |
| UOS | 10,000 a | no | yes | 100 °C | Notch | no |
| Supercontainer (NL) | 1,000 a | yes | yes | 100 °C | circumferential | no |
| Nagra Canister | 1,000 a | no | yes | 150 °C | Trunnion | no |

Table 5-1 shows a summary of the waste packages and their safety functions. The analysis shows that none of the waste packages can fulfil all the safety functions needed by COVRA. Also, the way of handling and the disposal of all the waste packages are different to COVRA´s concept. The waste packages are to be either laid down in galleries or to be put into vertical boreholes. So, none of the waste packages can be directly transferred to the Dutch disposal concept. The development of a new waste package is necessary, if the Dutch disposal concept remains unchanged.

For the design of the waste package, a design similar to the Nagra-type waste package has been chosen. Waste packages like the KBS-3 or UOS with an outer corrosion-resistant barrier were discarded because corrosion resistance to guarantee a long lifetime is not needed in the Dutch concept. Also, corrosion-resistant materials like stainless steel, copper, or titanium are more expensive than materials with lower corrosion resistance. It is expected that the waste package will be made of mild or carbon steel.

Carbon steel is considered as waste package material in the United Kingdom as well. King & Watson (2010) identified the following benefits of carbon steel:

(i) good combination of strength and ductility
(ii) generally good corrosion properties in the expected repository environment
(iii) extensive experience with fabricating and sealing large cylindrical objects
(iv) relative abundance and low cost of the material

The waste package will be manufactured by forging. The waste package is closed with a lid. The lid is first inserted into the waste package to temporarily close it inside the encapsulation plant. Later, the waste package will be welded airtight. It is expected that welding will be done by electron beam welding or by submerged arc welding. The waste package is equipped with only one lid. It is expected that this is sufficient because the Dutch waste inventory is already packed into leak-tight canisters. Thus, there is no risk of contamination when the waste package is transported between different stations in a future encapsulation plant.

In the German research project RESUS, the maximum possible heat output for a waste package in rock salt to guarantee a temperature below 100 °C is 2.4 kW (Bertrams et al., 2020). In Verhoef et al. (2016), a maximum heat output for the ECN canisters of 33 W per canister is given at the time of disposal (approximately in 2130); the CSD-V canisters will have a heat output of around 200 W per canister. Technically, a waste package could hold around 12 CSD-V canisters or 72 ECN-Canisters. A waste package containing that much inventory would be too large and too heavy to handle in the planned galleries and shaft. Therefore, the inventory for both waste types is reduced. It is planned that the waste package will either hold two ECN Canisters or six CSD-V canisters. In this case, the expected weight lies in a range comparable with other waste canister concepts, see section 3. The expected heat output per waste package is then 66 W for the package holding two ECN canisters or 1.2 kW for the waste package holding six CSD-V canisters. So, it is ensured that the temperature limit of 100 °C as defined by Bonté et al. (2012) is not reached for both inventories.

The possibility of criticality occurring within the waste package, especially for the ECN canisters containing spent fuel from research reactors, has to be further investigated in the future. For the development of the conceptual design, it is assumed that criticality cannot occur. In the ECN canisters, the spent fuel elements are loaded in baskets of borated steel, which ensure that criticality cannot occur (Verhoef et al. 2016). Further scenarios that might be relevant for criticality are connected to corrosion and the availability of brine. Fischer-Appelt et al. (2013) summarised the investigation of criticality for the former Gorleben Site in Germany. The exclusion of criticallity was investigated for single POLLUX® casks and BSK canisters. It was concluded that for low enriched spent fuel, criticality can be excluded as long as no water or brine is available. In the presene of brine, the nuclide $^{35}$Cl acts as neutron absorber and provides a reducing effect to the activity, respectively the neutron multiplication factor. This statement is relevant for the Dutch repository and waste package as well.

The planned inventories affect the necessary inner geometry of the waste package, see Figure 5-1. For the ECN canisters, it is planned to stack two of them into one waste package. Therefore, an inner cylinder with at least 730 mm diameter and 2,472 mm height is necessary. For the CSD-V, it is planned to stack two layers of three CSD-V canisters in the waste package. Therefore, the inner diameter of the waste package has to be at least 945 mm, and the height of the inner cylinder has to be at least 2,670 mm. Both waste types need different inner spaces of the waste packages. Therefore, it was decided that two different waste packages will be developed. The difference between them is only geometrical, with the waste package for CSD-V being larger in diameter and height. The two ECN canisters will be stacked on top of each other; in the variant for CSD-V canisters, a cage made of steel will separate and secure the canisters.

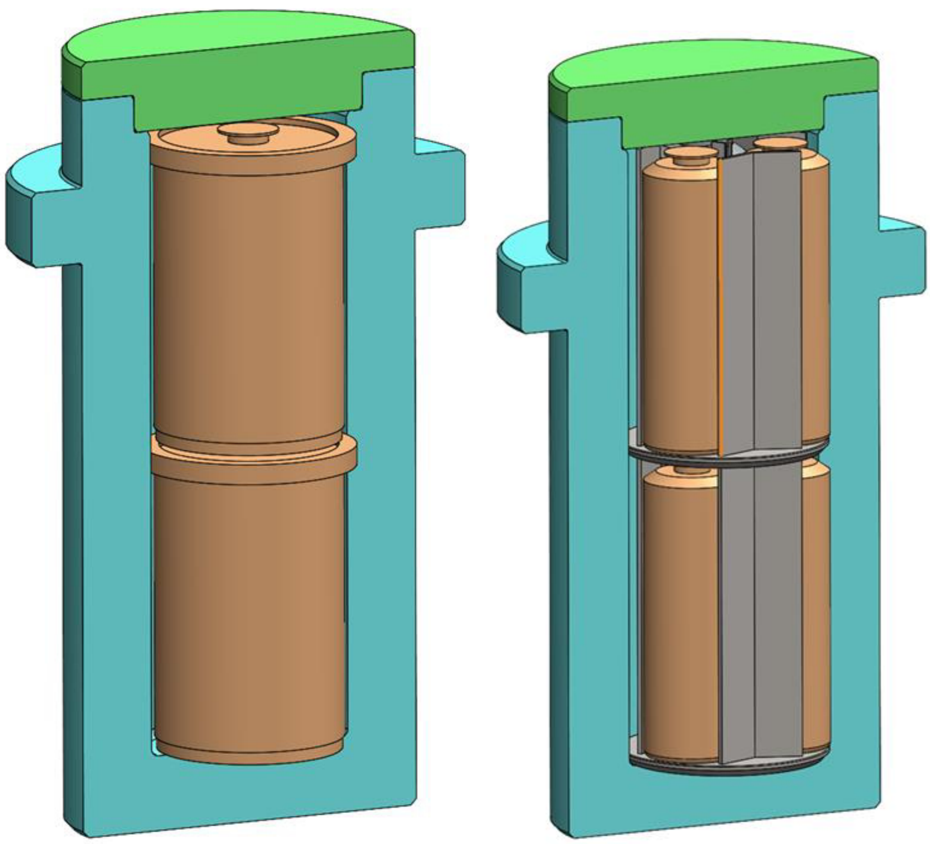


*Figure 5-1: Shielded waste package for two ECN canisters on the left and waste package for six CSD-V on the right (not to scale)*

The wall thickness of the waste package will be defined in an iterative process, taking into account mechanical aspects, corrosion rates, as well as shielding against radiation. As initial assumption, the wall thickness is set to 300 mm. It is assumed that this wall thickness is sufficient for the necessary mechanical strength, the shielding and lifetime in corrosive environment. The necessary wall thickness will be further determined in section 6, where the mechanical analyses, shielding calculations, and corrosion predictions are made.

The next design step is the definition of the attachment points for waste package handling. In the currently existing international waste package concepts, different types of attachment points are used. Either trunnions on the waste package top or sides, lifting grooves at the waste package side, or undercuts as part of the lids are used. In the ENCON waste package concept, a carrying ring for handling is used, see Figure 5-2. All these handling concepts are usually designed only for emplacement. The use during retrieval after having been exposed to the conditions of disposal and possible corrosion and degradation is not considered. At the moment it is unclear, if attachment points like bolted or screwed trunnions are reliable in the long term for retrieval operations. Comparable considerations were already described by Herold et al. (2018) at that time. At this time, it was concluded that the feasibility of demonstrating the operability of trunnions is not given. Therefore, in the new waste package design, the attachment point for handling the waste package is the outer shell of the waste package itself. Through an integrated carrying ring, the waste package is form-fit during emplacement and retrieval operations when the waste package is grabbed by handling vehicles. It is planned that the waste package is grabbed by the vehicle below the carrying ring. The carrying ring then prevents that the waste package slips out of a grapple or other handling devices. The use of the outer shell instead of the waste package top also reduces the necessary space in the emplacement galleries.

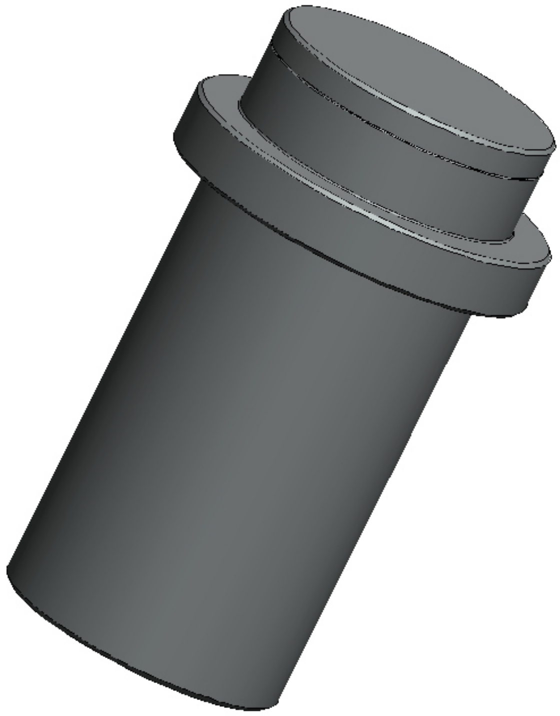


*Figure 5-2: Sketch of the waste package for two ECN canisters. See the carrying ring as handling point for both canister types.*

The waste package for two ECN canisters has a height of 3,102 mm, a diameter of 1,460 mm, and a weight of approximately 29 tons (empty). The waste package for CSD-V canisters has a height of 3,387 mm, a diameter of 1,540 mm, and a weight of approximately 35 tons. This waste package has been designed for CSD-V canisters, but it will also be possible to use it for example for CSD-C canisters because the primary waste package geometry is the same. For further details of both waste packages, see the technical drawing in the appendix of this report.

# 6 Analysis of the Waste Package

## 6.1 Corrosion

The corrosion rate depends on various factors. To estimate the corrosion rate at this early conceptual stage, various estimations and simplifications are carried out. The corrosion attack on the fine-grained structural steel TStE 355 will be reduced to the ferrous content. The steel aggregates are neglected at this point. The corrosion is assumed to be anaerobic and uniform on the surface of the canister.

The maximum temperature on the canister surface is limited to 100 °C. The surface is exposed to the brine and will corrode first. The brine is a saturated aqueous solution containing $MgCl_2$ or alternative NaCl salt.

The corrosion rate can be estimated graphically using the values given by (Kusten et al., 2004). The values from table 4-5 in Kusten et al. (2004) are transferred into Figure 6-1 as diagram with all values plotted. The diagram shows that an increase in temperature accelerates the corrosion rates. Furthermore, temperature-dependent differences between different brines are observed. For a temperature of up to 100 °C, Q-brine represents the most aggressive/corrosive environment. Corrosion rates up to 70°µm per year are documented. Similar results can be found in Kienzler (2017).

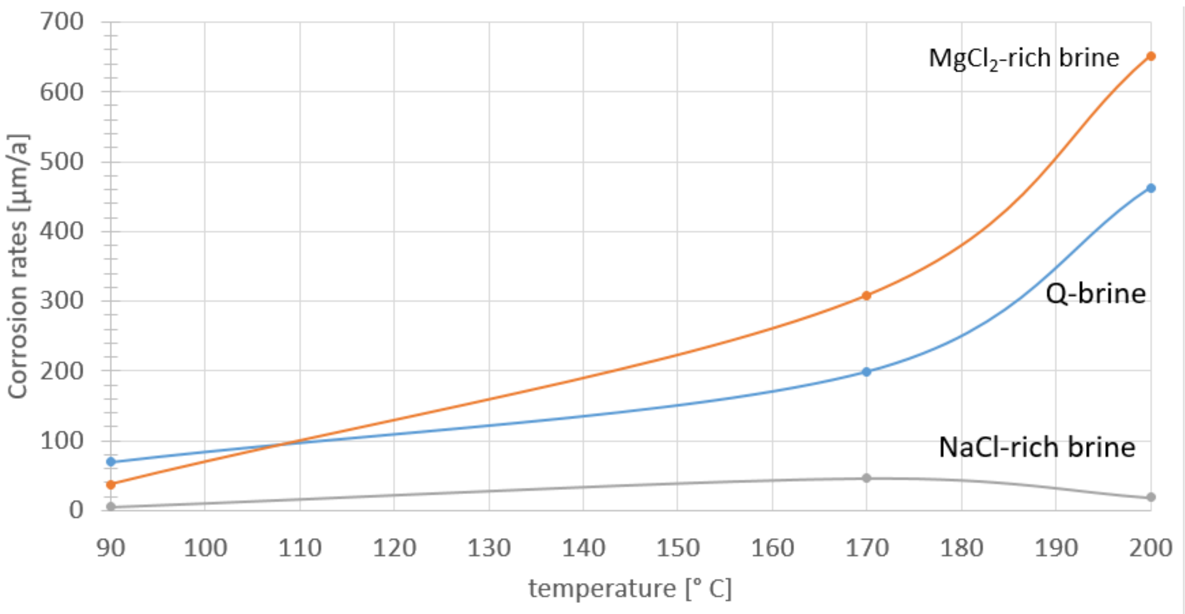



*Figure 6-1: Corrosion rates of fine-grained steel TStE 355 in $MgCl_2$- und NaCl-rich salt brines, based on Kusten et al. (2004)*

If an unlimited availability of brine and constant temperature conditions (approximately 100 °C) are assumed, a constant corrosion rate of 70 µm per year can be estimated for further calculations. For a period of 1,000 years, this results in a uniform loss of 70 mm. Pitting and intergranular corrosion could not be observed for the steel TStE 355 in the presence of a Q-Brine (No relevant influence of gamma radiation on corrosion process due to low dose rate).

For a maximum wall thickness of 300 mm, corrosion of the steel would require more than 4,000 years. For a wall thickness of 220 mm, corrosion of the steel would require more than 3,000 years. Starting from the minimum containment period of 1,000 years, a theoretical thickness of 70 mm would be sufficient to resist the corrosion for this timeframe.

Areas of welds and of mechanical loads due to rock pressure must be investigated separately with regard to corrosion estimations. Here, deviating values for the local corrosion rates may occur.

## 6.2 Gas Production

The gas production is driven by the corrosion of the ferrous content of the steel. The corrosion of iron (Fe) in aqueous environment produces magnetite ($Fe_3O_4$) and hydrogen.

$$3\,Fe + 4\,H_2O \;\rightarrow Fe_3O_4 + \;4\,H_2 \uparrow \tag{5}$$

The maximum value of molecular hydrogen formed is 0.535 $m^3$ (normal cubic meter) of $H_2$ per kg of Fe when the iron is completely converted to magnetite $Fe_3O_4$ (GRS 2008). Here, the normal cubic metre of a gas is specified for a standard temperature of 0 °C, 273.15 K and a standard pressure of 1,013.25 hPa.

Two waste package designs will be considered: Configuration loaded with CSD-V and configuration with ECN canister. The circumferential carrying ring is neglected in both waste package configurations. The volume of the configuration loaded with CSD-V is estimated $V_{loss,CSD\text{-}V} = 4.4\ m^3$ and the volume of the ECN-canister configuration is estimated $V_{loss,ECN} = 3.77\ m^3$. With the given specific density of steel $\rho_{TStE355} = 7{,}830\ kg/m^3$.

$$V_{H2} = \rho_{TStE355} \cdot V_{loss} \cdot 0{,}535 \frac{m^3}{kg} \tag{6}$$

Using the formula above, this will result in a generated Gas Volume $V_{H2\text{-}CSD\text{-}V} = 18{,}438\ m^3$ for the CSD-V-compatible canister waste package and $V_{ECN} = 15{,}756\ m^3$ for an ECN-compatible waste package.

## 6.3 Shielding

The shielding analysis was conducted by GNS Gesellschaft für Nuklear-Service mbH on behalf of BGE TECHNOLOGY GmbH. The following contains the summarised results of the analysis.

### 6.3.1 Boundary Conditions

The canister concept (Figure 5-1) includes a self-containing vessel made of fine-grained structural steel TStE355 with a wall thickness of 300 mm on all sides. The thickness of the steel lid is also 300 mm. For handling purposes, the canister also has an integrated lifting ring.

There are 2 different configurations, depending on the loading, for the waste package layout. One configuration carries six CSD-V canisters and the other configuration is loaded with two ECN canisters.

The activity of the inventory is given for the disposal point of time in the year 2130. The magnitude of the gamma and neutron sources is derived from this date. The inventory data is taken from the OPERA Safety Case (Verhoef et al., 2017).

### 6.3.2 Inventory

The calculation considers the long decay time of the inventory in the Dutch disposal concept. Hence, the dose output is substantially determined by the emitting of neutrons. To determine the source of neutrons, it is important to analyse which material acts as moderator. For CSD-V, the magnitude of neutrons is driven by the reactions with the borosilicate glass. Whereas for the ECN canisters, the driving component of neutron output is the reaction with the $UAl_x$-matrix. In both cases, more neutrons are emitted than in pure $UO_2$-fuel.

After this long decay time, the only relevant nuclide left as driver for the gamma radiation emitting inventory is Cs-137. For the configuration with six CSD-V, an activity of $1.98 \cdot 10^{15}$ Bq is calculated. The ECN configuration has an activity of $1.22 \cdot 10^{14}$ Bq.

The most important neutron emitters are the nuclides Pu-238, Am-241, Cm-244 and Cm-246 in which only Cm-244 and Cm-246 show the presence of spontaneous fission neutrons.

Origin-S is used for both configurations to calculate the magnitude of neutron output. The CSD-V configuration emits $1.22 \cdot 10^{9}$ neutrons/s, whereas the ECN configuration emits with $2.57 \cdot 10^{7}$ neutrons/s approximately 50 times fewer neutrons. Hence, only the CSD-V version is analysed in the following steps because it covers both cases with the dose output.

### 6.3.3 Calculation

The calculation is performed with MCNP©. The following dimensions are used in the simulation:

- Wall thickness (wall, lid, and bottom) of 300 mm
- Inner diameter of 940 mm
- Inner height of 2,787 mm
- Inner basket is only calculated as two 38 mm thick steel plate in two planes

Figure 6-2 illustrates the used model.

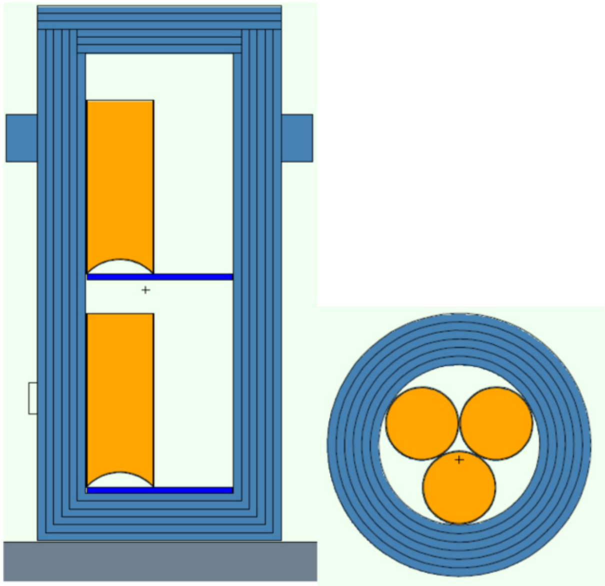


*Figure 6-2: MCNP© Calculation model of the disposal canister in six CSD-V configuration*

The CSD-V are modelled as stainless steel cylinders without top cover system. A spherical calotte on the bottom is modelled. The most important dimensions are:

- Glass diameter of 420 mm
- Glass fill height of 1,100 mm
- Steel mantle thickness of 5 mm

The materials used are summarised in Table 6-1.

*Table 6-1: Composition of the materials used in the simulation*

| Material | composition in mass-% | | | | Density, $g/cm^3$ |
|---|---|---|---|---|---|
| Steel | Fe 100 | | | | 7.8 |
| Steel TStE355 | Cr 0.3<br>C 0.18 | Mn 1.3<br>Si 0.3 | Ni 0.3 | Fe 97.62 | 7.83 |
| Glass | Li 0.239<br>Si 22.96 | B 5.59<br>Zr 11.2 | O 47.75<br>Mo 1.533 | Na 6.603<br>GD 2.51 | 2.7 |
| Air | N 75.55 | O 23.21 | Ar 1.24 | | $1.2*10^{-3}$ |

### 6.3.4 Results

The analysis shows the peak radiation on the surface of the canister below the lifting ring on a plane with the CSD-V. The results of the dose rates are shown in Table 6-2.

*Table 6-2: Results of the shielding calculations with 300 mm wall thickness*

| | |
|---|---|
| Primary gamma dose rate: | 0.004 mSv/h |
| Neutron dose rate including secondary gamma dose rate: | 4.3 mSv/h |
| Combined dose rate: | 4.3 mSv/h |

The Neutron dose rate is the determining component of the overall dose rate. The calculated dose rate is below the threshold of 10 mSv/h as set in the requirements. Hence, a reduction in wall thickness to cut manufacturing costs is possible. Simulations with less wall thickness are conducted, and the combined dose rate for a wall thickness of 220 mm is 6.76 mSv/h. It is expected that the threshold of 10 mSv/h is reached for a wall thickness of approximately 170 mm.

This shows that even the thinner wall thickness out of the mechanical design concepts fulfils the dose rate maximum limits given by the requirements. Even with a change in the composition of the glass to the maximum allowable B- and Li-values, the dose rate will not exceed 9 mSv/h. So, it is shown that the waste package concept meets the shielding requirements set by COVRA.

## 6.4 Mechanical Analysis

### 6.4.1 Introduction

The scope of this section is to verify the mechanical integrity of spent fuel canisters. The plan is to deploy the canisters in a short borehole starting from a gallery. The borehole is straight, half the length of the canister, and unprotected. During performance of this assessment, the final borehole design is not completed. Thus, as maximum permissible load for the weight load of the rock above is assumed.

For this calculation, a plastic strain hardening of areas with stress concentration is assumed. A notch factor that describes the stress relief due to plastic deformation in these areas. With this notch factor, the van Mises stresses obtained from a finite element analysis (FEA) and the material properties, the degree of utilisation of the material can be determined. The degree of utilisation describes how much of the overall material strength is used to handle the applied load. A degree of utilisation below 1 indicates that the stresses induced by the load do not exceed the material properties. In case the degree of utilisation is above 1, it is more likely that the specimen will fail when loaded with the given load. The safety factors for the uncertainties of the model are included in the degree of utilisation.

### 6.4.2 Boundary conditions

#### 6.4.2.1 Material

A non-corrosion-resistant steel is considered. This fine-grained structural steel has mediocre mechanical properties compared with other steels. The benefits are in the uniform corrosion attack without the risk of pitting. Product quality is considered as good. Table 6-3 shows the material properties used for calculation.

*Table 6-3: Material properties according to DIN EN 10025-3*

| | ISO | UNS | $R_{p0.2\%}$ | UTS | A |
|---|---|---|---|---|---|
| fine-grained steel | 1.0566 | K02010 | 275 MPa | 450 MPa | 21% |

For the strength assessment, an ideal plastic material model is considered. The elongation is considered as infinitive as the load exceeds $R_p$.

#### 6.4.2.2 Loads

The disposal galleries are backfilled with crushed rock salt. The density of the host rock is assumed to be 2,200 kg/m³ and also covers the average density of the overburden, see Bollingerfehr et al. (2018) and Bertrams et al. (2021). With this average density, the ambient pressure at a depth of 850 m is approximately 20 MPa. As described in section 2.1, the backfill compaction will be finished after 1,000 years at the latest. It has to be assumed that the full rock pressure will lay on the canister. Hence, the ambient pressure is distributed evenly over the specimen. Furthermore, a margin is added to compensate the stresses that are induced by the heat expansion of the rock salt. The safety margin for uncertainties is considered with 10 MPa.

A further increase in the load due to long-term evolution is not considered. Load increase e.g. by glaciers during a future ice age will affect the repository after several ten thousands of years, which is beyond the expected canister lifetime.

### 6.4.3 Method of Calculation

The canister is a pressure vessel loaded with outside pressure. An assessment for buckling safety for such a case is usually performed following Eurocode 3 (DIN EN 1993). However, the dimensions of the canister that is considered as pressure vessel, loaded with ambient pressure, is outside the scope of Eurocode 3. Hence, Eurocode 3 is not applicable for this scenario.

The AD2000-B6 (Beuth 2020) contains a calculation method for cylinders loaded with pressure on the outside. For this guideline, the assumed wall thickness of the canister is out of scope.

Therefore, the FKM Guideline (FKM 2020) for machinery components is considered. This guideline contains a part for static loads for a wide range of materials. The calculation has to be conducted by using the FEA method. The programme that is used for this calculation is Z88.

### 6.4.4 Geometry

The minimum inner diameter is 860 mm. The initial wall thickness was assumed to be 300 mm. For a more detailed mechanical analysis, a wall thickness of 170 mm, resulting in an outer diameter of 1,200 mm, is considered. This thickness corresponds to the minimum shielding thickness, see section 6.3. An isometric view of the simplified complete waste package is persented in *Figure 6-4*. *Figure 6-3* illustrates a quarter-cut of the waste package.

The length of the usable enclosure area is considered to be 2,502 mm. The radii in the corners are considered large enough to have an even stress distribution over the inner surface.

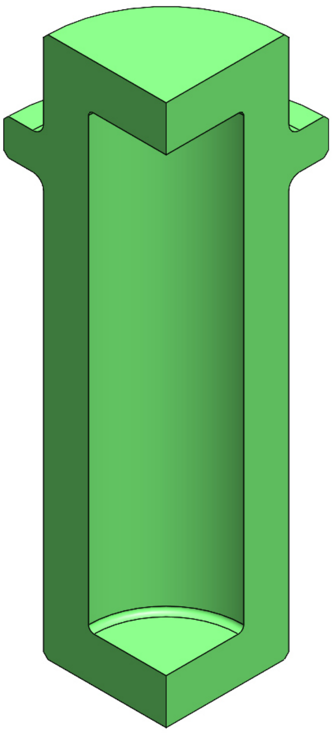


*Figure 6-3: Quarter-cut of the waste package with encapsulation area*

*Figure 6-4: Isometric view of the simplified complete waste package*

### 6.4.5 Assessment according to FKM Guideline

#### 6.4.5.1 Generating a Model for Simulation

To reduce the number of elements for this large sized model, the model is divided into a quarter model and cut in half. The geometry of the bottom and top plate is identical on the inside.

The quarter model can be split in half to get boundary conditions that are more robust for the simulation process. The weld on the top is neglected in this simulation. At this design stage, the type of the weld is not decided yet. A detailed simulation should be conducted later.

#### 6.4.5.2 Generating the mesh

The mesh size is chosen in such a way that the number of elements over the cross section is sufficient to get a low stress gradient over the elements. Best practice experience shows that a mesh of four finite-elements over a pressure vessel cross section is sufficient to fulfil this requirement. A refinement of the mesh on the radii is not necessary. The radii are large enough to have a high element count to get a good stress gradient and no stress peaks. Furthermore, the area around the radii is simplified. A detailed analysis of the lid design has to be conducted later in the design process. Figure 6-5 illustrates the mesh of the quarter-cut model of the waste package.

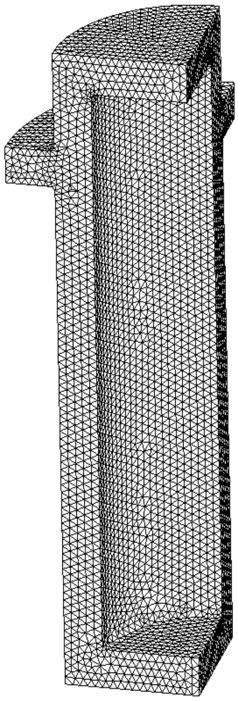


*Figure 6-5: Generated mesh for the linear and non-linear simulation*

#### 6.4.5.3 Boundary Conditions

It is necessary to supress rigid body movement during simulation. If there is no equilibrium between all forces during the simulation, the specimen will move and the simulation will abort without a result. At the same time, these boundary conditions should not affect the stress distribution of the part. Because the whole vessel is loaded with ambient pressure, it is not possible to get a safety distance to the boundary conditions. Therefore, elements have to be identified where no displacement occurs to anchor the specimen in the coordinate system.

The displacement in X, Y, and Z direction is zero at the symmetry surfaces. The surfaces are displayed in Figure 6-6. A pressure of 30 MPa is applied on the outside of the vessel.

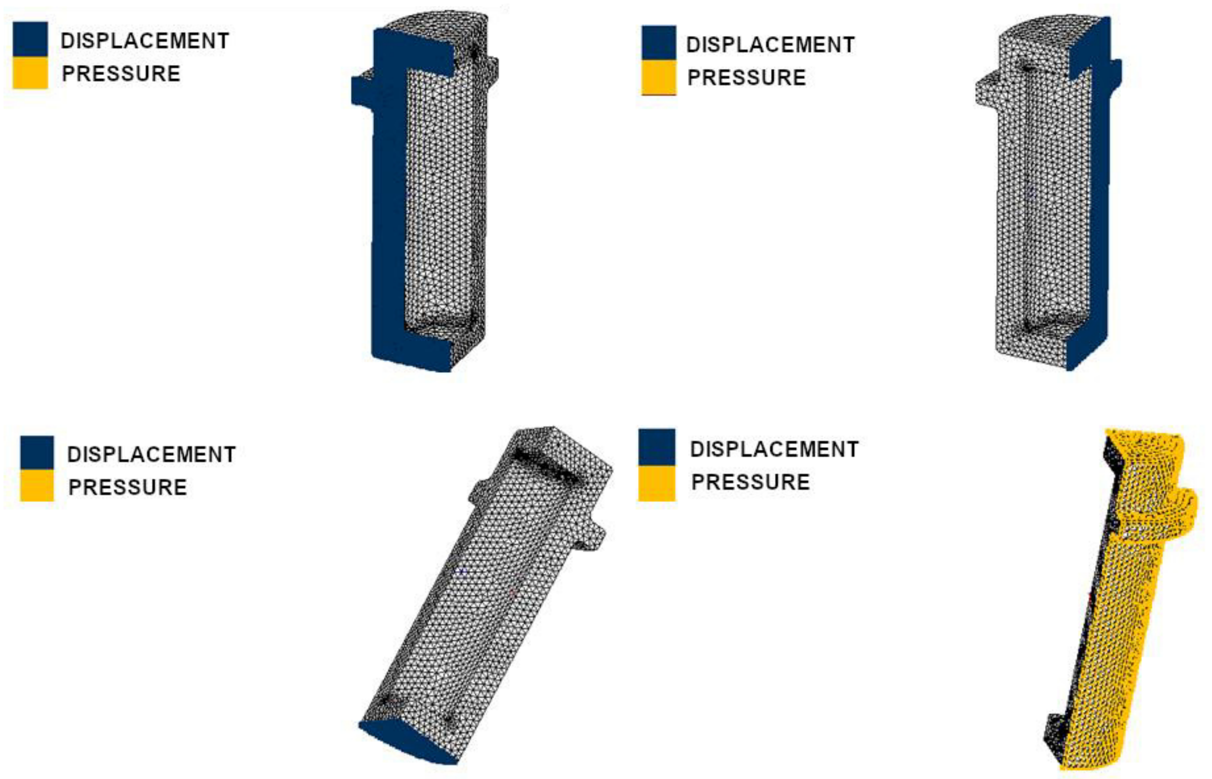



*Figure 6-6: Boundary conditions displayed on the surface of the model. Top left is the displacement in Z = 0. Top right is the displacement in X = 0. Bottom left is the displacement in Y = 0. Bottom right is pressure p = 30 MPa*

### 6.4.5.4 Calculating the Safety Factor for Fine-grained Structural Steel

**Section factor**

A plastification of certain cross sections can occur during static load. During the assessment, this plastification is considered as the section factor ($n_{pl}$). This factor is calculated by determining the plastic notch factor of the geometry. This notch factor is geometry- and material-dependend and can be determined by a non-linear finite element analysis. An analysis is started with a rising ambient pressure until the simulation stops to converge. At the load step when the load exceeds the elastic strength of the material, the solver switches to the ideal plastic material model. This point is an important parameter for the analysis. The plastification of the cross section increases with rising ambient pressure. At the load step when the ambient pressure exceeds the plastic load limit of the material and the whole cross section is affected by plastification, the plastic load limit is reached. At this point of the simulation, the solver will switch from elastic to plastic behaviour. Figure 6-7 illustrates the simulation results to determine the plastification factor.

$$K_P = \frac{plastic\ load\ limit}{elastic\ load\ limit} = \frac{90MPa}{20\ MPa} = 4.5 \qquad (7)$$

$$n_{pl} = MIN\left(\sqrt{\frac{E \cdot 0.05}{R_p}}; K_p\right) = MIN\left(\sqrt{\frac{212{,}000\ MPa \cdot 0.05}{275}}; 4.5\right) = 1.96 \qquad (8)$$

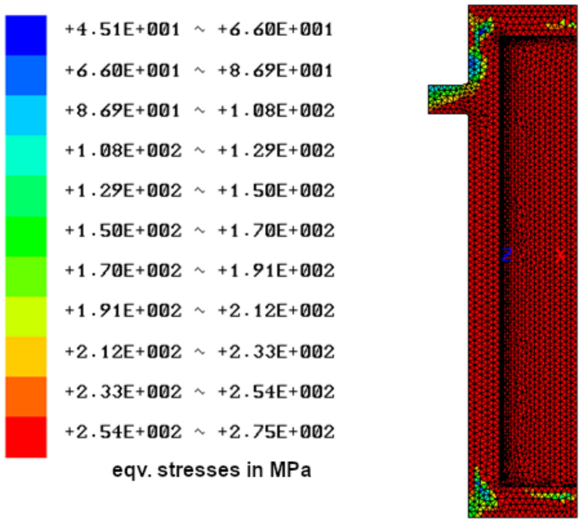



*Figure 6-7: Simulation results to determine the plastification factor. Red areas show the plastic deformation of the material with a load of 100 MPa. This is the last step, where the cross section of the specimen is not plastic deformed and the material is not torn appart.*

Therefore, the plastic component strength is used to represent the plastic strain hardening. The plastic deformation in a limited cross section will result in an increased tensile strength in this area, as the scope of this analysis only covers ductile materials.

$$\sigma_{CS} = R_p \cdot n_{pl} = 275\ MPa \cdot 1{,}96 = 539\ MPa \quad (9)$$

This is only valid for the specimen loaded with an ambient outside pressure.

**Safety factors**

The safety factor depends on the quotient of ultimate tensile strength (Table 6-3) and yield strength.

$$for\ \frac{R_p}{UTS} \leq 0.75\ ;\ j_{total} = \frac{j_{UTS} \cdot j_E \cdot R_p}{K_{T,UTS} \cdot UTS} \quad (10)$$

$$for\ \frac{R_p}{UTS} > 0.7\ ;\ j_{total} = \frac{j_{YS} \cdot j_E}{K_{T,Rp}} \quad (11)$$

$$\frac{R_p}{UTS} = \frac{275\ MPa}{450\ MPa} = 0.611 \quad (12)$$

The material properties are listed in Table 6-3. The material safety factor is taken from the FKM Guideline (FKM 2020). The consequences of failure of the canister is considered as “severe”. To provide a conservative safety factor the probability of occurrence is considered as “high”. The magnitude of the ambient pressure is well known and can be calculated very accurately.

According to the FKM Guideline (FKM 2020) $j_{UTS} = 2$ and $j_{YS} = 1.5$. The temperature safety factor is calculated:

$$K_{T,UTS} = 1 - 1.7 \cdot 10^{-3}(T - 100) \tag{13}$$

In this case, the temperature safety factor is 1 because the expected temperatures are below 100 °C. The environmental safety factor $j_e$ considers the occurrence of a corrosive fluid surrounding the part. In this case, the corrosive environment is present but the corrosion resistance of the material is considered as medium. The safety factor $j_E = 1.2$ is proposed by the FKM guideline. As the corrosion attack is already subtracted from the model, this can be considered as additional safety factor. It should be kept in mind that this is an early rough calculation and high safety factors can be reduced in later iterations, if confidence builds up.

With the given total safety factor of 1.47 (12), the static degree of utilisation can be determined (13).

$$j_{total} = \frac{j_{UTS} \cdot j_E \cdot R_p}{K_{T,UTS} \cdot UTS} = \frac{2 \cdot 1.2 \cdot 275}{1 \cdot 450} = 1.47 \tag{14}$$

$$a = \frac{\sigma_v \cdot j_{total}}{\sigma_{cs}} \tag{15}$$

**Degree of utilisation for ambient pressure**

The maximum equivalent stress on the object is $\sigma_v = 108\ MPa$. This stress is obtained from the simulation result (Figure 6-8).

This results a in a degree of utilisation of:

$$a = \frac{108\ MPa \cdot 1{,}47}{539\ MPa} = 0.29 \tag{16}$$

Under the given boundary conditions, this pressure vessel is considered as safe against failure.

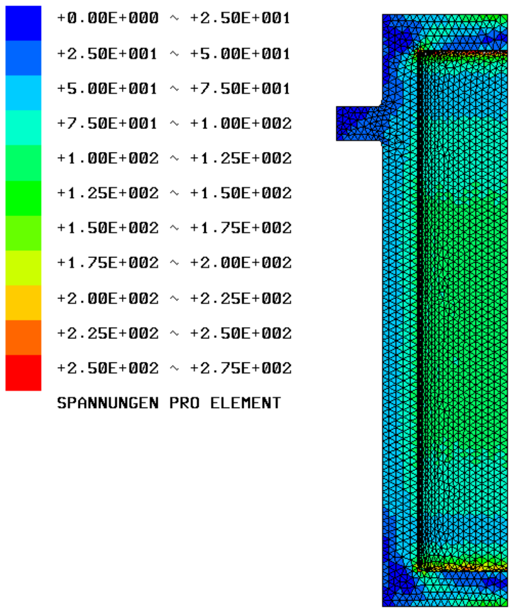



*Figure 6-8: Result of linear elastic FEA with 30 MPa ambient pressure showing von Mises stresses*

#### 6.4.5.5 Verifying the Calculation Results

**Ideal plastic simulation with design load**

In a first iteration, an ideal plastic calculation can be performed to determine the degree of utilisation. This ideal plastic calculation is linked to a linear elastic approach. In this calculation, the load is applied in several calculation steps and released afterwards. As a result, the simulation shows the residual stresses that occur after plastic deformation of the calculation specimen, see Figure 6-9.

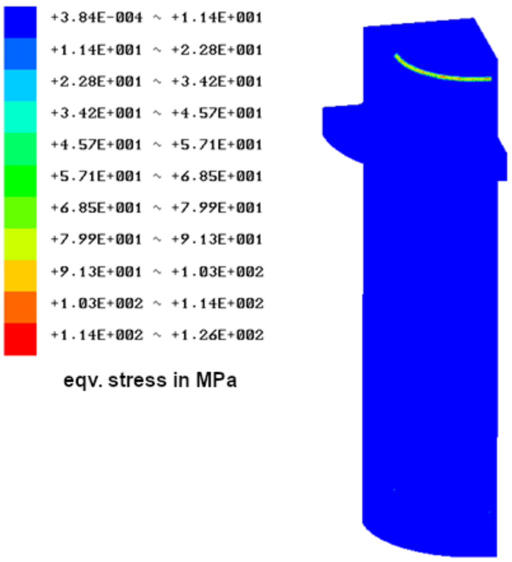



*Figure 6-9: Residual stresses after releasing the ambient pressure*

The residual stresses occur only in the radii between the lid and the centre body. This points out that the only area affected by plastic strain is the simplified region at the top and bottom. This region is considered as safe against static load. A further optimisation is possible in the following design process to remove the plastic strain completely.

**Approach to calculate loads without FEA**

The equivalent stresses in the cylindrical section of the canister may be calculated via Barlow's formula. The tangential, axial, and radial stresses need to be calculated.

Axial stress:

$$\sigma_a = \frac{(p_i \cdot {r_i}^2 - p_a \cdot {r_a}^2)}{{r_a}^2 - {r_i}^2} \tag{17}$$

Radial stress:

$$\sigma_r = -p_a \cdot \frac{{r_a}^2}{{r_a}^2 - {r_i}^2} \cdot \left(1 - \frac{r_i^2}{r_x^2}\right) \tag{18}$$

Tangential stress:

$$\sigma_t = -p_a \cdot \frac{{r_a}^2}{{r_a}^2 - {r_i}^2} \cdot \left(1 + \frac{r_i^2}{r_x^2}\right) \tag{19}$$

Equivalent stress:

$$\sigma_v = \sqrt{(\sigma_t^2 + \sigma_r^2 + \sigma_a^2 - \sigma_t \cdot \sigma_r - \sigma_t \cdot \sigma_a - \sigma_r \cdot \sigma_a)} \tag{20}$$

$$p_i = inner\ pressure$$
$$p_a = outer\ pressure$$
$$r_a = outer\ radius$$
$$r_i = inner\ radius$$
$$r_x = radius\ for\ calculation$$

The calculation for this load case is done as followed:

The inner diameter of the specimen is 860 mm, and the outer diameter after corrosion attack is 1,200 mm.

Axial stress:

$$\sigma_a = \frac{(-30\ MPa \cdot (600\ mm)^2)}{(600\ mm)^2 - (430\ mm)^2} = -61{,}7\ MPa \tag{21}$$

Radial stress:

$$\sigma_r = -30\ MPa \cdot \frac{(600\ mm)^2}{(600\ mm)^2 - (430\ mm)^2} \cdot \left(1 - \frac{(430\ mm)^2}{r_x^2}\right) \tag{22}$$

Tangential stress:

$$\sigma_t = -30\ MPa \cdot \frac{(600\ mm)^2}{(600\ mm)^2 - (430\ mm)^2} \cdot \left(1 + \frac{(430\ mm)^2}{r_x^2}\right) \tag{23}$$

The stress depends on the analysed position within the wall. The calculation is performed with Excel, and a graph is plotted. As shown in Figure 6-10, the calculation with Barlow's formula is on par with the results from the finite elements analysis.

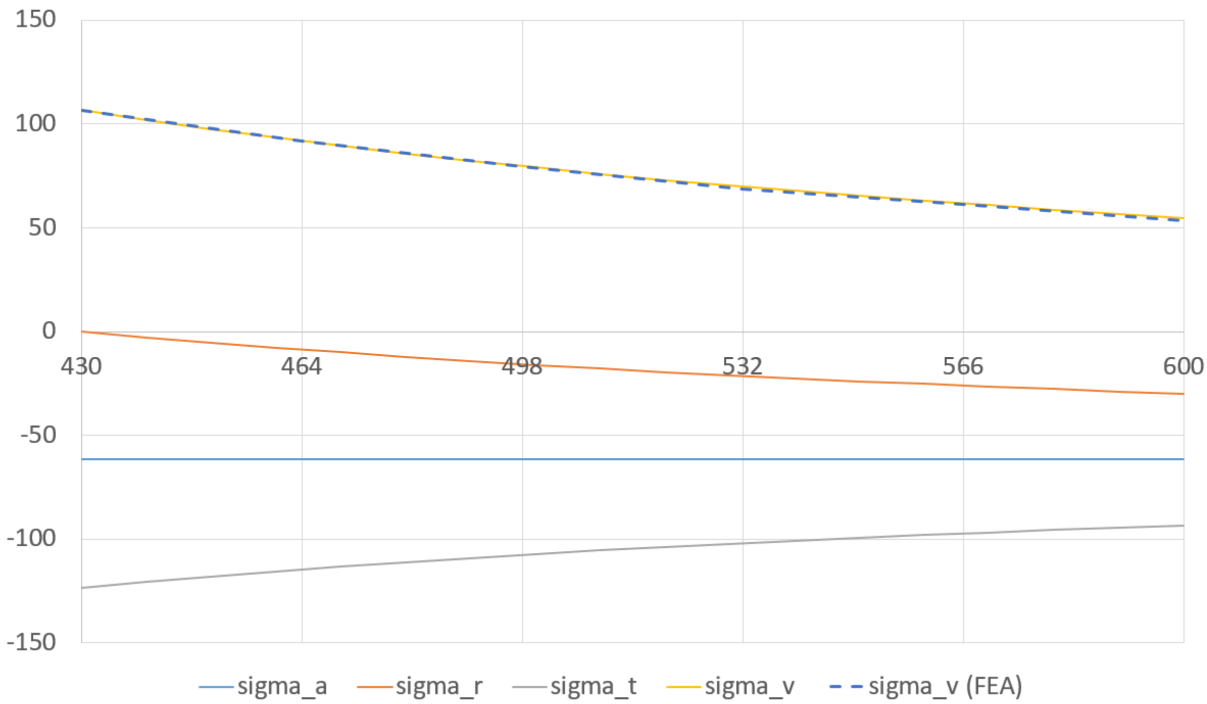



*Figure 6-10: Comparison of the equivalent stresses on the cylindrical section calculated with Barlow's formula and calculated by FEA*

For a rough first calculation, this formula can be used to determine the equivalent stress on the inner section of the cylindrical part. If there are no changes in material composition and properties formula (16) can be used to determine the degree of utilisation. If there is any change in material or a significant change in the inner to outer diameter ratio, it is necessary to redo the safety factor calculation. The plastic section factor $n_{pl}$ can be set to one, resulting in, $\sigma_{CS} = R_p$, if no FEA analysis is available for the specimen. This will reduce the possible load by a large percentage.

The lids at the top and bottom need to be calculated at the very end of the design process. As a first assumption, the wall thicknesses of the top and bottom should be the same as the rest of the cylinder wall. A detailed calculation should be conducted at the end of the design phase.

### 6.4.6 Handling Loads

Another important load case is the handling of the canisters. The shielding cask weighs about 30 tons, and the weight of the loaded canister with nuclear waste is approximately 35 tons. Hence, the load is 343 kN. The model is simplified to a quarter model. So, the load at the latch is 82.5 kN. The surface load is applied to the bottom of the canister inside. At the position where the gripper is located a displacement of zero is applied.

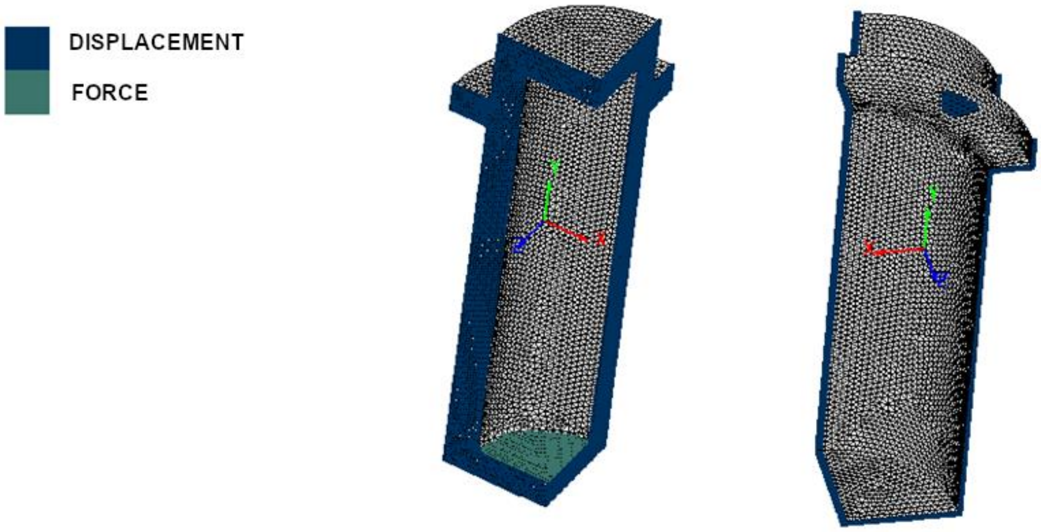



*Figure 6-11: Boundary conditions shown on the outside and inside of the simplified quarter model*

The equivalent stresses are shown in Figure 6-12. With approximately 14 MPa, the magnitude of the peak stress is far from the yield strength of the material. In this case, a simplified approach is chosen to determine the degree of utilisation. The plastic strain hardening is considered as $n_{pl} = 1$, thus $\sigma_{CS} = R_p$.

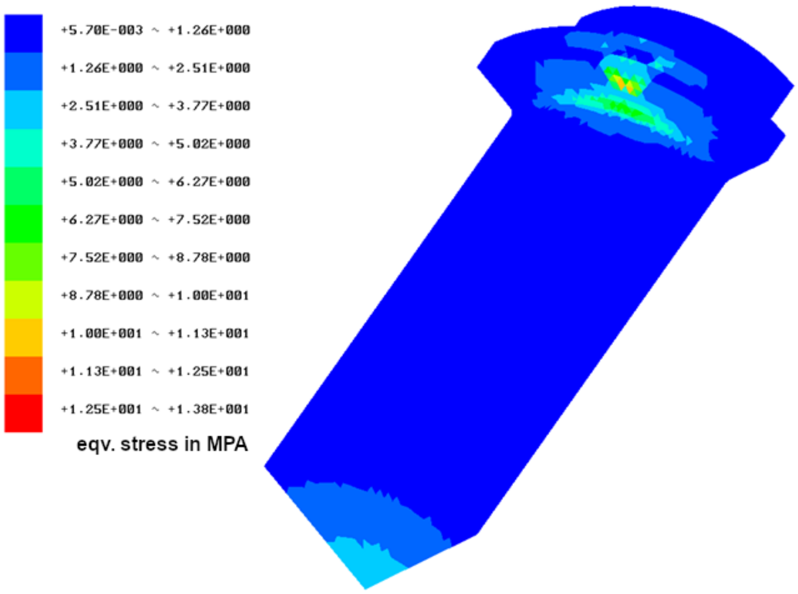



*Figure 6-12: Equivalent stress with handling load with a simulated gripper with small contact area at the outside of the flange*

The degree of utilisation is calculated with the simplified formula:

$$a = \frac{14\ MPa * 1{,}47}{275} = 0{,}075 \tag{24}$$

Under the given boundary conditions, the canister is considered as safe against failure when exposed to handling loads.

## 6.5 Discussion

The more detailed analysis of shielding, corrosion, and mechanical behaviour provides statements about the degree of utilisation or minimum required values of the canister wall thickness. To comply with the long-term safety requirements, the wall thickness must meet the requirements derived from analysis of each of these aspects.

The initial design includes a wall thickness of 300 mm, which is connected to an oversized mechanical stability, high shielding, and long corrosion resistance. From a mechanical point of view, the thickness could be reduced to approximately 120 mm. However, the shielding limit of 10 mSv/h is not given for such thin walls. With respect to the shielding, a minimum wall thickness of approximately 170 mm is required.

Long-term stability is determined by the corrosion rate. A corrosion rate of 70 µm per year is defined as worst case conditions (constant high temperature and unlimited availability of brine). Containment and thus integrity of the canister have to be guaranteed at least as long as the backfill compaction is not yet finished. It is assumed that backfill compaction runs for a period of approximately 1,000 years. During this period, mechanical stability and containment are required. Following the numbers given before, 120 mm are needed for mechanical stability, and 70 mm for corrosion. As a result, the minimum wall thickness required is 190 mm. Taking into account an additional safety margin, a wall thickness of 220 mm, as discussed in the mechanical analysis and shielding analysis, is proposed as final wall thickness.

# 7 Summary and Conclusions

For the Dutch repository concept in a domal salt formation, a suitable DWP has to be developed. The work followed the idea of designing a simple but robust waste package that – in combination with the other barriers – can provide isolation as well as containment over the required time. The canister itself must not provide all the safety functions during the full long-term phase. The six major safety functions of the waste package are containment, radiation shielding, sub-criticality, limiting temperature, limiting corrosion, and operability. Also, it should be possible to retrieve the waste package.

For the waste package design, existing waste packages of other nuclear waste management organisations were described first. None of these waste packages could meet all the requirements set by COVRA, so it was decided to develop a new waste package concept. As waste package concept, a simple steel cask is proposed. This waste package shall be made by forging and will be closed by welding to ensure the necessary containment of radionuclides.

It was a requirement that the waste package be made of materials that are extensively known and that practical experience of their use is available. It is planned to use carbon steel as material for the waste package. Carbon steel is also used in German waste package concepts in rock salt or in Switzerland in clay. The behaviour of carbon steel under disposal conditions was extensively researched in Germany both in the laboratory and in in-situ experiments. Therefore, carbon steel can be estimated as predictable material. Carbon steel is also widely used in practically all fields of industry.

It was also required by COVRA that only present day construction methods be used for the manufacturing of the waste package. The waste package hull itself is planned to be made by either casting or forging. For the closure of the package, either electron beam welding or submerged arc welding can be used. All the manufacturing methods are known and have a widespread use in different fields of industry.

The waste package should provide containment for at least 1,000 years under disposal conditions. Therefore, the waste package must have a predictable degradation behaviour. The most important degradation mechanism for carbon steel under disposal conditions is corrosion. For carbon steel, corrosion rates under disposal conditions can be found in literature. A corrosion assessment showed that the waste package can ensure a lifetime of about 1,000 years and therefore meets the requirements for containment and retrieval. With the initially chosen wall thickness of 300 mm and the assumed corrosion rate of 70 µm per year, the waste package has a lifetime of more than 4,000 years.

The maximum dose rate on the waste package surface should be below 10 mSv/h. With a wall thickness of around 300 mm, it is expected that the waste package will fulfil the shielding requirement. Detailed analyses of the shielding capabilities showed that the surface dose rate of the waste package lies below the threshold. Hence, even with a reduction of the wall thickness by 80°mm the waste package would still comply with the shielding requirement. 220 mm is proposed as optimised wall thickness.

The waste package will reach a mass of around 35 tons. In disposal mock-ups in Germany, dummy waste packages with masses of up to 65 tons have already been handled and transported in a safe manner. Therefore, it can be expected, that a waste package with a mass of around 35 tons can also be handled and transported safely.

The maximum temperature of the waste package should not exceed 100 °C at its surface. With the planned load of the waste package and the expected heat-generation, just a limited heat output of maximal 1.2 kW is expected. Exceeding the design temperature is not expected under these conditions.

Also, the gas formation during degradation of the waste package should be minimised. With the use of carbon steel, an actively corroding material under disposal conditions, there will be some gas formation in the form of hydrogen. In a dry scenario in rock salt, only slight corrosion generally occurs. To prevent additional gas formation due to degradation of organic material, no organic material like polyethylene or graphite is used in the waste package. A review of the gas generation of the waste package was done in the project. A simple formula was given to predict the gas production should the waste package corrode completely.

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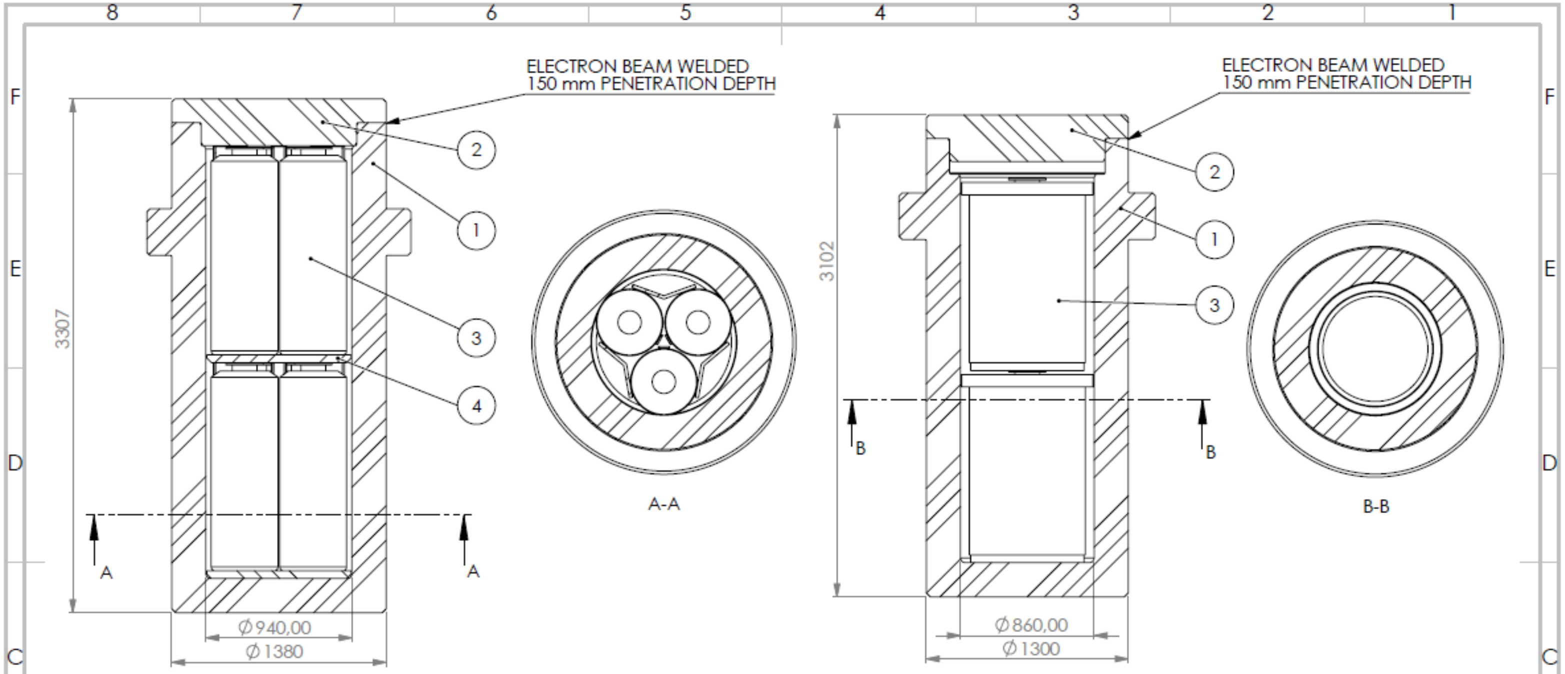
## Appendix 1 – Technical drawing with 220 mm wall thickness for both types of DWP




CONFIGURATION LOADED WITH CSD-V

| POS-NR. | PART NUMBER | DESCRIPTION | QTY. | WEIGHT [KG] |
|---|---|---|---|---|
| 1 | TEC00004 | CANISTER | 1 | 21005.10 |
| 2 | TEC00005 | CANISTER LID | 1 | 2482.27 |
| 3 | TEC00002 | CSD-V CASK | 6 | max. 850 |
| 4 | TEC00007 | CAGE | 2 | 508.44 |

CONFIGURATION LOADED WITH ECN CANISTER

| POS-NR. | BENENNUNG | DESCRIPTION | QTY. | WEIGHT [KG] |
|---|---|---|---|---|
| 1 | TEC00004 | CANISTER | 1 | 18289.91 |
| 2 | TEC00005 | CANISTER LID | 1 | 2297.09 |
| 3 | TEC00003 | ECN CANISTER | 2 | approx. 1000 |

all dimensions in mm

| Oberfläche / Surface Finish: | Allgemeintoleranz / General Tolerances: | Werkstoff / Material: |
|---|---|---|
| NONE | DIN ISO 2768 - mK | SEE SEPERATE PART LIST |

BGE TEC
BGE Technology GmbH
Eschenstraße 55
31224 Peine

Erstellt von / Created by: DS 31.01.2023
Geprüft von / Checked by: PH 31.01.2023
Gewicht / Weight: kg
Beschreibung / Description: CANISTER ASSEMBLY
Technische Zeichnung / Technical Drawing: TEC000006
Maßstab / Scale: 1:25
Format: A3
Revision: B

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