5 Shielding Evaluation

5.0 Overview

	Name, Function	Date	Signature
[redacted]			
[redacted]			

In this chapter, the shielding analysis of the CASTOR® geo69 dual purpose cask (transport and storage cask) is presented in its storage configuration. The CASTOR® geo69 storage cask as a part of DSS, in the following text also as (simply) storage cask identified, is designed to accommodate 69 spent nuclear fuel assemblies (SNF) from boiling water reactors (BWR). There are six types of SNF described in the contents description (see section 1.2.3) authorised for storage in the cask.

In order to offer more flexibility in the fuel storage, three loading patterns TR1 to TR3 defined in section 1.2.3 are used to distribute the SNF into six position groups of fuel with identical characteristics (see Figure 5.0-1). These patterns are either uniform (TR1) or regionalised (TR2 and TR3) loadings, mainly identified by corresponding decay heats per SNF. It is shown in the following sections that the homogeneous pattern TR1 with its bounding source terms covers the other two patterns in a sense of external dose rate.

Storage configuration of the cask, analysed in this chapter, coincides with its transport version besides the absence of the impact limiters in storage. An optional storage frame and a protection cover being an integral part of the DSS are conservatively not modelled and are not discussed any further in this chapter.

This chapter will demonstrate that the design of the CASTOR® geo69 storage cask fulfils the following acceptance criteria outlined in the Standard Review Plan NUREG-1536 [1] and NUREG-2215 [2]:

- The radiation shielding features of the proposed dry storage system must be sufficient for it to meet the radiation dose requirements in 10 CFR 72.104. This is demonstrated by providing a shielding analysis of the surrounding dose rates that contribute to off-site doses at appropriate distances for a cask (typical array of casks in the most bounding site configuration) with bounding source terms for normal conditions of storage and anticipated occurrences (off-normal conditions).
- Dry storage system contents and design features important for shielding are adequately described for evaluating shielding effects and dose rates.
- Radiation shielding features must be sufficient for the design to meet the requirements in 10 CFR 72.106. This is demonstrated by calculating dose rates and doses at appropriate distances for relevant accident conditions for appropriate configurations of the cask and assumptions regarding accidents.
- Dose rates from the cask must be consistent with a well-established "as low as reasonably possible" (ALARA) programme for activities in and around the storage site.
- The proposed shielding features should enable a general licensee that uses the dry storage system to meet the regulatory requirements prescribed in 10 CFR Part 20.

- Appropriate distances for the foregoing criteria are distances that are consistent with, or bounding for, the distances to the controlled area boundaries of potential cask users. The minimum distance to the controlled area boundary is 100 meters.

[redacted]

Figure 5.0-1 Position groups in the basket

This chapter contains the following information demonstrating full compliance with the aforementioned criteria:

- A description of the shielding features of the storage cask
- A description of the bounding source term (identical to the source term defined in the corresponding SAR (transport) documentation [3])
- A description of the shielding analysis methodology as well of the shielding model and materials
- Analyses of the external dose rates in the vicinity of the storage cask important for the handling operation in view of ALARA practices. These analyses include the configuration under normal, off-normal and accident conditions of storage
- Analyses of the dose rates and doses at the controlled area boundary performed with the covering configuration of the storage cask array

For flexibility regarding configuration of the storage site, a two-fold assessment of the dose at the site boundary is done:

- A minimum distance to the site boundary is determined, where a most penalising array of the DSS complies with the radiation dose limit from 10 CFR 72.104
- A minimum concrete wall thickness of the storage building is determined so that the radiation dose limit from 10 CFR 72.104 is met at a distance of 100 m

The use of the CASTOR® geo69 cask loading unit system (CLU, see section 1.2) including the transfer cask, the transfer lock and further equipment [redacted] its contribution to the dose at the storage site boundary is neglected. A discussion of the estimated occupational exposures for the CLU is considered in chapter 11 together with the exposures of the CASTOR® geo69 DSS. A corresponding shielding model of the CASTOR® geo69 CLU is presented in section 5.5

In this chapter, the dose rates and doses from the direct neutron and gamma radiation stemming from the storage cask are calculated. The discussion of the possible release of the radioactive materials from the storage cask is presented in chapter 7.

List of References

[1] NUREG-1536, Revision 1, Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility, Office of Nuclear Material Safety and Safeguards, July 2010

[2] NUREG-2215, Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities, Office of Nuclear Material Safety and Safeguards, April 2020

[3] 1014-SR-00001, Rev. 0
Safety Analysis Report
Type B(U)F Transport Package CASTOR® geo69
Docket No.: 71-9383, 23.12.2020

5.1 Discussion and Results

	Name, Function	Date	Signature
[redacted]			
[redacted]			[redacted]

The principal sources of radiation from SNF are:

- Gamma radiation originating from a decay of actinides and radioactive fission products, of fuel and hardware activation products generated during reactor operation, as well as secondary gamma particles from neutron capture,
- Neutron radiation from spontaneous fission, from (α,n)-reactions in fuel materials, from secondary neutrons produced by fission via subcritical multiplication, and from (γ,n)-reactions. The latter source is however negligible.

The major parts of the storage cask relevant for the shielding of radiation sources are:

- the basket with 69 positions to accommodate SNF,
- the canister with its lid,
- the cask body (with its lid) incorporating moderator rods and plates.

Shielding from gamma radiation is provided by the steel structure of the canister and the lid system and by the ductile cast iron (DCI) of the cask body. In order to make neutron shielding effective, the neutrons have to be thermalised and then absorbed. For this purpose, the moderator rods and plates made of unborated ultra high-molecular weight polyethylene (UHMW PE) are incorporated into the cask body. Together with relatively high carbon contents in the DCI, they provide an effective way to thermalise neutrons. Sufficient DCI behind the polyethylene rods towards the external surface of the cask not only allows for an efficient absorption of neutrons, but greatly supresses the high energetic secondary gamma radiation.

The borated structures of the basket ([redacted]) are not primarily aimed to improve the shielding performance of the cask, but nevertheless diminish the thermal part of the neutron spectrum around the SNF to some extent. This helps to reduce the dose rate contributions (mostly) from the inner fuel assemblies.

Additional basket elements (round segments and shielding elements) made of aluminium are added to the basket not only to stabilise it and support heat dissipation, but also to provide some additional gamma shielding.

The cross-sectional top view of the storage cask (quarter cut) is presented in Figure 5.1-1, while the elevation section of the cask is displayed in Figure 5.1-2. These views are directly generated from the calculation inputs at normal conditions. The colour code corresponds to the materials used in the calculations. Different colours selected for the moderator rods are used to distinguish longer and shorter rods. In section 5.3 the shielding model of the storage cask is presented in greater detail.

[redacted]

Figure 5.1-1 Cross sectional view of the cask model (2020 mm above cask bottom)

[redacted]

Figure 5.1-2 Elevation cut through the cask model

The most important dimensions of the storage cask for the shielding calculations in its central plane are presented in Figure 5.1-3 (standard MCNP colours are desaturated). The dimensions according to the technical drawings in section 1.5 are highlighted as blue text, the implementation into the shielding model as green text. The thicknesses of the materials relevant for the shielding analysis are set to their minimum. [redacted]

[redacted]

Figure 5.1-3 Cross sectional view of the cask model with major dimension (in mm)

Axially the main shielding is provided by:

- a [redacted] cask bottom part consisting of thick DCI part ([redacted]), [redacted] bottom moderator plate, and a [redacted]) closure plate;
- the lid system consisting of the canister lid [redacted] moderator plate [redacted]), and cask lid [redacted].

No credit has been taken of the protection cover (see section 1.2).

The dose rates on the lid side of the storage cask are much smaller than on the shell side, which is decisive for the design, [redacted].

The densities of the materials are reduced relative to their nominal values as discussed in section 5.3.2. [redacted].

A unique feature of the CASTOR® geo69 are the moderator rods placed directly in the cask body. The rods are made of UHMW PE without neutron absorbing additives and serve the neutron moderation purpose only. From point of view of storage, few standard situations are to be considered:

[redacted]

Two shielding models, cold and hot, are hence analysed. One of the two delivering the highest external dose rates is considered as representative for the cask design. Figure 5.1-4 and Figure 5.1-5 illustrate these two shielding models for the inner and outer moderator rods, respectively. In both

cases, the geometry of the setup is implemented according to section 3.3, and the tolerances are chosen such that the air gaps are maximised.

[redacted]

Figure 5.1-4 Inner moderator rods under different operating conditions (dimensions in mm)

[redacted]

Figure 5.1-5 Outer moderator rods under different operating conditions (dimensions in mm)

The information about particular dose rates is gained from detectors positioned all around the cask. Besides this geometry-independent mesh of detectors ([redacted]), separate volumetric detectors are modelled in order to control the calculation process. The maximum dose rates are always searched for in different dose rate locations.

High burn-up spent fuel is intended to be stored in the DSS beyond 20 years of dry storage, therefore, the impact of the fuel failure under normal, off-normal and accident conditions of storage is evaluated according to NUREG-2224 [1]. It is assumed that under normal and off-normal conditions the DSS remains vertically oriented, thus the source occurring due to the fuel failure is relocated towards the bottom of the canister into the region [redacted] The corresponding dose rate evaluations are performed for normal and off-normal conditions with the source terms after 20 years of dry storage.

Besides the fuel failure there are no factors influencing the shielding performance of the storage cask under off-normal storage conditions. Contrary to this, under accident conditions of storage there is a certain probability to completely lose the neutron moderator as a result of fire. It is shown, however, in chapters 3 and 12 that the basket, canister, and the cask remain largely unaltered in different accident scenarios and that the complete loss of the neutron moderator is excluded. Nevertheless, shielding analysis is performed with this highly conservative assumption. A fuel failure of 100 % according to [1] with fuel relocation is also considered. For this scenario, no additional cooling down of the contents is assumed.

The code system MCNP6 in version 2.0 [2] is used to calculate the dose rates.

The bounding source terms are not calculated anew, they are taken over from the SAR (transport) [3], as the CASTOR® geo69 is supposed to transport and store the very same contents. Furthermore, the shielding configurations for the transportation and storage do not differ considerably, it is only the presence of the impact limiter for the former one.

As discussed in section 5.4, the dose rates for the uniform loading pattern (TR1) bound the dose rates for both regionalised loading patterns (TR2 and TR3), therefore, the fuel failure scenarios are only performed for the former loading pattern. For the same reason, the dose rates and doses at the far distances from the storage cask are calculated for the uniform loading pattern only.

In order to calculate the dose to public and understand, where the boundary of the restricted area should be established, a bounding array of storage casks is investigated. [redacted]

[redacted]

Figure 5.1-6 Bounding array of storage casks

With respect to the configuration flexibility of the storage facility, it is separately investigated, what the minimum distance of compliance with the requirements of 10 CFR 72.104 is, which minimum thickness of the storage building is needed to comply with the requirements of 10 CFR 72.104 already at a distance of 100 m, and whether the requirements of 10 CFR 72.106 are fulfilled.

Apparently, none of the off-normal conditions have any impact on the shielding analysis. The only significant difference is that under off-normal conditions 10 % fuel failure is assumed instead of 3 % for the storage period beyond 20 years. Other boundary conditions remain identical for the purpose of the shielding evaluation.

The 10 CFR 72.104 criteria for radioactive materials in effluents and direct radiation during normal and off-normal operations are:

- During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ.
- Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10 CFR 20 Subpart C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 11 addresses these regulations. For this chapter, a dose rate limit at the restricted area boundary of 0.02 mSv/h (2 mrem) is of particular interest.

The 10 CFR 72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

- Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to to skin or any extremity may not exceed 0.5 Sv (50 rem). The minimum distance from the spent

fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

In accordance with ALARA practices, sufficiently low dose rates have to be established in the vicinity of the storage cask, at its surface and in the area close to the cask. Table 5.1-1 presents an overview of the total dose rates (sum of the neutron radiation, direct gamma radiation, induced gamma radiation and gamma radiation from fuel hardware) around the storage cask for normal, off-normal and accident conditions. The dose rates calculated within different fuel failure scenarios discussed above are presented as well. All values presented include 2σ statistical uncertainty as described in section 5.4.

The following conclusions can be made:

- Even without impact limiters the storage cask complies with the transport requirements according to 10 CFR 71.
- [redacted]
- The dose rates from the uniform loading pattern (TR1) are higher than those from the regionalised ones at the shell side of the storage cask, which is decisive for the distant dose rate due to its magnitude.
- Fuel failure under normal or off-normal conditions (for details see section 5.4) does not lead to an increase of the dose rate. For that reason, the storage site evaluations are performed with an undamaged fuel model.
- Fuel failure under accident condition leads to a local increase of the dose rate at the storage cask surface. On the shell side, this effect vanishes with increasing distance from the cask. At a distance of 1 m from the cask, the dose rates caused by the undamaged fuel are higher. The same is true also for the fuel failure case, when the storage cask remains upright oriented. The fuel failure followed by the concentration of the rubble at the lid side of the cask can only occur in case of over-tipping, for which the shell side of the storage cask plays a major role for the external dose rate. Summarising the accident related findings, the accident without fuel damage is bounding and shall generate the highest dose rate at the site boundary.

The maximum dose rate at the cask surface under normal (and off-normal) conditions is dominated by [redacted] The total dose rate e.g. cold cask loaded according to TR1 sums up as follows: 1.47 mSv/h = [redacted], with the contributions from primary gamma radiation, neutrons including secondary gamma radiation, and hardware radiation, respectively. The details about dose rate distributions at the cask surface (and at a distance of 1 m from the surface as well) are presented in section 5.4.

At 1 m distance from the cask surface [redacted], and the total dose rate at maximum is [redacted]: 0.162 mSv/h = [redacted], with the contributions from primary gamma radiation, neutrons including secondary gamma radiation, and hardware radiation, respectively.

An annual dose (8766 hour annual occupancy) from a standalone storage cask without any additional protection is presented in Figure 5.1-7 as a function of distance from the cask. The angles of 45° and 90° are analysed as they reproduce different shielding geometries (see Figure 5.1-1). As one sees, the distant annual doses are not very different with a marginal excess at [redacted] The neutron radiation share as well as the combined contribution of primary, secondary and ^{60}Co gamma radiation from the fuel hardware (PG+SG) are displayed. The annual dose limits for a controlled area (10 CFR 72.104) and a restricted area (10 CFR 20.1301, recalculated to an annual dose for easier presentation) are shown in red colour. Easy to see, the 0.25 mSv/a-limit is met at a distance of [redacted] At least [redacted] around the storage cask has to be restricted as the dose rate exceeds 0.02 mSv/h within this circle.

Figure 5.1-8 plots the annual dose from the bounding cask array (see Figure 5.1-6) at various distances from the centre of the long side of the array. As in Figure 5.1-7, the total annual dose and the contributions from gamma and neutron radiation are displayed. This conservative arrangement of [redacted] of controlled area on its long side. Almost [redacted] space has to be reserved for the restricted area according to 10 CFR 20.

An alternative solution is to place the storage cask in a storage building. At least [redacted]) are needed, so that the 10 CFR 72.104 dose limit is complied with at a distance of 100 m (see Figure 5.1-9).

After an accident, when all moderator material is assumed to be lost, it is also assumed that no building is left around to provide additional shielding. The 30 days-dose in this case is dominated by neutron radiation but remains safely under the dose limit of 10 CFR 72.106 at 100 m distance. The dose limit is strictly met at [redacted] distance (see Figure 5.1-10).

[redacted]

Figure 5.1-7 Annual dose as a function of distance from the storage cask

[redacted]

Figure 5.1-8 Annual dose from the storage cask array as a function of distance from the array

Table 5.1-1 Total dose rates in the vicinity of a single storage cask under different conditions

Loading Pattern	Maximum Dose Rate at the Shell Surface for Normal and Off-Normal Operation (for Accident), mSv/h			Maximum Dose Rate at the Lid Surface for Normal and Off-Normal Operation (for Accident), mSv/h		
	Cold Cask	Warm Cask	Fuel Failure 3 % / 10 % (100 % bottom / top)	Cold Cask	Warm Cask	Fuel Failure 3 % / 10 % (100 % bottom / top)
TR1	1.47	1.55	[redacted]	0.042	0.041	[redacted]
	(6.29)		[redacted]	(0.324)		[redacted]
TR2	1.00	1.05	-	0.050	0.050	-
	(7.02)		-	(0.370)		-
TR3	1.34	1.42	-	0.047	0.048	-
	(6.61)		-	(0.352)		-

Loading Pattern	Maximum Dose Rate at 1 m from Cask Shell for Normal and Off-Normal Operation (for Accident), mSv/h			Maximum Dose Rate at 1 m from Cask Lid for Normal and Off-Normal Operation (for Accident), mSv/h		
	Cold Cask	Warm Cask	Fuel Failure 3 % / 10 % (100 % bottom / top)	Cold Cask	Warm Cask	Fuel Failure 3 % / 10 % (100 % bottom / top)
TR1	0.162	0.158	[redacted]	0.016	0.016	[redacted]
	(2.32)		[redacted]	(0.110)		[redacted]
TR2	0.133	0.127	-	0.017	0.017	-
	(2.57)		-	(0.129)		-
TR3	0.134	0.129	-	0.017	0.017	-
	(2.43)		-	(0.123)		-

Loading Pattern	Maximum Dose Rate at 2 m from Cask Shell for Normal and Off-Normal Operation (for Accident), mSv/h			Maximum Dose Rate at 2 m from Cask Lid for Normal and Off-Normal Operation (for Accident), mSv/h		
	Cold Cask	Warm Cask	Fuel Failure 3 % / 10 % (100 % bottom / top)	Cold Cask	Warm Cask	Fuel Failure 3 % / 10 % (100 % bottom / top)
TR1	0.093	0.090	[redacted]	0.009	0.009	[redacted]
	(1.22)		[redacted]	(0.072)		[redacted]
TR2	0.076	0.071	-	0.013	0.011	-
	(1.35)		-	(0.083)		-
TR3	0.078	0.073	-	0.013	0.011	-
	(1.28)		-	(0.079)		-

[redacted]

Figure 5.1-9 Annual dose from the storage building as a function of distance

[redacted]

Figure 5.1-10 Annual dose from the unshielded storage cask array under accident storage conditions

Summarising the evaluation, to comply with the dose limit of 0.25 mSv/a to any real individual at or beyond the controlled area boundary (10 CFR 72.104) one has to:

- Make sure that the boundary is [redacted] far off from the storage cask array, or
- Guarantee that the storage casks are placed into a storage building [redacted] In this case, the dose limit is met at the minimum distance of 100 m (10 CFR 72.106).

A detailed site specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10 CFR 72.212.

The complete loss of the moderator material and the storage building as a result of an accident seriously affects the dose generated by the considered bounding storage cask array. Assuming an accident duration of 30 days, the accumulated dose at the controlled area boundary would be [redacted] safely below the dose limit of 50 mSv.

For the accident conditions the compliance with the 10 CFR 72.106 limit is demonstrated without any additional requirements.

List of References

[1] NUREG-2224, Dry Storage and Transportation of High Burnup Spent Fuel
Office of Nuclear Material Safety and Safeguards, November 2020

[2] C.J. Werner (ed.), MCNP User's Manual - Code Version 6.2, LA-UR-17-29981, 2017

[3] 1014-SR-00001, Rev. 0
Safety Analysis Report
Type B(U)F Transport Package CASTOR® geo69
Docket No.: 71-9383, 23.12.2020

5.2 Source Specification

	Name, Function	Date	Signature
Prepared	[redacted]		[redacted]
Reviewed	[redacted]		

The procedure how the bounding source terms for each loading pattern TR1 to TR3 are determined is described in SAR (transport) [1] in detail. For a sake of completeness, the most important properties of the radioactive contents are summarised here.

Each basket position is characterised by the maximum allowed decay heat power (see section 1.2.3). Within one chosen position group (see Figure 5.0-1) all the position have identical requirements. The minimum cooling times required to reach a certain decay heat for a particular loading pattern are presented in Table 5.2-1. Some field are left vacant, because the cooling times are smaller than the minimum cooling times for this particular SNF type (see section 1.2.3).

Table 5.2-1 Minimum cooling times (in years) needed to reach certain decay heat

[redacted]

5.2.1 Gamma Source

The gamma-radiation fuel is analysed in seven energy groups (see Table 5.2-2), representing energies relevant for the dose rate outside of the storage cask. The gamma particles with lower energy are so well shielded that they do not significantly contribute to the outer dose rate. The high energy gamma particles possess tiny source term strengths and, therefore, do not contribute to the dose rate either.

[redacted]

Table 5.2-2 Gamma energy structure

Energy group, i =	1	2	3	4	5	6	7
Average group energy, MeV	0.575	0.85	1.25	1.75	2.25	2.75	3.5
Lower group energy, MeV	0.45	0.7	1	1.5	2	2.5	3
Upper group energy, MeV	0.7	1	1.5	2	2.5	3	4

Besides gamma particles stemming from the fuel pellet stack, there is also a gamma radiation from activated hardware: the end fittings and plenum springs. The primary source of activity in the non-fuel regions of a SNF arise from the activation of ^{59}Co to ^{60}Co. The activities of ^{60}Co are determined during burn-up and depletion calculation together with the calculations for the primary gamma radiation. The flux scaling factors of 0.1 for the top end fittings and of 0.2 for the bottom end fittings and plenum springs are applied according to PNL-6906 [2]. The top handles and the lower parts of the SNF bottom end pieces are conservatively taken into account by assuming that the whole ^{59}Co from the top or bottom end piece is concentrated in the upper respectively lower tie plate.

Yet another gamma source arises from (n,γ) reactions in the materials of the storage cask. This source is properly accounted for in MCNP, when neutron calculations are performed in a coupled neutron-gamma mode, which is the case for the present shielding analysis.

The bounding gamma source terms are reported in Table 5.2-3 for TR1, in Table 5.2-4 for TR2, and in Table 5.2-5 for TR3 decoded after position groups. The sources are normalised to one megagram of heavy metal to account for the differences in the mass of the SNF including the mass in the shielding model.

Table 5.2-3 Bounding gamma source term for TR1

[redacted]

Table 5.2-4 Bounding gamma source term for TR2

[redacted]

Table 5.2-5 Bounding gamma source term for TR3

[redacted]

The total source strength in each gamma energy group is calculated by summation of the 24 axial nodal values stemming from the conservative axial burn-up profile discussed above. The corresponding axial gamma source strength profile used in present shielding calculations are displayed in Figure 5.2-1. Different shapes of the profiles in various groups can be related to the most contributing nuclides.

[redacted]

Figure 5.2-1 Axial gamma source strength distributions

5.2.2 Neutron Source

The neutron sources are determined in an analogous way as the gamma ones. The two relevant neutron energy spectra – from spontaneous fission and from (α,n) reactions – exhibit different axial distributions due to the weaker burn-up towards the ends of the SNF (see Figure 5.2-2). The bounding source terms used for the shielding analysis are reported in Table 5.2-6 for TR1, in Table 5.2-7 for TR2, and in Table 5.2-8 for TR3.

It is checked, whether energy distributions of these two spectra could be realised by internal means of the MCNP code system (continuous energy distributions, see [3]). While the energy spectrum from the spontaneous fission could be nicely described by the Watt spectrum from ^{244}Cm (parameters a = 0.902523 MeV, b = 3.72033 MeV^{-1}, see [3]), for the (α,n) energy spectrum no internal function has been found as it is rather a modified Maxwell distribution currently not implemented in the code. Finally, for (α,n)-neutrons a histogram function generated by the performed burn-up and depletion calculations is utilised in the analysis (see Table 5.2-9).

The subcritical neutron multiplication is properly taken into account during particle transport with MCNP. [redacted]

[redacted]

Figure 5.2-2 Axial neutron source strength distributions

Table 5.2-6 Bounding neutron source for TR1

[redacted]

Table 5.2-7 Bounding neutron source for TR2

[redacted]

Table 5.2-8 Bounding neutron source for TR3

[redacted]

Table 5.2-9 Bounding (α,n)-reaction neutron source as a function of energy

[redacted]

The largest total neutron source per storage cask stems from TR2 and accounts to [redacted] neutrons/s and corresponds to complete loading with the bounding source.

List of References

[1] 1014-SR-00001, Rev. 0
Safety Analysis Report
Type B(U)F Transport Package CASTOR® geo69
Docket No.: 71-9383, 23.12.2020

[2] PNL-6906 Vol. 1 to Vol. 3, UC-85
A. Luksic, Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal, 1989

[3] C.J. Werner (ed.), MCNP User's Manual - Code Version 6.2, LA-UR-17-29981, 2017

5.3 Model Specification

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]	[redacted]	[redacted]

The shielding analysis of the DSS is performed with MCNP6 2.0 [1]. For the storage cask, a separate MCNP calculation is performed for each position group A to F (see Figure 5.0-1) for every twelve sources (seven gamma groups from the fuel pellet stack, two neutron spectra, ^{60}Co radiation from top end fittings, bottom end fittings, and plenum springs).

In this section, the shielding models used in the calculation are discussed. The information about individual constituents of the cask and the materials used in models are described.

In total, a set of shielding models for the storage cask and cask array are prepared to cover all the aspects of the safe storage. Except possible fuel failure scenarios due to the storage period beyond 20 years, none of the off-normal conditions have any impact on the shielding analysis. Therefore, normal and off-normal conditions are generally identical.

The shielding models for a single storage cask are as follows:

[redacted]

The shielding models for a bounding array of storage casks are as follows:

[redacted]

[redacted]

In all the array calculations it is additionally checked, at which distance a boundary of the restricted area has to be set. This boundary is regulated by the restricted boundary dose rate limit of 0.02 mSv/h (2 mrem/h) according to 10 CFR 20.1301.

In order to speed the calculations up, a surface source file is utilised for the calculations of the storage cask arrays. For a generation of the surface source a uniform loading pattern (TR1) is used, because it generates the highest dose rates at the cask surface as well as at distances of 1 m and 2 m (see section 5.4).

5.3.1 Description of the Radial and Axial Shielding Configurations

The technical drawings of the storage cask (see section 1.5) are used to create MCNP models used for the shielding calculations.

The elevation cut through the model is presented in Figure 5.1-2. The axial null of the scale corresponds to the cask bottom edge.

Being of low relevance for the shielding analysis, the screws, compression springs and gaskets are not modelled. When appropriate they are substituted by air, e.g. the heads of the lid bolting or the top of the basket, or by surrounding material, e.g. inside the lid.

For the model with fuel reconfiguration due to failure under normal (off-normal) conditions, an additional fuel region with 3 % (10 %) of the source volume and strength according to [2] is considered at the canister bottom (see Figure 5.3-1). The rubbleised mixture of fuel and cladding is relocated within the basket cell, the mass packing fraction for the rubble is 0.58 [2]. Conservatively it is assumed that the rest of the fuel is not present anymore thus reducing the shielding effects. Solely the bottom nozzles are taken into account in the 10 %-rubble scenario (see Figure 5.3-1, right). Otherwise, the standard sources from the uniform loading pattern are attributed to the remaining fuel pellet

stack and contribute to the external dose rate with a strength of 97 % (90 %). The end fittings and plenum springs from the standard NCT model contribute with a full strength.

After an accident, a 100 % fuel failure is assumed. As no damage of the basket structures occurs (see section 3), the rubble consisting of fuel and cladding (the rest of the SNF structures is conservatively neglected) remains in the basket cell (see Figure 5.3-2).

[redacted]

Figure 5.3-1 Source location in fuel failure models – 3 % (normal conditions, left) and 10 % (off-normal conditions, right)

[redacted]

Figure 5.3-2 Accident rubble of 100 % at the bottom (left) and at the top (right) of the cask

The principal components of the shielding model are explained in the following subsections. The methods and main measurements are presented.

5.3.1.1 Spent Fuel

All 69 SNF are placed into the corresponding basket cells. Conservatively, Atrium-10A design is selected for the shielding model. The main reason for this is the lattice configuration. With the large water channel and part length rods the self-shielding effects are minimised. Additionally, this fuel design has the thinnest cladding and fuel channel thickness among other SNF.

The fuel rods including cladding are modelled individually. The cross section through the fuel model in its lower part, where all the rods are present, is presented in Figure 5.3-3. [redacted]

It is assumed that the SNF are complete and do not contain dummy rods. Possible loss of self-shielding due to possible absence of fuel rods is overcompensated by the actual loss of source strength at this location. The heavy metal mass of the modelled SNF amounts to 184.8 kg.

The end fittings enter the calculations in a simplified fashion as tie plates only. This approach is justified, since the source strength is scaled with the full mass of the end fittings. The plenum springs are modelled homogenised as steel pieces with reduced density (ρ = 0.79 g/cm^3).

[redacted]

Figure 5.3-3 Spent fuel model (dimensions in mm)

5.3.1.2 Basket

The basket in the shielding model (see Figure 5.1-1) consists of [redacted] and round segments made of aluminium (SB-209 5454). The mounting elements are left out and replaced by air. As discussed in section 5.1 and shown in Figure 5.1-3, the thicknesses of the sheets are minimised and the outer diameter of the round segments is reduced according to the design tolerances. [redacted]

5.3.1.3 Shielding Elements

Shielding elements out of aluminium (SB-209 5454) are modelled as solid blocks (see Figure 5.3-4). [redacted]

[redacted]

Figure 5.3-4 Shielding elements

5.3.1.4 Canister

[redacted]

[redacted] The elevation view of the canister is presented in Figure 5.3-5.

[redacted]

Figure 5.3-5 Cask elevation cut (dimensions in mm)

5.3.1.5 Cask Body and Lid

The cask body with lid is modelled with the unfavourable combination of the tolerances in radial direction. [redacted] Several so-called splitting levels (thin black lines in the model illustrations) are implemented into various components of the calculation model in order to increase its statistical performance using variance reduction techniques.

The LAP are modelled as solid pieces made of steel (upper trunnions, SA-479M 414) or DCI (lower tilting studs), the corresponding flattenings of the cask body possess minimum dimensions.

The cask lid is modelled with the lid moderator plate mounted on its lower surface. The bolting of the lid is not modelled; the bolt heads are replaced with air. [redacted]

[redacted]

5.3.1.6 Cooling Fins

The cooling fins are modelled explicitly (see Figure 5.3-6). It was found that homogenisation of this areas does not necessarily lead to conservative evaluation. [redacted]

[redacted]

Figure 5.3-6 Cooling fins (dimensions in mm)

5.3.1.7 Moderators

The moderator rods made of polyethylene ([redacted] are modelled [redacted] They are analysed in two different configurations, hot and cold (under accident conditions not modelled at all), the design geometry representing the situation shortly after loading and the state of thermal equilibrium (see Figure 5.1-4 and Figure 5.1-5). Divergent from the design density of [redacted] the resulting polyethylene at equilibrium is conservatively reduced according to the maximum temperature over the entire length of the rod (no axial temperature profile is assumed, see chapter 3). For the moderator plates (bottom and lid ones) only the density is reduced in equilibrium state, no expansion is implemented.

5.3.1.8 Environment and Detectors

Depending on the modelling situation, the storage cask or array of casks is surrounded by sufficient amount of air to take scattering effects into account. [redacted]

The information about particular dose rates is gained from a mesh of detectors positioned all around the storage cask. A cylindrical mesh is utilised for the model with a single storage cask, while a

rectangular raster is more convenient when exploring the dose rate field around the storage cask array. Besides this geometry-independent mesh of detectors, separate volumetric detectors are modelled in order to control the calculation process.

5.3.2 Shield Regional Densities

Compositions and densities of the materials used in the shielding model are presented in Table 5.3-1. For the moderator rods two densities are given, the nominal one for the shielding configuration shortly after loading (cold), and the low one for the equilibrium configuration (hot). The steel specification for the retention ring ([redacted]) is very similar to that of the [redacted], for this reason no new material has been introduced.

The materials in Table 5.3-1 are arranged from inside to outside. [redacted]

For the moderator material the temperature effects are studied as discussed in section 5.1. [redacted]

The design basis for the material data is given in chapter 8.

Table 5.3-1 Material properties in the shielding model

[redacted]

Table 5.3-1 Material properties in the shielding model (cont.)

[redacted]

Table 5.3-1 Material properties in the shielding model (cont.)

[redacted]

List of References

[1] C.J. Werner (ed.), MCNP User's Manual - Code Version 6.2, LA-UR-17-29981, 2017

[2] NUREG-2224, Dry Storage and Transportation of High Burnup Spent Fuel
Office of Nuclear Material Safety and Safeguards, November 2020

[3] 1014-SR-00001, Rev. 0
Safety Analysis Report
Type B(U)F Transport Package CASTOR® geo69
Docket No.: 71-9383, 23.12.2020

5.4 Shielding Evaluation

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	[redacted]
Reviewed	[redacted]		

As discussed in section 5.3 the MCNP6 [1] code is used for the shielding analysis. The cross section data are based on ENDF/B-VII data. The MCNP code system is benchmarked against experimental data for a broad spectrum of gamma [2] and neutron [3] problems. Described shielding problems cover a wide range of energies and material compositions and involve both scattering and deep penetration. A good agreement between measured and calculated values has been demonstrated for all the validation scenarios.

The dose rates are calculated using volumetric mesh tallies (f4), multiplied by an appropriate flux-to-dose-rate conversion factor, heavy metal mass of the SNF in the shielding model, and by the total source strength (per megagram heavy metal) for every radiation source term. Since the mesh is stretched around the entire geometry of the storage cask, the locations of dose rate maxima are determined explicitly. Its use also allows for the evaluation of the wrapping dose rate distributions. Volumetric detectors are used to determine the dose rates from the storage cask array at various distances.

The calculations are performed separately for six position groups introduced in Figure 5.0-1. The elementary external dose rates from different position groups, folded with corresponding sources and superposed, allow for the evaluation of an arbitrary loading pattern provided it consists of no more than the six position groups defined in chapter 1. In this evaluation, the loading patterns TR1 to TR3 are assessed.

The calculation with the bounding cask array are performed based on the generated surface source file from the individual storage cask loaded according to TR1.

[redacted]

Each set of the MCNP calculations is foreseen with a message digest (md5) providing its unique identification. The summary of the cases is given in Table 5.4-1.

Table 5.4-1 Message digest overview

Calculation Series	Model	Message Digest, md5
Standalone storage cask	[redacted]	
Array of storage casks	[redacted]	

As MCNP calculates fluxes, these values have to be converted into dose rate (dose) using corresponding response functions. The conversion of the spectral neutral and gamma flux density to the ambient equivalent dose is performed with the flux-to-dose-rate conversion factors according to ANSI/ANS-6.1.1-1977 [4]. The conversion factors are exhibited in Table 5.4-2 and Table 5.4-3 for gamma radiation and neutrons, respectively.

Table 5.4-2 Conversion factors for gamma radiation

Gamma Energy, MeV	Conversion Coefficient (Including Quality Factor), mSv/h/(γ/cm²·s)	Gamma Energy, MeV	Conversion Coefficient (Including Quality Factor), mSv/h/(γ/cm²·s)
0.01	3.96E-05	1.4	2.51E-05
0.03	5.82E-06	1.8	2.99E-05
0.05	2.90E-06	2.2	3.42E-05
0.07	2.58E-06	2.6	3.82E-05
0.1	2.83E-06	2.8	4.01E-05
0.15	3.79E-06	3.25	4.41E-05
0.2	5.01E-06	3.75	4.83E-05
0.25	6.31E-06	4.25	5.23E-05
0.3	7.59E-06	4.75	5.60E-05
0.35	8.78E-06	5	5.80E-05
0.4	9.85E-06	5.25	6.01E-05
0.45	1.08E-05	5.75	6.37E-05
0.5	1.17E-05	6.25	6.74E-05
0.55	1.27E-05	6.75	7.11E-05
0.6	1.36E-05	7.5	7.66E-05
0.65	1.44E-05	9	8.77E-05
0.7	1.52E-05	11	1.03E-04
0.8	1.68E-05	13	1.18E-04
1	1.98E-05	15	1.33E-04

Table 5.4-3 Conversion factors for neutrons

Neutron Energy, MeV	Conversion Coefficient (Including Quality Factor), mSv/h/(n/cm²·s)
2.50E-08	3.67E-05
1.00E-07	3.67E-05
1.00E-06	4.46E-05
1.00E-05	4.54E-05
1.00E-04	4.18E-05
1.00E-03	3.76E-05
0.01	3.56E-05
0.1	2.17E-04
0.5	9.26E-04
1	1.32E-03
2.5	1.25E-03
5	1.56E-03
7	1.47E-03
10	1.47E-03
14	2.08E-03
20	2.27E-03

The external radiation levels calculated with MCNP for an individual storage cask are modified by the corresponding statistical uncertainties provided by MCNP in the same tally. Thereby, instead of using a Gaussian error propagation, two standard deviations of MCNP are conservatively added to every single tallied value (penalising error propagation).

A sample input file for MCNP (neutron case) is presented in section 5.5.

5.4.1 Dose Rates in the Near Field of the Cask Under Normal and Off-Normal Conditions

The maximal values of the dose rates in the vicinity of the storage cask for all three loading patterns assuming that the storage cask has recently been loaded (cold shielding model) are presented in Table 5.4-4. The contributions of the gamma and neutron radiation from fuel and fuel hardware are presented as well. The uncertainties from the penalising error propagation are included.

Among three loading patterns analysed, the uniform one (TR1) is the most demanding from the shielding point of view. It generates the maximum dose rate at the storage cask surface as well as at 1 m and 2 m from the cask surface.

The corresponding dose rate wrap ups on the surface of the storage cask loaded with the fuel according to loading pattern TR1 are displayed in Figure 5.4-1. The distributions for other two loading patterns are similar, solely the absolute values change. [redacted] While one can still visually see the contributions from the nozzles and plena in the dose rate distribution at 1 m from the cask (see Figure 5.4-2, left), they nearly disappear at 2 m from the cask (see Figure 5.4-2, right).

Figure 5.4-3 demonstrates fractional standard deviations – Gaussian error propagation – of the calculated dose rate for TR1 at the surface of the cask and at a distance of 2 m from the cask.

The dose rates on the lid side of the storage cask are much lower than on the shell side. [redacted] In 1 m from the cask lid the dose rate distribution is close to be uniform, the irregularities vanish (see Figure 5.4-4, right).

Table 5.4-4 Maximum external dose rates for the cold storage cask

[redacted]

[redacted]

Figure 5.4-1 Dose rate distributions (TR1, cold model) at the surface of the storage cask in mSv/h: fuel gamma (top left), fuel hardware gamma (top right), neutron (bottom left) and total (bottom right)

[redacted]

Figure 5.4-3 Fractional standard deviation distributions at the surface of the storage cask (left) and in 2 m from the cask surface (right) for the cold model in mSv/h

[redacted]

Figure 5.4-4 Total dose rate distributions (TR1, cold model) at the lid surface of the storage cask (left) and in 1 m (right) from the storage cask lid in mSv/h

Table 5.4-5 provides maximum external dose rates for the storage cask under regular storage conditions, when the thermal equilibrium is attained and the moderator material expands (hot model).

[redacted] For the far distances, the cold case is bounding. For that reason, all the storage site calculations are performed with the cold cask model. The dose rate distributions from the hot storage cask are fairly similar to those from the cold one, therefore they are not presented here.

Fuel failure under normal and off-normal conditions are evaluated for TR1 only (see Table 5.4-6), since this loading pattern delivers the highest dose rates. Two cases according to NUREG-2224 [5] are considered: 3 % failed fuel forming a rubble under normal conditions and 10 % fuel failure under off-normal conditions of storage. Fuel reconfiguration is considered for loading after 20 years of dry storage [5], the source terms have been adjusted accordingly by selecting maximum source terms among all possible combinations of fuel assemblies after 20 years (see Table 5.2-1 and [6] for details). The redistribution of the fuel does not affect the shell dose rate in a negative way. The neutron dominated maximum dose rates (at the surface and 2 m from the surface of the storage cask) are lower than those from the failure unaffected shielding models.

Table 5.4-5 Maximum external dose rates for the hot storage cask

[redacted]

In total, sufficiently low dose rates are established in the vicinity of the storage cask in accordance with ALARA practices. Important to mention that the bounding dose rates are presented. In reality, lower values are expected.

Table 5.4-6 Maximum external dose rates assuming fuel failure during storage

[redacted]

5.4.2 Dose Rates in the Near Field of the Cask Under Accident Conditions

The calculated dose rates for the storage cask after design basis accident are presented in Table 5.4-7 for all three loading patterns. Additionally, a 100 % fuel reconfiguration shifted towards the bottom and the top of the storage cask is summarised in Table 5.4-8. [redacted] For this reason, the regular accident cask model with upright oriented casks is selected to evaluate the dose rates at the site boundary.

Table 5.4-7 Maximum external dose rates for design basis accident

[redacted]

Table 5.4-8 Maximum external dose rate for design basis accident with complete fuel damage

[redacted]

5.4.3 Long Term Radiation Load on the Cask Materials

Besides the requirements on the external dose rates or dose equivalents, there are further demands on the radiation load on the storage cask and its materials (see chapter 8). Among other things, it is to be demonstrated that the materials of the storage cask are able to tolerate the radiation load on the long term. For this purpose, the energy doses and neutron flux densities are determined for different components of the storage cask for the three loading patterns (TR1 to TR3). The resulting values are the maxima of the three individual values determined for each loading pattern. The results of the calculations are presented in Table 5.4-9. [redacted]

Table 5.4-9 Energy doses and neutron fluxes for different storage cask components

[redacted]

5.4.4 Dose Rates and Annual Doses at the Storage Site

Regarding the horizontal orientation of the storage cask relative to the nearest site boundary, it is not important, which side of the cask is directed towards this boundary. The azimuthal distribution of the dose rate at 2 m from the storage cask (see Figure 5.4-2) is relatively smooth without stringent maxima. This statement is supported by the range dependence of the annual dose from the standalone cask (see Figure 5.1-7), [redacted] For the calculation of the annual dose an exposure time of 8766 hours is assumed.

[redacted] (see Table 5.4-10). The neutron dose equivalent is mainly due to the spontaneous fission, the major gamma radiation contribution comes from the fuel gammas at energies between [redacted]

Table 5.4-10 Relative contributions to the annual dose at 100 m from the storage cask

[redacted]

Clearly seen that at the minimum distance to the controlled area boundary of 100 m according to 10 CFR 72.106 the annual dose equivalent limit to public of 0.25 mSv according to 10 CFR 72.104 is [redacted] For this reason, the distance to the nearest boundary of the controlled area has to be larger. Alternatively, a deployment of a storage building made of e.g. concrete is possible.

Assuming a bounding array of [redacted] storage casks (see Figure 5.1-6), one can determine a minimum distance to public needed to meet the requirements of 10 CFR 72.104. This point is located in the centre of the long side of the array [redacted] away from the cask array (see Figure 5.1-8). An area of roughly [redacted] has to be classified as restricted to comply with the requirements of 10 CFR 20. For easier understanding the line representing a dose rate limit of 0.02 mSv/h according to 10 CFR 20.1301 is scaled to the annual dose equivalent.

In case it is necessary to meet the requirements of 10 CFR 72.104 at the minimum distance to public of 100 m, a building [redacted] is required around the entire array of the storage casks (see Figure 5.1-9). For this scenario, the annual dose equivalent to any real individual beyond the controlled area would not exceed 0.25 mSv (10 CFR 72.104).

For design basis accident, it is assumed that all [redacted] storage casks have lost their neutron moderators completely. The upright arrangement as discussed in section 0 is retained. For the duration of accident conditions 30 days are implied. The resulting annual dose equivalents are presented in Figure 5.1-10. It can be clearly seen that the limiting total effective dose equivalent of 50 mSv according to 10 CFR 72.106 is safely complied with. [redacted]

List of References

[1] C.J. Werner (ed.), MCNP User's Manual - Code Version 6.2, LA-UR-17-29981, 2017

[2] Daniel J. Whalen, David E. Hollowell, and John S. Hendricks, Photon Benchmark Problems, LA-12196, Los Alamos National Laboratory, 1991

[3] Daniel J. Whalen, David A. Cardon, Jennifer L. Uhle, and John S. Hendricks, Neutron Benchmark Problems, LA-12212, Los Alamos National Laboratory, 1991

[4] American Nuclear Society. Working Group ANS-6.1.1; American National Standards Institute. American national standard neutron and gamma-ray flux-to-dose-rate factors, La Grange Park, Ill.: The Society, 1977

[5] NUREG-2224, Dry Storage and Transportation of High Burnup Spent Fuel
Office of Nuclear Material Safety and Safeguards, November 2020

[6] 1014-SR-00001, Rev. 0
Safety Analysis Report
Type B(U)F Transport Package CASTOR® geo69
Docket No.: 71-9383, 23.12.2020

5.5 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed	[redacted]		[redacted]

Appendix 5-1: Shielding Model for Cask Loading Unit

Appendix 5-2: Sample Input File for MCNP

Appendix 5-1 Shielding Model for Cask Loading Unit

This supplement is focussed on describing the shielding model of the CASTOR® geo69 cask loading unit (CLU) comprising the transfer cask, the transfer lock (or port) and further not explicitly modelled equipment. A shielding evaluation of the CLU is only of interest because of the occupational exposures of the personal in view of ALARA practices, while [redacted]

In general, the shielding analysis of the CASTOR® geo69 CLU represented by the transfer cask only is very similar to that of the storage cask. All the analysis methods are identical with those described in chapter 5. The vast part of the shielding model including contents, basket and canister is identical. The only component which differs is the cask body itself. The transfer cask is designed to be a lightweight loading unit with a water neutron shield needed to facilitate [redacted] To protect the personnel from gamma radiation a lead gamma shield is designed.

A detailed shielding model of the transfer cask is presented in Figure 5.5-1. The cask itself is conservatively modelled without transfer cask lid. The basket is drained, both water chambers are filled with water. [redacted]

Conservatively, the contents of the canister are assumed to be loaded according to the uniform loading pattern (TR1). Typical dose rate distributions on the shell surface of the transfer cask and in 1 m from the cask are displayed in Figure 5.5-2. [redacted]

Regarding handling of the canister during initial loading into the storage cask, the configuration when the storage cask and the transfer cask are connected via transfer port (see Figure 5.5-3) is very important (see chapter 11). [redacted] The maximum external dose rates would be observed, when the canister is positioned in the middle of its travel. The time it takes to perform the whole reloading action between transfer and storage cask including cask closure is limited, nevertheless the exposure of the personnel during this time has to be minimised. The individual handling operations complemented with corresponding dose levels are summarised in chapter 11.

[redacted]

Figure 5.5-1 Shielding model of the transfer cask

[redacted]

Figure 5.5-2 Dose rate distributions (in mSv/h) on the shell surface (left) and in 1 m distance from the transfer cask (right)

[redacted]

Figure 5.5-3 Shielding model of the canister transfer

Appendix 5-2 Sample Input File for MCNP

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

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6 Criticality Evaluation

6.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed	[redacted]		

This chapter describes the proof of subcriticality for the content described in subsection 1.2.3 in accordance with requirements from 10 CFR 72.

The maximum values for the effective neutron multiplication factor k and the calculational bias with its uncertainty Δk_U with a 95 % probability at a 95 % confidence level fulfil the acceptance criteria $k + \Delta k_U < 0.95$ for all considered cask loadings and demonstrate the compliance with requirements for normal, off-normal and accident conditions during handling, packaging, transfer and storage, as required by § 72.124 and § 72.236.

6.1 Discussion and Results

	Name, Function	Date	Signature
Prepared	[redacted]		[redacted]
Reviewed	[redacted]		

The cask CASTOR® geo69 (referred to as "storage cask") consists of the cask body (cast iron, polyethylene) with bolted lid system (stainless steel, polyethylene), as described in section 1.2, and is used to accommodate the cylindrical canister (stainless steel) with bolted lid system (stainless steel) during storage. The fuel basket is designed to accommodate up to 69 BWR FA and is placed into the canister.

During the loading and unloading of the FA into/out of the canister and for the transfer of the canister into/out of the storage cask a transfer cask, as described in section 1.2, is used. The body of the transfer cask consists of stainless steel and lead shields combined with two independent water chambers for optimized protection against gamma and neutron radiation.

For ensuring the criticality safety of the storage and transfer casks, a combination of the following design measures is used:

- limitation of the fissile content of the fuel,
- geometrical positioning of the FA within a fuel basket and
- fixed neutron absorbing structures in the fuel basket.

The following conservative assumptions are made in the criticality safety analysis:

[redacted]

The applicable codes and standards are summarized as follows:

- 10 CFR 72,
- ANSI/ANS-8.1-2014 [1].

The storage cask is designed to exclude the water leakage into the canister cavity under normal, off-normal and accident conditions during storage as shown in the respective analyses in Section 3.5 and 3.6. Due to a very low reactivity of dry fuel, the behavior of the spent fuel as a result of accident conditions during dry storage (ACS) and the storage periods beyond 20 years do not need to be explicitly evaluated and are bounded by the reactivity of the fully flooded cask with pure unborated water, as assumed in the bounding model for normal conditions of storage (NCS), described in subsections 6.3.1.2 and 6.3.2.

As shown in chapters 3 and 12, the off-normal and accident conditions during the loading and unloading have no impact on the structure of the canister, the fuel basket and the content. The normal, off-normal and accident conditions during handling, packaging and transfer are thus also bounded by the reactivity of the fully flooded cask with pure unborated water, as assumed in the bounding model for NCS, described in subsections 6.3.1.2 and 6.3.2.

As the outer shell structures of the storage and transfer casks provide different neutron reflection conditions, the full neutron reflection is separately determined for fully flooded storage and transfer casks.

The final results of the criticality evaluation as well as the corresponding internal moderation and external moderation and reflection conditions are provided in the summary Table 6.1-1 for all investigated FA types (FA no.). The results are obtained under bounding conditions for NCS, as discussed in subsections 6.4.2.1 through 6.4.3, and include the maximum effective neutron multiplication factors k and the calculational bias with its uncertainty Δk_U. All contributions to the final results are determined with a 95 % probability at a 95 % confidence level, as discussed in subsection 6.5.2.1 and Appendix 6-1.

For reference purposes a unique alphanumeric identification number (calc. ID) is assigned to each calculation.

The following cases are evaluated in the criticality safety analysis:

- infinite array of flooded storage casks (reference case for NCS),
- infinite array of dry storage casks,
- single, fully reflected storage cask,
- single, fully reflected transfer cask.

The evaluated cases are bounding for normal, off-normal and accident conditions during handling, packaging, transfer and storage.

Table 6.1-1 Summary table of criticality evaluation

FA no.	Case	Infinite array of flooded storage casks (bounding model for NCS)		Infinite array of dry storage casks	
	Internal moderation	100 % (Water)		0 % (Void)	
	External moderation and reflection	0 % (Void)		0 % (Void)	
	Fuel type	$k + \Delta k_U$	calc. ID	$k + \Delta k_U$	calc. ID
1	GE 8x8-1	0.84386	20tZ09ny02	0.35051	20EF09bR02
2	GE 8x8-2	0.85061	20da09vS02	0.34374	20hN09sh02
3	SPC 8x8-2	0.84625	20rq09Tj02	0.33861	20DB09dw02
4	GE9B 8x8	0.88137	20lI09Vo02	0.36399	20JJ09zw02
5	GE12 LUA	0.90514	20Xb09ei02	0.39113	20La09rj02
6	ATRIUM-10A	0.93729	20nS09hZ02	0.42457	20rk09gn02

Table 6.1-1 Summary table of criticality evaluation (continued)

FA no.	Case	Single, fully reflected storage cask		Single, fully reflected transfer cask	
	Internal moderation	100 % (Water)		100 % (Water)	
	External moderation and reflection	100 % (Water)		100 % (Steel)	
	Fuel type	$k + \Delta k_U$	calc. ID	$k + \Delta k_U$	calc. ID
1	GE 8x8-1	0.84387	20hr09qR02	0.84410	21Vu02YZ08
2	GE 8x8-2	0.85034	20bM09JV02	0.85044	21lv02wr08
3	SPC 8x8-2	0.84629	20AV09iY02	0.84641	21HQ02yr08
4	GE9B 8x8	0.88145	20ed09nP02	0.88149	21oa02SS08
5	GE12 LUA	0.90527	20Np09QB02	0.90541	21NW02dx08
6	ATRIUM-10A	0.93725	20Lf09eS02	0.93740	21jq02Sv08

As provided in Table 6.1-1, the results demonstrate the compliance with requirements for normal, off-normal and accident conditions during handling, packaging, transfer and storage, as required by § 72.124 and § 72.236.

List of References

[1] ANSI/ANS-8.1-2014: Nuclear Criticality Safety in Operations with Fissile Material Outside Reactors.

6.2 Spent Fuel Loading

	Name, Function	Date	Signature
Prepared	[redacted]		[redacted]
Reviewed	[redacted]		[redacted]

The criticality safety analysis is performed for the content described in subsection 1.2.3. The calculation model takes upper limits for the use of unirradiated fissile material (max. fuel density, max. fuel enrichment) in the FA into account. All investigated fuel types are in a solid metal dioxide form (UO_2). [redacted]

6.3 Model Specification

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

6.3.1 Description of Calculational Model

6.3.1.1 General Considerations

This subsection addresses the assessment methodology used to evaluate criticality of the cask.

The CASTOR® geo69 cask is designed for the accommodation of different types of BWR spent FA. To take a large number of relevant model parameters into account, for example the fabrication tolerances and the uncertainties in material compositions of FA or basket structures, the calculations of the neutron multiplication factors are performed using bounding models.

The analyzed FA represent square-pitched lattices of fuel rods. The qualitatively same impact of some model parameters on the system reactivity for all FA no. can be assumed. For example, the decrease of the moderator density or the increase of the absorber concentration of the fuel basket structures will reduce the reactivity of all FA no.

However, the distinctions of different FA no. lead to quantitative differences in the reactivity impact of the same model parameter, i.e. the criticality calculations must be performed for each FA type and each cask loading pattern separately. For such calculations the same set of conservative model parameters can be applied.

Based on these assumptions, bounding calculation models (bounding models) for NCS and ACS are derived as described below and shown in Figure 6.3-1.

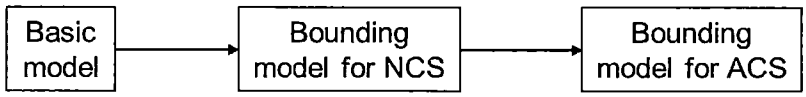

Figure 6.3-1 Development of bounding models

The bounding model for NCS is determined using sensitivity analyses based on a basic calculation model (basic model), as described in subsection 6.4.2.1. The basic model is based on the storage cask and content descriptions from section 1.2 and contains either nominal, representative or expected to be bounding values for geometry, material compositions and densities of the cask, fuel basket and content.

Based on this basic model sensitivity analyses are performed for a full homogeneous cask loading with a reference FA (ATRIUM-10A) only, as described in subsection 6.4.2.1. The sensitivity analyses bound all fabrication tolerances, uncertainties in material compositions, axial and radial FA displacements within the basket receptacles as well as optimum moderation conditions. The effect

of a variation of a certain parameter is described as a deviation (Δk in pcm, i.e. 10^{-5}) of the calculated k-value to the k-value of the basic model. If the variation of a certain parameter leads for the reference FA to an increase in reactivity, the same behavior can be expected for all the other FA no. The corresponding basic model of each FA no. is then adapted according to the possible tolerance range of this parameter.

With the above mentioned approach a bounding model for NCS for each FA no. is developed. The bounding model for NCS is described in subsections 6.3.1.2 and 6.3.2.

On the basis of this bounding model for NCS additional proof of its conservativity as well as analyses concerning reflection conditions (separately for storage and transfer casks) and the cask behavior under ACS are usually performed leading for each FA no. to a bounding model for ACS. The proof of the conservativity of the bounding model for NCS and the analysis of the reflection conditions are provided in subsection 6.4.2.3.

As the damage of the cask wall and bottom as well as the loss of integrity of the cask and canister lid systems under ACS are excluded, as shown in chapters 3 and 12, no explicit bounding model for ACS is developed. ACS are bounded by the reactivity of the fully flooded cask with pure unborated water, as assumed in the bounding model for NCS, described in subsections 6.3.1.2 and 6.3.2.

Based on the bounding model for NCS and taking into account the full neutron reflection for storage and transfer casks, the compliance with the requirements of § 72.124 and § 72.236 is demonstrated in subsection 6.4.3.

Every possible mixed loading of the cask with FA no. for which safe subcriticality can be demonstrated is bounded by the maximum effective neutron multiplication factor k of a full homogeneous cask loading with one of these FA no.

6.3.1.2 Model Configuration

In the bounding calculation model for NCS the cylindrical canister and cask body with the corresponding bottom and top (lid system) structures are considered. The model represents an infinite array of densely packed and fully flooded storage casks. The volume outside the casks is filled with void.

The calculation model does not include an explicit consideration of neither the sealing system nor other outer parts, e. g. moderator rods inside the cask body, cooling fins, trunnions or impact limiters, as well as the axial gaps between the canister and the cask.

The radial and axial cross sections of the bounding model for NCS are shown in Figure 6.3-2 and Figure 6.3-3.

[redacted]

Figure 6.3-2 Radial cross section of the bounding model for NCS (dimensions in mm)

[redacted]

Figure 6.3-3 Axial cross section of the bounding model for NCS (dimensions in mm)

[redacted]

[redacted]

[redacted]

[redacted]

Figure 6.3-4 Cross sections of the analyzed FA types

[redacted]

Figure 6.3-5 Radial cross section of the bounding model for NCS for a [redacted]

[redacted]

Figure 6.3-6 Axial cross section of the bounding model for NCS for a [redacted]

Compared to the basic model used as a starting point for the development of the bounding model for NCS, as described in subsection 6.4.2.1, the following modifications are incorporated in the bounding model for NCS (cf. subsection 6.4.2.2):

- bounding material densities and compositions,
- maximum clad inner diameter,
- minimum clad outer diameter,
- minimum inner dimension of basket receptacles,
- radial displacement of all FA towards the center of the fuel basket.

6.3.1.3 Transfer Cask

In the calculation model of the transfer cask, which is placed instead of the storage cask, only the relevant structures of the transfer cask (bottom, lid and shell) are modelled. The calculation model of the transfer cask is used only for the evaluation of reflection conditions during handling operations, as described in subsection 6.4.2.3.3.

The radial and axial cross sections of the calculation model for the transfer cask are shown in Figure 6.3-7 and Figure 6.3-8.

[redacted]

Figure 6.3-7 Radial cross section of the calculation model for the transfer cask (dimensions in mm)

[redacted]

Figure 6.3-8 Axial cross section of the calculation model for the transfer cask (dimensions in mm)

6.3.2 Cask Regional Densities

The bounding material densities and compositions are determined using the sensitivity analysis, as described in subsection 6.4.2.1, and are listed in Figure 6.4-1 and Table 6.4-2.

The fuel basket contains fixed neutron absorbing structures (boronated basket sheets). The presence and the proper distribution of boron in the basket sheets at time of fabrication are ensured by quality measures. Loss of absorber material as a result of physical, chemical and corrosive mechanisms can be excluded according to chapter 3.

The loss of absorber material through direct neutron absorption (and, thus, transmutation to a non-absorbing isotope) is inconsequential because any measurable depletion would take millions of years as a result of the extremely low neutron flux levels in a subcritical system.

[redacted]

6.4 Criticality Evaluation

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

6.4.1 Calculational or Experimental Method

The criticality calculations for the CASTOR® geo69 are performed using the 3-dimensional Monte-Carlo program KENO-VI from the SCALE 6.2 code package [1]. The neutron multiplication factors are calculated using the 252-group neutron cross sections based on the ENDF/B-VII.1 evaluation (V7.1-252n, T = 293 K).

[redacted]

In the criticality calculations the number of neutron generations with 20,000 neutrons per generation as well as the number of first neutron generations to be skipped is chosen in such a way, that the standard deviation of the calculated neutron multiplication factors is below 20 pcm.

[redacted]

The criticality safety analysis as well as the validation of the used calculation system, i.e. the Monte-Carlo program together with the cross-section library, is performed under the same or comparable boundary conditions.

6.4.2 Fuel Loading or other Contents Loading Optimization

Based on the basic model, the sensitivity analysis for the full cask loading with the most reactive FA type (ATRIUM-10A) is performed and bounding model for NCS is derived, as described in section 6.3.

6.4.2.1 Development of the Bounding Model for NCS

The sensitivity analyses for the development of the bounding model for NCS include the following evaluations:

- material densities and compositions,
- pellet diameter,
- length of the active zone,

- clad inner and outer diameter,
- thickness of the FA internals,
- inner and outer dimensions of the FA channel,
- axial and radial displacement (including clustering) of the FA,
- axial position of part-length fuel rods,
- FA orientation within basket receptacles,
- temperature,
- dimension of the basket receptacles,
- thickness of the basket sheets,
- thickness of the outer sheets,
- canister dimensions,
- radial displacement of the canister within the cask cavity,
- outer boundary conditions for the cask,
- partial vertical and horizontal flooding of the cask cavity,
- moderator rod material in the cask wall,
- shielding element structure.

The results of the analysis are presented in subsections 6.4.2.1.1 through 6.4.2.1.22.

6.4.2.1.1 Material Densities and Compositions

[redacted]

[redacted]

[redacted]

Table 6.4-1 Determination of material densities for the bounding model

[redacted]

Table 6.4-2 Determination of material compositions for the bounding model

[redacted]

6.4.2.1.2 Pellet Diameter

The pellet diameter is parametrically varied within a range of -200 µm/+150 µm at a fixed fuel density of [redacted] The results given in Table 6.4-3 and Figure 6.4-3 show no statistically significant influence of this parameter on the reactivity.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.3 Length of Active Zone

The length of the active zone is parametrically varied between ±20 mm, that corresponds to a height of about two single fuel pellets. The results provided in Table 6.4-3 and Figure 6.4-3 show no statistically significant influence of this parameter on the reactivity.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.4 Clad Inner and Outer Diameter

The clad inner diameter is varied between -150 µm/+200 µm; the clad outer diameter between ±200 µm. The results for both variations are given in Table 6.4-3 and Figure 6.4-3. It can be seen that the clad thickness has a significant influence on the reactivity.

As a result, the maximum inner and minimum outer clad diameters are considered in the bounding model.

6.4.2.1.5 Thickness of the Water Channel

The thickness of the water channel of the reference FA (ATRIUM-10A) is parametrically varied by the variation of the outer channel dimension between ±200 µm at a fixed inner channel dimension. The results given in Table 6.4-3 and Figure 6.4-3 show a significant increase of reactivity at smaller thicknesses of the water channel.

As a result, the minimum thicknesses of the FA internals, as already implemented in the basic model, are considered in the bounding model.

6.4.2.1.6 Inner and Outer Dimensions of the FA Channel

The inner and outer dimensions of the FA channel are separately varied between ±200 µm, resulting in a variation of the wall thickness of the FA channel. The results of criticality calculations are given in Table 6.4-3 and Figure 6.4-3. It can be seen that the wall thickness of the FA channel has no statistically significant influence of the reactivity.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.7 Axial Displacement of the FA

The axial displacement of the FA is varied between ±40 mm. The results provided in Table 6.4-3 and Figure 6.4-3 show no significant influence on reactivity.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.8 Radial Displacement of the FA

The reactivity impact of the radial displacement of the FA within the fuel basket is parametrically investigated via a simultaneous shift of all FA towards the basket center [redacted] or towards the basket periphery [redacted] The results of these calculations are shown in Table 6.4-3 and Figure 6.4-3. It can be seen that the displacement towards the basket center leads to a significant increase of reactivity.

As a result, the maximum possible displacements of the FA towards the basket center are considered in the bounding model.

6.4.2.1.9 Clustering

Additionally to the radial displacement of all FA towards the basket center or the basket periphery, as discussed in subsection 6.4.2.1.8, the building of FA conglomerates (clustering) within the fuel basket is investigated. The selected configuration is shown in Figure 6.4-1 and results in a significant decrease of reactivity (see Table 6.4-3) compared to the reference (basic) model with FA centered within the basket receptacles.

As a result, the maximum possible displacements of the FA towards the basket center are considered in the bounding model, as discussed in subsection 6.4.2.1.8.

[redacted]

Figure 6.4-1 FA clustering

6.4.2.1.10 Axial Position of Part-Length Fuel Rods

The axial position of the part-length fuel rods is varied up to 40 mm beginning at the lower edge of the full-length fuel rods. The results presented in Table 6.4-3 and Figure 6.4-3 indicate the reactivity decrease by the axial shift of the part-length fuel rods towards the center of the active zone.

As a result, no axial displacement of the part-length fuel rods, as already implemented in the basic model, is considered in the bounding model.

6.4.2.1.11 FA Orientation within Basket Receptacles

During the basket loading with FA different orientations of the FA internals (water rods/channel) relative to the neighboring FA are possible. This effect is investigated via a fuel lattice rotation of 90°, 180° and 270° for exemplary selected FA in the calculation model. The investigated configuration is presented in Figure 6.4-2 and shows no statistically significant influence (see Table 6.4-3) of the FA orientation within the basket receptacles on the reactivity.

As a result, the same orientation of all FA, as already implemented in the basic model, is considered in the bounding model.

[redacted]

Figure 6.4-2 FA orientation

6.4.2.1.12 Temperature

The temperature of the fuel and the structure materials is parametrically varied between 0 °C and 170 °C. The results presented in Table 6.4-3 and Figure 6.4-3 confirm the reactivity decrease by increased temperature due to the Doppler broadening.

As a result, the room temperature (20 °C), as already implemented in the basic model, is considered in the bounding model.

6.4.2.1.13 Dimension of the Basket Receptacles

The inner dimension of the basket receptacles is parametrically varied within the possible tolerance range between [redacted] at a fixed thickness of the basket sheets. The results provided in

Table 6.4-3 and Figure 6.4-3 show a significant reactivity increase by decreased basket receptacles.

As a result, the minimum possible inner dimension is applied to all basket receptacles in the bounding model.

6.4.2.1.14 Thickness of the Basket Sheets

The thickness of the borated aluminium sheets of the basket is parametrically increased up to [redacted] As shown in Table 6.4-3 and Figure 6.4-3, the increased thickness of the basket sheets leads to a significant decrease of reactivity.

As a result, the minimum possible thickness of the basket sheets, as already implemented in the basic model, is considered in the bounding model.

6.4.2.1.15 Thickness of the Outer Sheets

The thickness of the outer basket sheets is parametrically increased up to [redacted] The results given in Table 6.4-3 and Figure 6.4-3 show no statistically significant influence of this parameter on the reactivity.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.16 Canister Dimensions

The reactivity impact of the canister dimensions, such as inner and outer diameter (I.D. and O.D.) as well as the wall thickness (T), is investigated via calculations for all possible combinations of minimum and maximum parameter values. The results are provided in Table 6.4-3 and confirm that the parameter set implemented in the basic model (maximum O.D. and T) is conservative.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.17 Radial Displacement of the Canister

The radial displacement of the canister within the cask cavity is investigated via a canister shift towards the cask wall. The result is provided in Table 6.4-3 and shows no influence of the radial canister position on the reactivity.

As a result, the canister centered within the cask cavity, as already implemented in the basic model, is considered in the bounding model.

6.4.2.1.18 Outer Boundary Conditions

The outer boundary conditions, such as external moderation and reflection, are evaluated for a single cask as well as for an infinite array of similar densely packed casks. For a single cask, the

full reflection is achieved by surrounding the cask with 20 cm water reflector. The results of the investigations with and without external moderation (water or void) are shown in Table 6.4-3. It can be seen that the external moderation has no impact on the reactivity due to the thick cask wall.

As a result, an infinite array of similar, densely packed casks without external moderator, as already implemented in the basic model, is considered in the bounding model.

6.4.2.1.19 Partial Flooding of the Cask

The partial flooding of the cask is investigated for the vertical and horizontal orientations of the cask. The results for both orientations are given in Table 6.4-3 and Figure 6.4-3 and indicate that the fully flooded condition is conservative.

As a result, the fully flooded cask cavity with water (ρ = 1.0 g/cm^3), as already implemented in the basic model, is considered in the bounding model.

6.4.2.1.20 Moderator Rod Material

The impact of the moderator rod material as well as of the corresponding boreholes on the reactivity is investigated. The results are given in Table 6.4-3 and indicate that neglecting the moderator material and replacing it by the material of the cask body (cast iron) is conservative.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.21 Shielding Element Structure

The reactivity impact of the drain support in one shielding element is investigated. The results are given in Table 6.4-3 and indicate that neglecting the drain support and replacing it by the material of the shielding element (aluminium) is conservative.

As a result, no changes need to be considered in the bounding model.

6.4.2.1.22 Summary Results of Sensitivity Analysis

The results of the sensitivity calculations described in the previous subsections are summarized in Table 6.4-3 and presented in Figure 6.4-3.

Table 6.4-3 Summary results of the sensitivity analysis

	FA no. 6	
k-value for basic model	[redacted]	20dr09cc03
variation	**Δk, pcm**	**calc. ID**
Pellet diameter, Δ in μm		
-200	-75	20ln09oF03
-150	-30	20un09uS03
-100	-41	20WZ09nh03
-50	-15	20ai09fw03
0	0	20Sx09iF03
50	12	20Rm09fs03
100	-9	20Xv09ra03
150	36	20xK09Vg03
Length of active zone, Δ in mm		
-20	8	20Dt09zS03
-15	10	20WF09aQ03
-10	-2	20cH09Cy03
-5	-34	20mw09cw03
0	0	20hr09qH03
5	-6	20aQ09do03
10	3	20MU09hO03
15	-21	20yX09zr03
20	-31	20So09UA03
Clad inner diameter, Δ in μm		
-150	-452	20YL09eg03
-100	-339	20Pk09nU03
-50	-170	20MR09KC03
0	0	20Gz09tK03
50	168	20yt09Ow03
100	272	20gT09vt03
150	474	20hd09Py03
200	625	20xU09cC03
Clad outer diameter, Δ in μm		
-200	759	20GE09hI03
-150	583	20vO09xi03
-100	333	20nR09ZP03
-50	222	20OQ09kK03
0	0	20ES09hD03
50	-207	20Sn09oe03
100	-370	20Ww09qY03
150	-613	20aG09kN03
200	-760	20XV09iY03

	FA no. 6	
k-value for basic model	[redacted]	20dr09cc03
variation	**Δk, pcm**	**calc. ID**
Thickness of water channel, Δ in μm		
-200	66	20BC09Zb03
-150	64	20ZW09cK03
-100	13	20vx09FJ03
-50	13	20TC09LA03
0	0	20rJ09Jb03
50	-46	20Yj09vu03
100	-51	20RM09rZ03
150	-71	20ho09Py03
200	-86	20yJ09Fb03
Inner dimension of FA channel, Δ in μm		
-200	-28	20HW09KZ03
-150	1	20ee09xg03
-100	15	20RN09vw03
-50	-36	20kO09QD03
0	0	20Sz09vW03
50	-5	20AZ09tW03
100	-14	20FI09qe03
150	-39	20MI09mK03
200	-29	20JT09PF03
Outer dimension of FA channel, Δ in μm		
-200	-22	20DO09FD03
-150	-25	20Vz09ap03
-100	-6	20yW09yN03
-50	21	20Cf09CN03
0	0	20vJ09AI03
50	37	20Ut09wv03
100	31	20Rz09FI03
150	-1	20yB09bg03
200	-5	20EB09cR03
Axial displacement of FA, Δ in mm		
-40	-6	20xD09kv03
-30	-17	20wM09EO03
-20	12	20jx09yn03
-10	-30	20yD09HL03
0	0	20oW09RI03
10	-20	20II09TN03
20	-47	20pX09qY03
30	15	20fW09Bo03
40	9	20aZ09XJ03

Table 6.4-3 Summary results of the sensitivity analysis (continued)

	FA no. 6	
k-value for basic model	0.92065	20dr09cc03
variation	**Δk, pcm**	**calc. ID**
Radial displacement of FA, Δ in mm		
[redacted]	510	20sO09Bl03
	459	20XO09Xb03
	413	20jJ09Yw03
	354	20Jz09Oo03
	273	20mu09jy03
	152	20vz09EF03
	0	20hd09rU03
	-135	20el09ST03
	-336	20Ee09Pg03
	-519	20Nn09fU03
	-739	20xe09uW03
	-967	20HE09KN03
	-1359	20tq09jA03
Axial position of part-length fuel rods, Δ in mm		
0	0	20Ri09vC03
5	-7	20Py09oN03
10	-33	20uF09Jx03
15	-70	20lo09BY03
20	-78	20rZ09Au03
25	-85	20ly09pr03
30	-145	20GX09LP03
35	-118	20is09cK03
40	-120	20JH09eK03
FA orientation within basket receptacles		
Clustering	-235	20ek09Fc03
Rotation	19	20SY09XK03
Material temperature, °C		
0	-12	20nN09Et03
10	-36	20Jx09CU03
20	0	20lJ09Xw03
30	-124	20xv09pr03
40	-253	20oR09DF03
70	-483	20MH09BO03
120	-1034	20Gr09xg03
170	-1490	20Fq09vG03
Dimension of basket receptacles, Δ in mm		
[redacted]	179	20SD09Wf03
	0	20oK09za03
	-214	20rn09KB03
	-368	20WN09UL03
	-563	20ZY09dL03
Thickness of basket sheets, Δ in mm		
[redacted]	0	20EZ09ED03
	-136	20Rm09NM03
	-224	20ar09Dj03
	-315	20Xv09jv03
	-488	20Gi09Ao03
	-551	20FM09DC03

	FA no. 6	
k-value for basic model	0.92065	20dr09cc03
variation	**Δk, pcm**	**calc. ID**
Thickness of outer sheets, Δ in mm		
[redacted]	0	20ut09la03
	-4	20Rt09xV03
	7	20sY09rg03
	-2	20bF09LL03
	-27	20mJ09yW03
	5	20lu09oH03
Canister dimensions, mm		
[redacted]	-248	20zE09by03
	-47	20jv09Ba03
	26	20ih09lf03
	0	20Jv09pj03
Radial displacement of canister		
to the cask wall	-24	20aL09cq03
Outer boundary conditions		
Array, ext. mod. 0 %	0	20wm09SY03
Array, ext. mod. 100 %	4	20Wi09TH03
Single, ext. mod. 100 %	-8	20Mr09LJ03
Single, ext. mod. 0 %	-39	20pE09xn03
Partial flooding (vertical), Δ in cm		
-200	-619	20UM09uz03
-150	-405	20Pb09Kz03
-100	-193	20zN09RC03
-90	-164	20Yo09nW03
-80	-80	20Hd09Uc03
-70	-23	20qo09Jv03
-60	29	20lA09eP03
-50	5	20VA09ro03
-40	-32	20jd09sj03
-30	17	20SR09OW03
-20	31	20ab09at03
-10	0	20cd09aw03
0	0	20rt09gh03
Partial flooding (horizontal), number of dry FA rows		
0	0	20kO09BF03
1	-230	20jx09MZ03
2	-474	20OH09eX03
3	-918	20Fy09Vy03
4	-1523	20yt09LW03
5	-2463	20sR09TV03
Moderator rod material		
Neglected	0	20JQ09jh03
PE	-189	20Fc09va03
Void	12	20li09CF03
Water	-236	20hl09FS03
Shielding element structure		
Drain support	-61	20Jl09UQ03

[redacted]

Figure 6.4-3 Graphical representation of the summary results

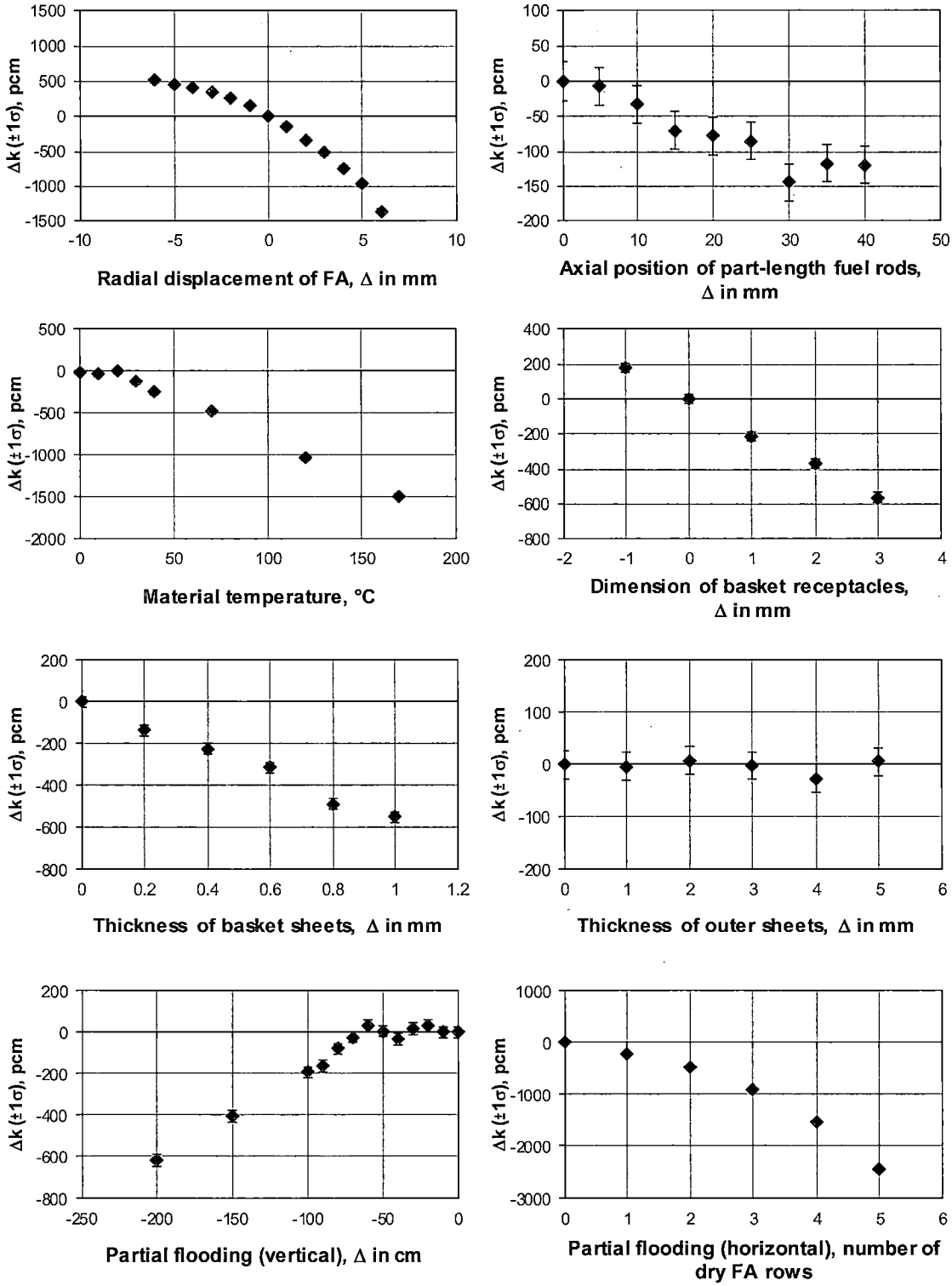

Figure 6.4-3 Graphical representation of the summary results (continued)

6.4.2.2 Results for the Bounding Model for NCS

As discussed in subsection 6.4.2.1, the following conservative assumptions for the basic model are confirmed:

- fully flooded gap between fuel pellet and clad,
- nominal pellet diameter,
- nominal length of the active zone,
- minimum thickness of the FA internal structures,
- nominal dimensions of the FA channel,
- nominal axial position of the FA,
- axial position of part-length fuel rods beginning at the lower edge of the active zone,
- same orientation of all FA within the basket receptacles,
- nominal temperature,
- nominal thickness of the basket sheets,
- nominal thickness of the outer sheets,
- maximum canister outer diameter with maximum wall thickness,
- no radial displacement of the canister within the cask,
- void outside the cask (external moderation 0 %),
- infinite array of densely packed casks,
- fully flooded cask (internal moderation 100 %),
- neglected moderator rods and sheets,
- neglected drain support in the shielding element.

Based on the sensitivity analysis in subsection 6.4.2.1, the following changes are applied to the basic model to develop the bounding model for NCS:

- bounding material densities and compositions,
- maximum clad inner diameter,
- minimum clad outer diameter,
- minimum inner dimension of the basket receptacles,
- radial displacement of all FA towards the center of the fuel basket.

The criticality calculations using the bounding model for NCS are performed for all analyzed FA. The results are summarized in Table 6.4-4 and include the maximum effective neutron multiplication factors k and the calculational bias with its uncertainty Δk_U. All contributions to the final results are determined with a 95 % probability at a 95 % confidence level, as discussed in subsection 6.5.2.1 and Appendix 6-1.

A sample input file for the bounding model for NCS with the FA of type ATRIUM-10A is provided in Appendix 6-2.

Table 6.4-4 Results for the bounding model for NCS

FA no.	Case	Infinite array of flooded storage casks (bounding model for NCS)	
	Internal moderation	100 % (Water)	
	External moderation and reflection	0 % (Void)	
	Fuel type	$k + \Delta k_U$	calc. ID
1	GE 8x8-1	0.84386	20tZ09ny02
2	GE 8x8-2	0.85061	20da09vS02
3	SPC 8x8-2	0.84625	20rq09Tj02
4	GE9B 8x8	0.88137	20ll09Vo02
5	GE12 LUA	0.90514	20Xb09ei02
6	ATRIUM-10A	0.93729	20nS09hZ02

6.4.2.3 Proof of Conservativity of the Bounding Model for NCS

Additional evaluations as a proof of the conservativity of the bounding model for NCS is based on this model and includes:

- partial cask loading,
- planar fuel enrichment,
- reflection conditions for storage and transfer casks.

The results of the analysis are presented in subsections 6.4.2.3.1 through 6.4.2.3.3.

6.4.2.3.1 Partial Cask Loading

The impact of different empty basket positions on the reactivity is evaluated. The investigated single empty basket positions (No. 6, 9, 32, 49, 51, 63, 64 and 69) are radially distributed within the fuel basket, as shown in Figure 6.4-4. The results of these different calculations are given in Table 6.4-5 and Figure 6.4-5 and confirm that the full cask loading is conservative.

[redacted]

Figure 6.4-4 Evaluated empty basket positions

Table 6.4-5 Evaluation results for the partial cask loading

	FA no. 6	
k-value for NCT model	[redacted]	20Xg09mz04
variation	**Δk, pcm**	**calc. ID**
Number of empty basket position		
6	-93	20In09oF03
49	-409	20un09uS03
63	-1002	20WZ09nh03
69	-1463	20ai09fw03
64	-1197	20Sx09iF03
51	-741	20Rm09fs03
32	-415	20Xv09ra03
9	-168	20xK09Vg03

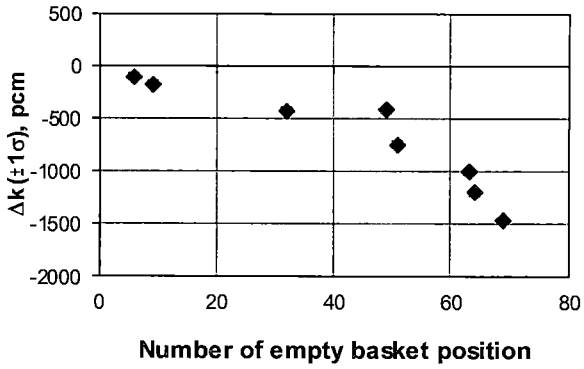

Figure 6.4-5 Graphical representation of results for the partial cask loading

6.4.2.3.2 Planar Fuel Enrichment

[redacted]

[redacted]

[redacted]

Figure 6.4-6 Cross section of the fuel lattice and the group numbers (Group no.) with different enrichments in the generic model

Table 6.4-6 Investigated enrichment distributions

[redacted]

The results of criticality calculations given in Table 6.4-7 and Figure 6.4-7 [redacted] and confirm that the use of the planar-averaged uniform enrichment is conservative.

Table 6.4-7 Evaluation results for the planar fuel enrichment

[redacted]

Figure 6.4-7 Graphical representation of results for the planar fuel enrichment

6.4.2.3.3 Reflection Conditions

The impact of different reflection conditions is separately evaluated for the storage and for the transfer casks. For the storage cask, the analysis is performed using the bounding model for NCS, as described in subsections 6.3.1.2 and 6.3.2. For the transfer cask, the structures of the storage cask in the bounding model for NCS are replaced by the structures of the transfer cask, as described in subsection 6.3.1.3. Single cask configurations with the following reflector materials are investigated:

- void (vacuum),
- 20 cm water shell (1 g/cm^3),
- 50 cm concrete shell (2.3 g/cm^3, referred to as "reg-concrete" in the Standard Composition Library chapter of the SCALE manual),
- 50 cm steel shell (8.03 g/cm^3, referred to as "SS316" in the Standard Composition Library chapter of the SCALE manual).

The results of these calculations are given in Table 6.4-8 and show no significant reactivity increase compared to the bounding model for NCS. In a conservative way, the following reflection conditions are selected to be bounding during handling, packaging, transfer and storage:

- 20 cm water shell for the storage cask,
- 50 cm steel shell for the transfer cask.

Table 6.4-8 Evaluation results for neutron reflection conditions

k-value for NCS model	FA no. 6 [redacted]	20Xg09mz04
variation	**Δk, pcm**	**calc. ID**
Reflection conditions for storage cask		
Void	-12	21oM02FU08
20 cm water	8	21VH02Jc08
50 cm concrete	-24	21CD02tz08
50 cm steel	-8	21uF02Le08

k-value for NCS model	FA no. 6 [redacted]	20Xg09mz04
variation	**Δk, pcm**	**calc. ID**
Reflection conditions for transfer cask		
Void	-65	21Qh02dP08
20 cm water	-75	21LU02hX08
50 cm concrete	-47	21WD02Cu08
50 cm steel	22	21ST02JB08

6.4.3 Criticality Results

6.4.3.1 Evaluation of Cask Arrays under Normal Conditions of Storage

6.4.3.1.1 Configuration

The evaluation model incorporates the following changes compared to the bounding model for NCS, described in section 6.3:

- infinite array of dry storage casks: array of dry densely packed storage casks with void between the casks.

According to the sensitivity analysis in subsection 6.4.2.1, the external moderation has no impact on the reactivity of the infinite array of fully flooded casks and, taking into account the low reactivity level of internally dry casks, does not need to be analyzed.

6.4.3.1.2 Results

The criticality calculations are performed for all analyzed FA. The results are summarized in Table 6.4-9 and include the maximum effective neutron multiplication factors k and the calculational bias with its uncertainty Δk_U. All contributions to the final results are determined with a 95 % probability at a 95 % confidence level, as discussed in subsection 6.5.2.1 and Appendix 6-1.

The results for the infinite array of dry storage casks show that the maximum effective neutron multiplication factors including the calculational bias with its uncertainty fulfil the acceptance criteria $k + \Delta k_U < 0.95$ for all investigated cask loadings and demonstrate the compliance with the requirements of § 72.124 and § 72.236. Moreover, the comparison with the results of criticality calculations using the bounding model for NCS provided in subsection 6.4.2.2 demonstrates that the reactivity of an infinite array of dry storage casks is bounded by the analysis of fully flooded casks.

Table 6.4-9 Results for infinite arrays of dry storage casks

FA no.	Case	Infinite array of dry storage casks	
	Internal moderation	0 % (Void)	
	External moderation and reflection	0 % (Void)	
	Fuel type	$k + \Delta k_U$	calc. ID
1	GE 8x8-1	0.35051	20EF09bR02
2	GE 8x8-2	0.34374	20hN09sh02
3	SPC 8x8-2	0.33861	20DB09dw02
4	GE9B 8x8	0.36399	20JJ09zw02
5	GE12 LUA	0.39113	20La09rj02
6	ATRIUM-10A	0.42457	20rk09gn02

6.4.3.2 Evaluation of Single Storage Cask

6.4.3.2.1 Configuration

The evaluation model incorporates the following changes compared to the bounding model for NCS, described in section 6.3:

- single, fully reflected storage cask: single, fully flooded storage cask surrounded with 20 cm water reflector.

As the water leakage into the canister cavity as well as any deformations of basket structures and content important to criticality safety under normal, off-normal and accident conditions of storage

are excluded, the configurations for a single storage cask under normal, off-normal and accident conditions of storage are bounded by the corresponding single, fully flooded and fully reflected storage cask, as described above.

6.4.3.2.2 Results

The criticality calculations are performed for all analyzed FA. The results are summarized in Table 6.4-10 and include the maximum effective neutron multiplication factors k and the calculational bias with its uncertainty Δk_U. All contributions to the final results are determined with a 95 % probability at a 95 % confidence level, as discussed in subsection 6.5.2.1 and Appendix 6-1.

The results show that the maximum effective neutron multiplication factors including the calculational bias with its uncertainty fulfil the acceptance criteria $k + \Delta k_U < 0.95$ for all investigated cask loadings and demonstrate the compliance with the requirements of § 72.124 and § 72.236.

Table 6.4-10 Results for a single, fully reflected storage cask

FA no.	Case	Single, fully reflected storage cask	
	Internal moderation	100 % (Water)	
	External moderation and reflection	100 % (Water)	
	Fuel type	$k + \Delta k_U$	calc. ID
1	GE 8x8-1	0.84387	20hr09qR02
2	GE 8x8-2	0.85034	20bM09JV02
3	SPC 8x8-2	0.84629	20AV09iY02
4	GE9B 8x8	0.88145	20ed09nP02
5	GE12 LUA	0.90527	20Np09QB02
6	ATRIUM-10A	0.93725	20Lf09eS02

6.4.3.3 Evaluation of Single Transfer Cask

6.4.3.3.1 Configuration

The evaluation model incorporates the following changes compared to the bounding model for NCS, described in section 6.3:

- single, fully reflected transfer cask: single, fully flooded transfer cask surrounded with 50 cm steel reflector.

As any deformations of basket structures and content important to criticality safety under normal, off-normal and accident conditions of storage are excluded, the configurations for a single transfer cask under normal, off-normal and accident conditions during handling, packaging and transfer operations are bounded by the corresponding single, fully flooded and fully reflected transfer cask, as described above.

6.4.3.3.2 Results

The criticality calculations are performed for all analyzed FA. The results are summarized in Table 6.4-11 and include the maximum effective neutron multiplication factors k and the calculational bias with its uncertainty Δk_U. All contributions to the final results are determined with a 95 % probability at a 95 % confidence level, as discussed in subsection 6.5.2.1 and Appendix 6-1.

The results show that the maximum effective neutron multiplication factors including the calculational bias with its uncertainty fulfil the acceptance criteria $k + \Delta k_U < 0.95$ for all investigated cask loadings and demonstrate the compliance with the requirements of § 72.124 and § 72.236.

Table 6.4-11 Results for a single, fully reflected transfer cask

FA no.	Case	Single, fully reflected transfer cask	
	Internal moderation	100 % (Water)	
	External moderation and reflection	100 % (Steel)	
	Fuel type	$k + \Delta k_U$	calc. ID
1	GE 8x8-1	0.84410	21Vu02YZ08
2	GE 8x8-2	0.85044	21lv02wr08
3	SPC 8x8-2	0.84641	21HQ02yr08
4	GE9B 8x8	0.88149	21oa02SS08
5	GE12 LUA	0.90541	21NW02dx08
6	ATRIUM-10A	0.93740	21jq02Sv08

List of References

[1] B. T. Rearden and M. A. Jessee, Eds.
SCALE Code System, ORNL/TM-2005/39, Version 6.2.2
Oak Ridge National Laboratory, Oak Ridge, Tennessee (2017)
Available from Radiation Safety Information Computational Center as CCC-834

6.5 Critical Benchmark Experiments

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

6.5.1 Benchmark Experiments and Applicability

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

c_k

Figure 6.5-1 Frequency distribution of correlation coefficients between the application and selected benchmarks

6.5.2 Results of the Benchmark Calculations

6.5.2.1 Bias Determination

[redacted]

[redacted] (5.1)

[redacted]

[redacted] (5.2)

[redacted]

[redacted]

[redacted] (5.3)

[redacted]

[redacted] (5.4)

[redacted]

[redacted]

[redacted] (5.5)

[redacted]

[redacted] (5.6)

[redacted]

[redacted] (5.7)

(5.8)

[redacted]

[redacted] (5.9)

[redacted]

[redacted]

[redacted] (5.10)

(5.11)

(5.12)

6.5.2.2 Validation Results

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Table 6.5-1 Results of benchmark calculations

[redacted]

Table 6.5-1 Results of benchmark calculations (continued)

[redacted]

List of References

[1] International Handbook of Evaluated Criticality Safety Benchmark Experiments
NEA Nuclear Science Committee
September 2019 Edition, NEA/NSC/DOC(95)03

6.6 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

Appendix 6-1: Methodological uncertainty

Appendix 6-2: Sample Input File for Bounding Model for NCS (Calc. ID: 20nS09hZ02)

Methodological uncertainty

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted] σ

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Table 1 Results for the evaluation of single calculations

[redacted]

Table 1 Results for the evaluation of single calculations (continued)

[redacted]

Table 2 Applied random numbers

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

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7 Containment

7.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

In this chapter the compliance of the storage cask CASTOR® geo69 regarding the containment requirements from 10 CFR 72 is shown and the containment boundary is described. The storage cask is loaded with the content described in Appendix 7-1 and based on subsection 1.2.3.

7.1 Containment Boundary

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

The cask CASTOR® geo69 is designed for FA with moderate burn-up as well as high burn-up fuel with an averaged burn-up above 45 GWd/Mg_{HM} (cf. Section 1.2.3). A double (inner and outer) containment as described in the following subsections in more detail is used.

7.1.1 Containment Vessel

The CASTOR® geo69 containment vessel is constituted by the following subassemblies:

a) Inner containment (canister lid, with item numbers from [1])
   - canister body (Item 2),
   - canister lid (Item 3) and clamping elements (Item 4), thread bolts (Item 6) and metal gasket (Item 16 (Ag) with a torus diameter of [redacted]), as well as
   - tightening plug (Item 10) in the canister lid and metal gasket (Item 13 (Ag) with a torus diameter of [redacted]).

b) Outer containment (cask lid, with Item numbers from [2])
   - cask body (Item 2),
   - cask lid (Item 55) and hexagonal screws (Item 62), hexagonal head screws for sealing (Item 63) and metal gasket (Item 69 (Ag) with a torus diameter of [redacted]),
   - protection cap (Item 113) in the cask lid, cap screws (Item 37) and metal gasket (Item 44 (Ag) with a torus diameter of [redacted]), as well as
   - pressure switch in the cask lid (see [3]), cap screws and metal gasket (with a torus diameter of [redacted]).

The outer jackets of all metal gaskets of the containment system are made of silver (Ag). The canister body (Item 2 from [1]) is designed by welding Items 2-2 to 2-5 from [2] together.

The inner and outer containment boundaries are shown in Figure 7.1-1.

[redacted]

Figure 7.1-1: Inner and outer containment (canister and cask)

7.1.2 Containment Penetrations

As described in Section 7.1.1, there is one penetration through the lid of the inner containment and two penetrations through the lid of the outer containment.

The penetration through the inner containment is shown in more detail at the leakage path in Figure 7.1-2. The penetrations through the outer containment are shown in more detail at the leakage paths in Figure 7.1-3 and Figure 7.1-4.

[redacted]

Figure 7.1-2: Containment boundary detail at the canister lid with tightening plug

[redacted]

Figure 7.1-3: Containment boundary detail at the cask lid with protection cap

[redacted]

Figure 7.1-4: Containment boundary detail at the cask lid with pressure switch

7.1.3 Seals and Welds

It should be noted that the monolithic cask body and the lids can be considered as leak-tight, so the containment analysis can be reduced to the gasket sealing system.

The potential leakage paths of the inner and outer containment are shown in Figure 7.1-5.

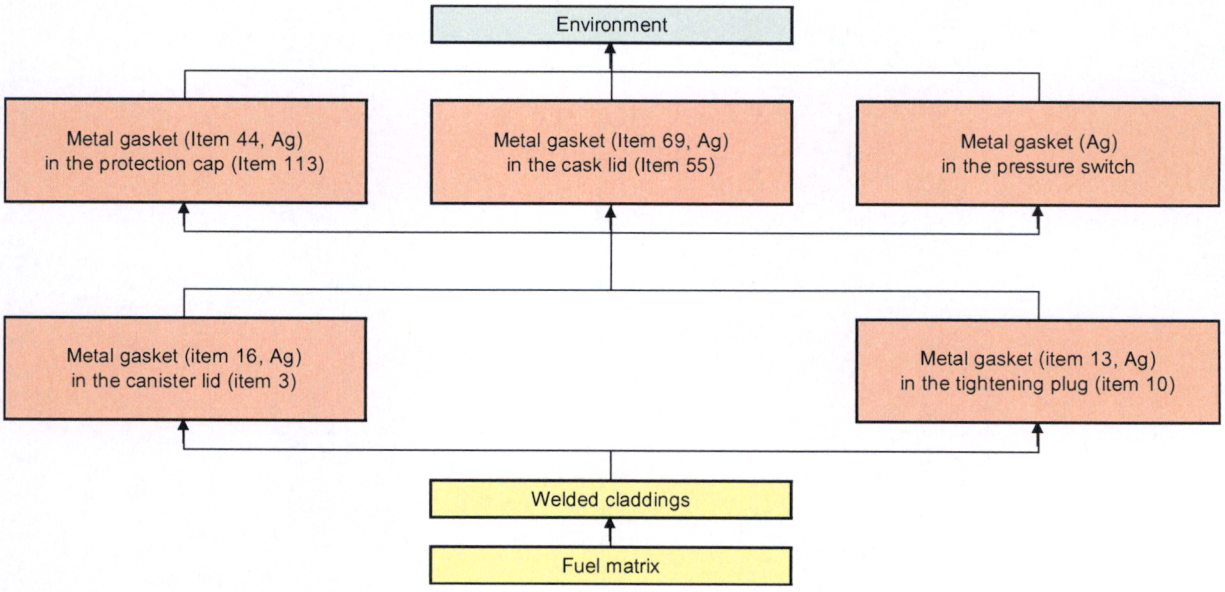

Figure 7.1-5: Leakage paths of the inner and outer containment

The sealing effect is the result of the sealing function of the metal gaskets employed. Each metal gasket consists of a helical spring made of Nimonic® surrounded by an inner jacket of stainless steel and an outer jacket of silver.

The sealing effect of a metal gasket is based on the plastic deformation of the outer jacket, which is the result of the pretension force induced by the screwed connection of the lid. The ductility is larger for the outer jacket of the metal gasket than for the inner jacket so that the gasket will adapt to the surface structure of the sealing surface. The function of the inner jacket is to distribute uniformly the force due to pressure that is generated during the compression of the helical spring over the outer jacket. For metal gaskets, capillary leakage is the only relevant potential leakage mechanism and continuous venting is precluded.

For each containment, a maximum standard helium leakage rate of [redacted] (leak test criterion) is proven by measurement after loading (see Chapter 9). This corresponds to a reference air leakage rate of [redacted]

7.1.4 Closure

The SNF is double contained by the canister and by CASTOR® geo69 cask. As described above, each containment and its penetrations are equipped with a closure consisting of a lid and a metal gasket. The canister lid is attached to the canister body with [redacted] thread bolts (Item 6 in [1]). The cask lid is attached to the cask body by [redacted] hexagonal screws (Item 62 in [2]) and [redacted] hexagon head screws for sealing (Item 63 in [2]).

This double containment is designed to maintain containment integrity during normal and off-normal conditions of storage as well as for hypothetical accident conditions.

List of References

[1] 1014-DPL-36855, Rev. 0, Design Parts List CASTOR® geo69, Canister

[2] 1014-DPL-30934, Rev. 0, Design Parts List CASTOR® geo69, Cask

[3] 1014-DD-38566, Rev. 0, Design Drawing Storage Configuration CASTOR® geo69

7.2 Containment Requirements for Normal Conditions of Storage

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

7.2.1 Release of Radioactive Material

As the structural integrity and the redundant containments are not impaired in NCS (cf. Chapter 3), the design leakage rate of the considered containment is not greater than 10^{-7} ref·cm³/s (leak-tight according to ANSI N14.5 [1]). Therefore, no dedicated activity release calculations and corresponding dose calculations as described in [2] are required.

Nevertheless, the activity mobilization inside the canister is given as described in Appendix 7-3. Starting with the activity content from Table 7.6-1, using the release fractions f_B, f_G, f_V, f_F and f_C for normal conditions from Table 7.6-3 and using the value for the free gas volume V inside the canister from Table 7.2-4, the activity mobilization inside the cask is shown in Table 7.2-1 for gases and volatiles, in Table 7.2-2 for fines and crud and in Table 7.2-3 summed up for the nuclide mixture for NCS. For nuclides that may be available as gases or volatiles, an additional contribution as fines is taken into account.

Table 7.2-1: Mobilized activity and activity concentration for gases and volatiles for NCS

[redacted]

Table 7.2-2: Mobilized activity and activity concentration for fines and crud for NCS

[redacted]

Table 7.2-3: Mobilized activity and activity concentration summed up for the nuclide mixture for NCS

[redacted]

7.2.2 Pressurization of Containment Vessel

The canister inside the CASTOR® geo69 cask contains SNF during NCS. The interior space inside the canister is drained, dried, evacuated and backfilled with helium gas prior to final closure of the canister. The dry interior space inside the cask with a loaded canister is evacuated and backfilled with helium gas prior to final closure of the cask. Therefore, no vapors or gases are present which could cause a reaction or explosion inside the canister and the cask. Procedural steps ensure a maximum absolute pressure of $p_{He,0}$ (cf. Appendix 7-2) inside the canister and $p_{He,cask,0}$ (cf. Appendix 7-2) inside the cask.

With the procedure described in Appendix 7-2, the maximum internal pressures for NCS are calculated as follows. Structural integrity and containment of the canister and the storage cask are not impaired in NCS (cf. Chapter 3) and both containment barriers remain leak-tight. Therefore, the pressures are calculated for the canister and the cask separately. There are no combustible gases inside the containment.

The maximum absolute pressure p_u = [redacted] inside the canister assuming no fuel rod failure is obtained with $p_u = p_{He,0} \cdot T_{gas} / T_{He,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (covering NCS value for the filling gas of the canister in Section 4.4); resp. p_u = [redacted] inside the canister assuming fuel rod failure with the boundary conditions given in Table 7.2-4.

The maximum absolute pressure p_u = [redacted] inside the cask is obtained with $p_u = p_{He,cask,0} \cdot T_{gas} / T_{He,cask,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (covering NCS value for the filling gas of the cask in Section 4.4).

The maximum normal operating pressure (MNOP) is the value of the upstream absolute pressure p_u for NCS, reduced by the atmospheric pressure at mean sea level, i. e. 101.3 kPa. With a maximum absolute pressure p_u = [redacted] for the MNOP.

Table 7.2-4: Boundary conditions for canister pressure calculation (NCS) assuming fuel rod failure

Parameter	Symbol	Value	Reference
Initial canister and cask filling gas pressure	$p_{He,0}$	[redacted]	Appendix 7-2
Initial canister and cask filling gas temperature	$T_{He,0}$	[redacted]	Appendix 7-2
Gas temperature (covering value from canister gas mixture)	T_{gas}	[redacted]	Section 4.8
Fuel rod failure fraction	f_B	[redacted]	Table 7.6-3
Fission gas release fraction	f_G	[redacted]	Table 7.6-3
Maximum produced amount of fission gas per loading	G_{FG}	[redacted]	Appendix 7-1
Amount of mobilized fission gas in the canister ($G_{FG} \cdot f_B \cdot f_G$)	n_{FG}	[redacted]	Calc.
Maximum amount of fuel rod filling gas helium per loading	G_{FR}	[redacted]	Appendix 7-1
Amount of mobilized fuel rod filling gas helium ($G_{FR} \cdot f_B$)	n_{FR}	[redacted]	Calc.
Sum of gas released from the content into the canister ($n_{FG} + n_{FR}$)	n	[redacted]	Calc.
Minimum free gas volume inside the canister	V	[redacted]	Chapter 1

List of References

[1] ANSI N14.5-2014, American National Standard
For Radioactive Materials – Leakage Tests on Packages for Shipment

[2] NUREG-2215, April 2020
Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities

7.3 Containment Requirements for Off-Normal Conditions of Storage

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

7.3.1 Release of Radioactive Material

As the structural integrity and the redundant containments are not impaired for off-normal conditions (cf. Chapter 3), the design leakage rate of the considered containment is not greater than 10^{-7} ref·cm³/s (leak-tight according to [1]). Therefore, no dedicated activity release calculations and corresponding dose calculations as described in [2] are required.

Nevertheless, the activity mobilization inside the canister is given as described in Appendix 7-3. Starting with the activity content from Table 7.6-1, using the release fractions f_B, f_G, f_V, f_F and f_C for off-normal conditions from Table 7.6-3 and using the value for the free gas volume V inside the canister from Table 7.3-4, the activity mobilization inside the cask is shown in Table 7.3-1 for gases and volatiles, in Table 7.3-2 for fines and crud and in Table 7.3-3 summed up for the nuclide mixture for off-normal conditions. For nuclides that may be available as gases or volatiles, an additional contribution as fines is taken into account.

Table 7.3-1: Mobilized activity and activity concentration for gases and volatiles for off-normal conditions

[redacted]

Table 7.3-2: Mobilized activity and activity concentration for fines and crud for off-normal conditions

[redacted]

Table 7.3-3: Mobilized activity and activity concentration summed up for the nuclide mixture for off-normal conditions

[redacted]

The mobilized content and the mobilized activity concentration in the canister increases for those mobility types where the release fraction f_B is used (gases, volatiles and fines) compared to NCS by the factor of 10.

7.3.2 Pressurization of Containment Vessel

The canister inside the CASTOR® geo69 cask contains SNF during off-normal conditions of storage. The interior space inside the canister is drained, dried, evacuated and backfilled with helium gas prior to final closure of the canister. The dry interior space inside the cask with a loaded canister is evacuated and backfilled with helium gas prior to final closure of the cask. Therefore, no vapors or gases are present which could cause a reaction or explosion inside the canister and the cask. Procedural steps ensure a maximum absolute pressure of $p_{He,0}$ (cf. Appendix 7-2) inside the canister and $p_{He,cask,0}$ (cf. Appendix 7-2) inside the cask.

With the procedure described in Appendix 7-2, the maximum internal pressures for off-normal condition are calculated as follows. Structural integrity and containment of the canister and the storage cask are not impaired for off-normal conditions (cf. Chapter 3) and both containment barriers remain leak-tight. Therefore, the pressures are calculated for the canister and the cask separately. There are no combustible gases inside the containment.

The maximum absolute pressure p_U = [redacted] inside the canister assuming fuel rod failure results with the boundary conditions given in Table 7.3-4.

The maximum absolute pressure p_u = [redacted] inside the cask, obtained with $p_u = p_{He,cask0} \cdot T_{gas} / T_{He,cask,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (as in section 7.2.2), is unchanged to NCS.

Table 7.3-4: Boundary conditions for canister pressure calculation (off-normal conditions) assuming fuel rod failure

Parameter	Symbol	Value	Reference
Initial canister and cask filling gas pressure	$p_{He,0}$	[redacted]	Appendix 7-2
Initial canister and cask filling gas temperature	$T_{He,0}$	[redacted]	Appendix 7-2
Gas temperature (covering value from canister gas mixture)	T_{gas}	[redacted]	Section 4.8
Fuel rod failure fraction	f_B	[redacted]	Table 7.6-3
Fission gas release fraction	f_G	[redacted]	Table 7.6-3
Maximum produced amount of fission gas per loading	G_{FG}	[redacted]	Appendix 7-1
Amount of mobilized fission gas in the canister ($G_{FG} \cdot f_B \cdot f_G$)	n_{FG}	[redacted]	Calc.
Maximum amount of fuel rod filling gas helium per loading	G_{FR}	[redacted]	Appendix 7-1
Amount of mobilized fuel rod filling gas helium ($G_{FR} \cdot f_B$)	n_{FR}	[redacted]	Calc.
Sum of gas released from the content into the canister ($n_{FG} + n_{FR}$)	n	[redacted]	Calc.
Minimum free gas volume inside the canister	V	[redacted]	Chapter 1

List of References

[1] ANSI N14.5-2014, American National Standard
For Radioactive Materials – Leakage Tests on Packages for Shipment

[2] NUREG-2215, April 2020
Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities

7.4 Containment Requirements for Hypothetical Accident Conditions

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

7.4.1 Fission Gas Products

For the cask content compliant with the stipulations in Section 1.2.3, the maximum produced amount of fission gas of G_{FG} = [redacted] is obtained (cf. Appendix 7-1). This amount corresponds to a cask loading of 69 UOX FA with maximum values for the final discharge burn-up of [redacted] and the heavy metal mass of [redacted] each. Therefore, all possible cask loadings are covered by the considered amount of fission gas.

For each fuel rod that is assumed to have failed, the fraction f_G (cf. Table 7.6-3) of the produced fission gas is assumed to be released into the cavity of the canister. For hypothetical accident conditions all fuel rods are assumed to have failed (f_B = 1.0, cf. Table 7.6-3).

According to [1], there are two separate cases to be analyzed for hypothetical accident conditions: accident fire conditions (HAC-fire) and accident impact conditions (HAC-impact).

According to [1], the fraction of fission gas release f_G is used as 0.15 (15 %) for HAC-fire as well as 0.35 (35 %) for HAC-impact which includes an extra 20 % fraction of the pellet-retained fission gases that might be released during a drop impact. This results in a maximum mobilized amount of fission gas inside the canister of $n_{FG} = G_{FG} \cdot f_B \cdot f_G$, which equals [redacted] for HAC-fire and [redacted] for HAC-impact.

The fission gas Xenon provides about [redacted] of this amount (cf. Table 7.6-2). The distribution of all considered fission gases is listed as values in parenthesis in Table 7.6-2 .

7.4.2 Release of Content

As the structural integrity and the redundant containments are not impaired for HAC-fire and HAC-impact (cf. Chapter 3), the design leakage rate of the considered containment is not greater than 10^{-7} ref·cm³/s (leak-tight according to [2]). Therefore, no dedicated activity release calculations and corresponding dose calculations as described in [3] are required.

Nevertheless, the activity mobilization inside the canister is given as described in Appendix 7-3. Starting with the activity content from Table 7.6-1, using the release fractions f_B, f_G, f_V, f_F and f_C for HAC-fire resp. HAC-impact from Table 7.6-3 and using the value for the free gas volume V inside the canister from Table 7.4-4 resp. Table 7.4-5, the activity mobilization inside the cask is shown in Table 7.4-1 for gases and volatiles, in Table 7.4-2 for fines and crud and in Table 7.4-3 summed up for the nuclide mixture for HAC-fire and HAC-impact. For nuclides that may be available as gases or volatiles, an additional contribution as fines is taken into account.

Table 7.4-1: Mobilized activity and activity concentration for gases and volatiles for HAC-fire and HAC-impact

[redacted]

Table 7.4-2: Mobilized activity and activity concentration for fines and crud for HAC-fire and HAC-impact

[redacted]

Table 7.4-3: Mobilized activity and activity concentration summed up for the nuclide mixture for HAC-fire and HAC-impact

[redacted]

The mobilized content and the mobilized activity concentration in the canister is about the same for HAC-fire and HAC-impact but differs in the contribution of gases and fines. This corresponds to the different release fractions listed in Table 7.6-3.

7.4.3 Pressurization of Containment Vessel

The canister inside the CASTOR® geo69 cask contains SNF during HAC-fire and HAC-impact. The interior space inside the canister is drained, dried, evacuated and backfilled with helium gas prior to final closure of the canister. The dry interior space inside the cask with a loaded canister is evacuated and backfilled with helium gas prior to final closure of the cask. Therefore, no vapors or gases are present which could cause a reaction or explosion inside the canister and the cask. Procedural steps ensure a maximum absolute pressure of $p_{He,0}$ (cf. Appendix 7-2) inside the canister and $p_{He,cask,0}$ (cf. Appendix 7-2) inside the cask.

With the procedure described in Appendix 7-2, the maximum internal pressures for HAC-fire and HAC-impact are calculated as follows. Structural integrity and containment of the canister and the storage cask are not impaired for HAC-fire and HAC-impact (cf. Chapter 3) and both containment barriers remain leak-tight. Therefore, the pressures are calculated for the canister and the cask separately. There are no combustible gases inside the containment.

<u>Accident-Fire Conditions:</u>

For HAC-fire the maximum absolute pressure p_u = [redacted] inside the canister assuming no fuel rod failure is obtained with $p_u = p_{He,0} \cdot T_{gas} / T_{He,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (covering NCS value for the filling gas of the canister in Section 4.6); resp. p_u = [redacted] inside the canister assuming fuel rod failure with the boundary conditions given in Table 7.4-4.

The maximum absolute pressure p_u = [redacted] inside the cask is obtained with $p_u = p_{He,cask,0} \cdot T_{gas} / T_{He,cask,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (covering HAC-fire value for the filling gas of the cask in Section 4.4).

Table 7.4-4: Boundary conditions for canister pressure calculation (HAC-fire) assuming fuel rod failure

Parameter	Symbol	Value	Reference
Initial canister and cask filling gas pressure	$p_{He,0}$	[redacted]	Appendix 7-2
Initial canister and cask filling gas temperature	$T_{He,0}$	[redacted]	Appendix 7-2
Gas temperature (covering value from canister gas mixture)	T_{gas}	[redacted]	Section 4.8
Fuel rod failure fraction	f_B	[redacted]	Table 7.6-3
Fission gas release fraction	f_G	[redacted]	Table 7.6-3
Maximum produced amount of fission gas per loading	G_{FG}	[redacted]	Appendix 7-1
Amount of mobilized fission gas in the canister ($G_{FG} \cdot f_B \cdot f_G$)	n_{FG}	[redacted]	Calc.
Maximum amount of fuel rod filling gas helium per loading	G_{FR}	[redacted]	Appendix 7-1
Amount of mobilized fuel rod filling gas helium ($G_{FR} \cdot f_B$)	n_{FR}	[redacted]	Calc.
Sum of gas released from the content into the canister ($n_{FG} + n_{FR}$)	n	[redacted]	Calc.
Minimum free gas volume inside the canister	V	[redacted]	Chapter 1

Accident-Impact Conditions:

For HAC-impact the maximum absolute pressure p_u = [redacted] inside the canister results with the boundary conditions given in Table 7.4-5.

The maximum absolute pressure p_u = [redacted] inside the cask for HAC-impact is obtained with $p_u = p_{He,cask,0} \cdot T_{gas} / T_{He,cask,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (covering HAC-impact value for the filling gas of the cask in Section 4.8).

Table 7.4-5: Boundary conditions for canister pressure calculation (HAC-impact)

Parameter	Symbol	Value	Reference
Initial canister and cask filling gas pressure	$p_{He,0}$	[redacted]	Appendix 7-2
Initial canister and cask filling gas temperature	$T_{He,0}$	[redacted]	Appendix 7-2
Gas temperature (covering value from canister gas mixture)	T_{gas}	[redacted]	Section 4.8
Fuel rod failure fraction	f_B	[redacted]	Table 7.6-3
Fission gas release fraction	f_G	[redacted]	Table 7.6-3
Maximum produced amount of fission gas per loading	G_{FG}	[redacted]	Appendix 7-1
Amount of mobilized fission gas in the canister ($G_{FG} \cdot f_B \cdot f_G$)	n_{FG}	[redacted]	Calc.
Maximum amount of fuel rod filling gas helium per loading	G_{FR}	[redacted]	Appendix 7-1
Amount of mobilized fuel rod filling gas helium ($G_{FR} \cdot f_B$)	n_{FR}	[redacted]	Calc.
Sum of gas released from the content into the canister ($n_{FG} + n_{FR}$)	n	[redacted]	Calc.
Minimum free gas volume inside the canister	V	[redacted]	Chapter 1

List of References

[1] NUREG-2224, November 2020
Dry Storage and Transportation of High Burnup Spent Nuclear Fuel

[2] ANSI N14.5-2014, American National Standard
For Radioactive Materials – Leakage Tests on Packages for Shipment

[3] NUREG-2215, April 2020
Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities

7.5 Containment Requirements for Short-Term Operations

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

7.5.1 Pressurization of Containment Vessel

The canister inside the transfer cask contains SNF during handling of the transfer cask inside the reactor building. The interior space inside the canister is drained, dried, evacuated and backfilled with helium gas prior to final closure of the canister.

The maximum absolute pressure p_u = [redacted] inside the canister assuming no fuel rod failure is obtained with $p_u = p_{He,0} \cdot T_{gas} / T_{He,0}$ (cf. Appendix 7-2) and T_{gas} = [redacted] (covering value for the filling gas of the canister in Section 4.7).

7.6 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

Appendix 7-1 Content

Appendix 7-2 Determination of design pressure values

Appendix 7-3 Activity mobilization

Appendix 7-4 Assumptions

Appendix 7-1 Content

The activity content of the cask is described in Section 1.2.3. The spent uranium oxide (UOX) FA have a maximum heavy metal mass of [redacted] and a maximum FA-averaged final discharge burn-up of [redacted]. Based on the activity values per FA given in Section 1.2.3 the total activity content per cask loading with 69 FA is determined by multiplying the maximum value per nuclide over all FA types (FA No 1 to FA No 6 given in Section 1.2.3) with the number of FA per cask loading. In addition to Section 1.2.3 the nuclide activity of ^{129}I is taken into account. These covering values are listed in Table 7.6-1.

Table 7.6-1: Activity content of a cask loading with 69 FA

[redacted]

For crud depositions, additional activity is taken into account. The surface specific crud activity is taken as $4.64 \cdot 10^7$ Bq/cm^2 ($1254 \cdot 10^{-6}$ Ci/cm^2, see [1]). As ^{60}Co is the only significant contributor to crud activity after short cooling of the FA, the whole crud activity is assumed as ^{60}Co (also see [1]). Taking the fuel types in Section 1.2.3 into account, a conservative value for the crud activity of [redacted] is estimated for a cask loading with 69 FA (k_{FA}) by the following calculation: [redacted] With this conservative assessment, potential residual contamination on the inside surfaces of the package

(e. g. from previous transports) is covered. According to [1], the decay of ^{60}Co is considered by the minimum time before loading of [redacted] (cf. Section 1.2.3) and results in a crud activity of [redacted].

The fission gas production is in very good approximation linearly dependent on the produced energy during reactor operation. The maximum fission gas masses ([redacted]) are taken from Section 1.2.3 and summarized in Table 7.6-2 as specific gas production values for each fission gas (in unit mol/GWd) for UOX FA. [redacted] also given in Table 7.6-2.

Table 7.6-2: Specific production of fission gas

[redacted]

The total produced amount of fission gas G_{FG} in a loading is calculated via:

[redacted]

For the cask content compliant to the stipulations in Section 1.2.3, the maximum produced amount of fission gas is obtained for 69 UOX FA with a final discharge burn-up of [redacted] and a heavy metal mass of [redacted]. In this case, G_{FG} = [redacted] is obtained.

The fuel rod filling gas helium is considered for the FA No 5 (GE 12) given in Section 1.2.3 as a covering amount for the other FA types. The gas volume in a fuel rod of [redacted] and the number of fuel rods [redacted] are given in Section 1.2.3. Higher values for the gas volume in a fuel rod of up to

[redacted] for other FA No in Section 1.2.3 result to a lower total gas volume for a FA because of lower numbers of FR per FA for those FA types. [redacted] a total amount of fuel rod filling gas for a cask loading with 69 FA is calculated to [redacted] by using the ideal gas law and set to G_{FR} = [redacted] for the design pressure calculations in Sections 7.2, 7.3 and 7.4.

List of References

[1] NUREG-2215, April 2020
Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities

Appendix 7-2 Determination of design pressure values

Information on the maximum pressure p_u inside the canister and cask is required for various analyses.

The absolute pressure for the canister is obtained by using the ideal gas law $p_u = n \cdot R_{univ} \cdot T_{gas} / V$, where n is the amount of gas, R_{univ} is the universal gas constant 8.314 J/mol/K, T_{gas} is the absolute gas temperature (volume average) and V is the free gas volume inside the canister. In addition, the assumed maximum helium filling partial pressure of the canister is temperature-corrected by the gas temperature T_{gas} under test conditions. Therefore, p_u for the canister is calculated with:

$$p_u = p_{He,0} \cdot T_{gas} / T_{He,0} + n \cdot R_{univ} \cdot T_{gas} / V.$$

Procedural steps ensure a maximum absolute pressure of $p_{He,0}$ = [redacted] cf. Chapter 9) inside the canister.

To determine the maximum absolute pressure p_u, the values of n and T_{gas} are maximized while V is minimized. Therefore, the influence parameters are estimated as follows:

- maximum amount of gas n:
  All relevant gas contributions have to be added. This includes the maximum amount of gas released from the content, i. e. fission gas and filling gas of the fuel rods. How this value is deduced is explained below.
- maximum absolute gas temperature T_{gas}:
  In the context of the thermal design calculations in Chapter 4, the maximum volume averaged gas temperature is calculated for various test conditions.
- minimum free gas volume V:
  The free gas volume is calculated based on the canister design. From the canister cavity volume, the displacement volumes of the basket and the fuel assemblies are subtracted. The free gas volume inside fuel rods that are considered to have failed is not included in the total free gas volume.

For maximum pressure considerations, the gas release from the content is calculated as follows:

- The fraction of failed fuel rods f_B is assumed as 0.01 (1 %) for NCS, 0.1 (10 %) for off-normal conditions and as 1.0 (100 %) for HAC-impact and HAC-fire (see Table 7.6-3, according to [1]).
- The maximum total amount of filling gas of the fuel rods is determined based on the information provided for the fuel rods. For each fuel rod that is assumed to have failed, the full amount of filling gas is assumed to be released into the canister.

- The produced amount of fission gas is determined via burn-up calculations (cf. Appendix 7-1). [redacted]
- According to [1], the fraction of fission gas release f_G is used as 0.15 (15 %) for NCS, off-normal and HAC-fire as well as 0.35 (35 %) for HAC-impact which includes an extra 20 % fraction of the pellet-retained fission gases that may be released during a drop impact (see Table 7.6-3). For each fuel rod that is assumed to have failed, the fraction f_G of the produced fission gas is assumed to be released into the cavity of the canister.

After the canister drying process, no residual water has to be assumed to be present as vapor after dispatch. Further gases are not formed during operation of the storage cask, either.

Procedural steps ensure a maximum absolute filling pressure of $p_{He,cask,0}$ = [redacted] (cf. Chapter 9) inside the cask. With a free cask volume of [redacted] (cf. Chapter 1) a total amount of Helium in the cask results to about [redacted] The assumed maximum helium filling partial pressure of the cask is temperature-corrected by the gas temperature in the cask $T_{gas,cask}$ under test conditions. Therefore, the absolute pressure p_u for the cask is calculated with:

$$p_u = p_{He,cask,0} \cdot T_{gas,cask} / T_{He,cask,0}.$$

The moderator disc between the canister lid and the cask lid might cause an additional amount of radiolysis gas from irradiation. The energy dose from gamma irradiation of the lid-end moderator disc, which is made of the ultrahigh molecular weight polyethylene [redacted], is given as about [redacted] in Chapter 5. Taking a G-value for hydrogen of 4 molecules per 100 eV for polyethylene (with ultrahigh molecular weight) into account (cf. [2]), a negligible amount of [redacted] results in a year.

No residual water has to be assumed to be present as vapor after dispatch. Further gases are not formed during operation of the storage cask, either.

<u>Remark:</u> The thermal evaluations in Chapter 4 provide two sets of temperature distributions for each, NCS and HAC-fire. The first set is based on calculations with an inert helium atmosphere inside the canister without assuming fission gases after fuel rod failure. For this calculation the maximum thermal load of the storage cask is used and represents the analysis up to 20 years of dry storage (cf. [1]). The other set in Section 4.8 is based on calculations assuming fission gases and represents with a reduced decay heat power the analysis > 20 years of dry storage (cf. [1]).

List of References

[1] NUREG-2224, November 2020
Dry Storage and Transportation of High Burnup Spent Nuclear Fuel

[2] AMEC/200615/001 Issue 3,
Determination of G-values for use in SMOGG gas generation calculations

Appendix 7-3 Activity mobilization

The activity content is classified in four categories, in line with the approach in [1]. The categories are gaseous substances, volatile substances, particulate substances from fuel and particulate substances from crud. Based on [1], the nuclides are classified as follows: ^{3}H, ^{85}K and ^{129}I as gases, ^{89}Sr, ^{90}Sr, ^{106}Ru, ^{134}Cs and ^{137}Cs as volatiles, all nuclides (except from ^{60}Co) from Table 7.6-1 as particulate substances from fuel and ^{60}Co as particulate substance from crud (cf. Appendix 7-1).

In Appendix 7-2, the fraction of failed fuel rods f_B and the fraction of fission gas release f_G are introduced. According to [1],

- the fraction of volatiles that are released due to a cladding breach is $f_V = 3 \cdot 10^{-5}$ for NCS, off-normal, HAC-impact and HAC-fire,
- the mass fraction of fuel that is released as fines due to a cladding breach is $f_F = 3 \cdot 10^{-5}$ for NCS, off-normal, HAC-impact and an increased mass fraction of $f_F = 3 \cdot 10^{-3}$ due to a conservatively assumed fuel oxidation for HAC-fire and
- the fraction of crud that spalls off rods is $f_C = 0.15$ for NCS and off-normal conditions and $f_C = 1.0$ for HAC-impact and HAC-fire.

The release fractions of radioactive materials that are considered in this analysis are summarized in Table 7.6-3.

Table 7.6-3: Release fractions of radioactive materials

Variable	Symbol	Normal Conditions (NCS)	Off-Normal Conditions (off-normal)	Accident-Fire Conditions (HAC-fire)	Accident-Impact Conditions (HAC-impact)
Fraction of Fuel Rods Assumed To Fail	f_B	0.01	0.1	1.0	1.0
Fraction of Fission Gases Released Due to a Cladding Breach	f_G	0.15	0.15	0.15	0.35
Fraction of Volatiles Released Due to a Cladding Breach	f_V	3E-05	3E-05	3E-05	3E-05
Mass Fraction of Fuel Released as Fines Due to a Cladding Breach	f_F	3E-05	3E-05	3E-03	3E-05
Fraction of Crud Spalling off Cladding	f_C	0.15	0.15	1.0	1.0

The activity concentration of the gases ^{3}H, ^{85}Kr and ^{129}I is obtained by multiplying the total activity of the concerned nuclides from Table 7.6-1 with the fraction of failed fuel rods f_B and the fraction of fission gas release f_G and dividing the result by the free gas volume V. In the same way but using the fraction of failed fuel rods f_B and the fraction f_V resp. f_F the activity concentrations for volatiles resp. particulate substances from fuel (fines) are defined. For the activity concentration of crud only the fraction f_C is taken into account.

List of References

[1] NUREG-2224, November 2020
Dry Storage and Transportation of High Burnup Spent Nuclear Fuel

Appendix 7-4 Assumptions

- The crud activity of [redacted]) is estimated for a cask loading (cf. Appendix 7-1).
- An initial absolute fuel rod filling gas pressure of [redacted] is assumed for all fuel rods of all FA in the cask loading (cf. Appendix 7-1).
- Residual water vapor is excluded in the calculations regarding design pressure values (cf. Appendix 7-2).

8 Materials Evaluation

8.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

The materials evaluation presented in this chapter ensures that materials will perform in a manner that supports the functions of the SSCs of the CASTOR® geo69 DSS and the CLU. This chapter provides information on materials of construction, including mechanical properties, thermal properties and the technical basis for material properties. The material properties summarized in this chapter form the basis for the structural, thermal, shielding, criticality and containment evaluation of the DSS and the CLU, if applicable.

Section 8.3 demonstrates that the materials will not undergo adverse environmental degradation, chemical reactions, or other reactions that could challenge the ability of SSCs to safety handle, package, transfer, and store the SNF over the intended storage period.

8.1 System Design

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

8.1.1 Drawings

The principle design of the CASTOR® geo69 DSS and the CLU is described in Section 1.2. Drawings and corresponding parts lists of the DSS and CLU components are included in Section 1.5. The parts lists and drawings include material specifications, alternatives, welding instructions and non-destructive examination (NDE) requirements.

8.1.2 Codes and Standards

The codes and standards applicable for the CASTOR® geo69 DSS design are listed in Table 8.1-1. The containment system is designed in accordance with Division 3, Subsection WC, considering the underlying requirements with respect to mechanical, design, material and fabrication issues. Seals and gaskets that are part of the containment system are leak tested in accordance with ANSI N14.5 [1]. Special lifting devices are designed in accordance with ANSI N14.6 [2]. For materials not covered by the BPVC, material data according to the manufacturer's catalogue and test data are used.

Table 8.1-1: Applicable codes and standards for the design of the CASTOR® geo69 DSS

Component	*Applicable Codes and Standards for Design*
Containment system with bottom/closure plate	Division 3, Subsection WC
Trunnions (storage cask) LAP of the canister lid LAP of the cask lid	ANSI N14.6 [2] as applicable Industry standards
Seals and gaskets	ANSI N14.5 [1] as applicable Material data according to manufacturer's catalogue Test data
Moderator	Material data according to manufacturer's catalogue
Fuel basket without shielding elements	GNS proprietary design methodology report Material data according to manufacturer's catalogue Test data
Shielding elements	Respective material standards including additional manufacturer's catalogue, as applicable
Protection cover	Industry standards

The CLU is designed in accordance with the applicable requirements for supports used during the handling of SNF. The structural design of the transfer cask and transfer lock follows the requirements according to the ASME BPVC, Section III, Division 1, Subsection NF [3]. The code is considered regarding the underlying requirements with respect to mechanical, design, material and fabrication issues. The applicable codes and standards for the design of the transfer cask and the transfer lock are listed in Table 8.1-2.

Table 8.1-2: Applicable codes and standards for the design of the CLU

Component	*Applicable Codes and Standards for Design*
Structural components of transfer cask and transfer lock	ASME BPVC Section III, Division 1, Subsection NF [3]
Trunnions of the Transfer Cask Load attachment points of the transfer lock	ANSI N14.6 [2] as applicable Industry standards
Lead shield	PB940R acc. DIN EN 12659 [ASTM B-29]

8.1.3 Welding

8.1.3.1 Fabrication and examination of welds in the DSS

Containment welds of category A, B and C according to Division 3, Subsection WC-3251 exist in the canister of the DSS. The canister body consists of a welded stainless steel construction The welding fabrication specifications included in the drawing of the canister (see Section 1.5) and the acceptance criteria for containment welds, as specified in Section 10.1, are consistent with the requirements of Division 3.

Only gas tungsten and submerged arc welding consumables according to SFA-5.9M [4] are used as welding material for the fabrication of the containment welds in the canister. Only welding processes that are capable of producing welds in accordance with the welding procedure qualification requirements of Section IX [5] and Division 3, Subsection WC-4000 are permitted. The manufacturer chooses the welding procedure under consideration of this requirement.

8.1.3.2 Fabrication and examination of welds in the CLU

Transfer cask and transfer lock consist of a welded steel construction. The transfer cask exhibits primary and secondary member welded joints according to Subsection NF-1215 [3], whereas the structural skeleton of the transfer lock only exhibits secondary member welded joints. The drawings of the CLU components (see Section 1.5) include the fabrication specification and the required NDE procedure for each welded joint. The acceptance standards for NDE of welds in the transfer cask and the transfer lock, as specified in Section 10.1, are in accordance with the acceptance standards of Subsection NF-5300.

Only welding processes that are capable of producing welds in accordance with the welding procedure qualification requirements of Section IX [5] and Division 1, Subsection NF-4000 are permitted for welds in the transfer cask and the transfer lock. The manufacturer chooses the welding procedure under consideration of this requirement.

List of References

[1] ANSI N14.5 – 2014
American National Standard for Radioactive Materials –
Leakage Tests on Packages for Shipment

[2] ANSI N14.6 – 1993
Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More

[3] ASME Boiler and Pressure Vessel Code (Edition 2017)
Section III Rules for Construction of Nuclear Facility Components
Division 1 – Subsection NF

[4] ASME Boiler and Pressure Vessel Code Section II Part C, Edition 2017 SFA-5.9M "Specification for Bare Stainless Steel Welding Electrodes and Rods"

[5] ASME Boiler and Pressure Vessel Code (Edition 2017)
Section IX Welding, Brazing, and Fusing Qualifications

8.2 Material Properties

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

This chapter documents the material data to be used for the evaluations of the CASTOR® geo69 DSS and CLU.

The report considers the specific materials of parts for the design parts lists of the transfer lock, the transfer cask and the protection cover of the CASTOR® geo69 DSS. For the materials that are also part of the design parts lists of the cask, the canister, the basket, and the shielding elements refer to the data given in the SAR (transport) for the transport package of the CASTOR® geo69 DSS.

If not stated otherwise, the applicable material properties according to the respective requirements of BPVC, Section II and Division 3 are taken into account. For applied materials other than specified as described above, the properties are considered as given below.

8.2.1 Mechanical and Thermal Properties

With respect to the design as dual-purpose cask, the properties of the materials not already specified as cited above are as given in Table 8.2-1:

Table 8.2-1 Mechanical and thermal material properties

Material	Standard	Type/Grade	Emission Coefficient ε [-]	Reference
Low Alloy Carbon Steel	[redacted]		0.36	[1]
Stainless Steel	[redacted]		0.36	[1]
Low Alloy Carbon Steel	[redacted]		0.36	[1]
Low Alloy Carbon Steel	[redacted]		0.36	[1]

8.2.2 Radiation Shielding and Criticality Control Materials

Table 8.2-2 Chemical composition lead [redacted]

Element	Max. content [wt-%]
Sb	0.0005
As	0.0005
Sn	0.0005
Cu	0.0010
Ag	0.0010
Bi	0.0015
Zn	0.0005
Pb	bal.

Density ρ [kg/ dm³] = 11.33

Reference acc. to [2]

8.2.3 Bolt Applications

Materials for bolt applications have adequate resistance to corrosion and brittle fracture and a coefficient of thermal expansion similar to the materials being bolted together. The materials follow the requirements for the mechanical properties, temperature limits and design stress intensity limits listed in BPVC, Section II, Part D, Table 4.

8.2.4 Seals

For seals, metal gaskets are applied. [redacted] By this, corrosion from inadvertent rainwater intrusion is excluded.

List of References

[1] Kern D.Q. „Process Heat Transfer" McGraw Hill Kogakusha (1950)

[2] ASTM Volume 02.04 Nonferrous Metals B 29 "Standard Specification for Refined Lead"

8.3 Environmental Degradation

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

8.3.1 Chemical, Galvanic and other Reactions

The materials of the CASTOR® geo69 DSS and CLU have been reviewed and as a result, no safety related component is significantly influenced by chemical, galvanic or other reactions during loading and storage operations.

During the loading operations the basket, canister and transfer cask are in contact with the pool water. These materials are stainless steels, aluminium alloys, aluminium boron carbide metal-matrix-composite (MMC) and different polymers with high chemical resistance. These materials are all compatible with the pool water. The Lead shield within the transfer cask is not exposed to the pool water.

The transfer lock is made of different stainless steels, coated carbon steel and polymer compounds. No Influence of the ambient conditions on the materials is implied.

During storage, the storage casks are equipped with a protection plate. The components are either stainless steel or coated carbon steel. No Influence of the ambient conditions on the materials is implied.

The interior of the CASTOR® geo69 DSS is dried and filled with helium after the loading operations. This provides a dry and inert environment during storage. Corrosion reactions depend on the presence of water and/or oxygen. The dry inert helium gas atmosphere in the CASTOR® geo69 DSS precludes corrosion during storage. Exterior surfaces and materials consist of stainless steels, aluminium alloys or coated materials. Therefore, no chemical, galvanic or other reaction have to be assumed.

The materials of the CASTOR® geo69 DSS and CLU are summarized in Table 8.3-1. In presence of water, dissimilar materials can form a galvanic couple. During loading, the aluminium alloys and the aluminium boron carbide (MMC) form a galvanic couple with stainless steel. Both types of aluminium develop a native passive layer that precludes significant corrosion effects. To minimize galvanic and other corrosive reactions, the aluminium components, which are in contact with stainless steels, are additionally anodized. In consequence, no galvanic reactions to the aluminium alloys or the stainless steels occur during the loading time.

The storage cask body is made of ductile cast iron and the exterior surfaces will be coated to preclude corrosion reactions at the surfaces. The cask cavity is filled with helium and no corrosion has to be assumed.

The lids of the storage cask are made of stainless steel and are in contact with the zinc-coated alloyed steel bolts. The bolts of the lid are not directly exposed to the ambient weather, because they

are covered by the protection plate during storage. Thus, no significant corrosion effects are to be implied for the bolts.

During loading operations of the canister, before drying and refilling with helium, only minor amounts of hydrogen gas will be generated due to the minimized galvanic reaction of the aluminium and stainless steel and due to radiolysis of the water. This hydrogen will be evacuated from the canister during the drying process and no significant concentration of hydrogen can occur.

Lubricants are used to coat the screw threads. Only permitted lubricants are used for the coating of the screw threads. Before assembly or loading, all cask components will be inspected and freed from any form of contamination or marking. The lubricants have no significant effect on the cask materials.

There are no significant chemical, galvanic or other reactions that could reduce the integrity of the cask during the loading and storage operations.

Table 8.3-1: Environment during loading and storage for DSS and CLU components

Material / Component	*Environment during loading*	*Environment during storage*
High alloyed stainless Steels: [redacted] Basket, canister	Stainless steels in contact with both borated and unborated water do not exhibit chemical or galvanic reactions or interactions with spent fuel.	The environment for these components will be an inert helium atmosphere. No further chemical, galvanic or other reactions are assumed.
Aluminium boron carbide (MMC) Basket	The aluminium boron carbide (MMC) forms a galvanic couple with stainless steels. The aluminium will be anodized to minimise any form of galvanic or other corrosion reactions. Due to the short loading time, in which they are in contact with pool water, the neutron absorber material is not exposed to significant chemical, galvanic or other reactions.	The environment for these components will be an inert helium atmosphere. No further chemical, galvanic or other reactions are assumed.

Material / Component	***Environment during loading***	***Environment during storage***
Aluminium alloys: [redacted] Basket, shielding elements	The aluminium alloy forms a galvanic couple with stainless steels. These aluminium components will be anodized to minimise any form of galvanic or other corrosion reactions. Due to the limited loading time in which they are in contact with pool water, the material is not exposed to significant chemical, galvanic or other reactions.	The environment for these components will be an inert helium atmosphere. No further chemical, galvanic or other reactions are assumed.
Steels: [redacted] Canister	These components are not in contact with the pool water.	The environment for these components will be an inert helium atmosphere. No further chemical, galvanic or other reactions are assumed.
High alloyed stainless steels: [redacted] Other stainless steel Transfer Cask	Stainless steels in contact with both borated and unborated water do not exhibit chemical or galvanic reactions or interactions with spent fuel.	The transfer cask is not part of the DSS. Components are exposed to ambient conditions. Stainless steels exhibit a native corrosion protection layer and no galvanic or other corrosion reactions have to be assumed. No chemical or other reactions are assumed.
Lead: [redacted] Transfer Cask	The lead shield is not in contact with pool water (enclosed by stainless steel components).	The transfer cask is not part of the DSS. Components are not exposed to ambient conditions. No chemical or other reactions are assumed.
Aluminium Transfer Cask	Not in contact with pool water.	The transfer lock is not part of the DSS. Components are exposed to ambient conditions. Aluminium and its alloys exhibit a native corrosion protection layer and no corrosion reactions have to be assumed. No chemical or other reactions are assumed.
Polymers: [redacted] Transfer Cask	Compounds with high chemical resistance. No degradation when in contact with pool water is implied. (Replaceable when transfer cask is not in use)	The transfer cask is not part of the DSS. The materials are exposed to ambient condition but no reactions are implied.

Material / Component	***Environment during loading***	***Environment during storage***
High alloyed stainless steels: [redacted] Other stainless steel Transfer lock	Not in contact with pool water.	The transfer lock is not part of the DSS. Components are exposed to ambient conditions. Stainless steels exhibit a native corrosion protection layer and no galvanic or other corrosion reactions have to be assumed. No chemical or other reactions are assumed.
Carbon steel: [redacted] Transfer lock	Not in contact with pool water.	The transfer lock is not part of the DSS. Material is coated and thus protected from ingress of moisture. Corrosion and other reactions are excluded.
Polymers: Iglidur® X PE HD PA Rubber Transfer lock	Not in contact with pool water. (Replaceable when transfer lock is not in use)	The transfer lock is not part of the Dry Storage System. The materials are exposed to ambient condition but no reactions are implied.
Ductile cast iron: [redacted] Storage Cask	This component is not in contact with the pool water.	The internal surfaces will be exposed to an inert helium atmosphere. Exposed external surfaces will be coated and will be maintained with a fully coated surface. No further chemical, galvanic or other reactions are assumed.
Polyethylene: [redacted] Storage Cask	This component is not in contact with the pool water.	The neutron moderator material has no contact to the outer environment. No chemical or galvanic reactions with DCI or steels are assumed. Other reactions are insignificant.
Carbon steel: [redacted] Protection cover	This component is not in contact with the pool water.	Material is coated and thus protected from ingress of moisture. Corrosion and other reactions are is excluded
High alloy stainless steel: [redacted] Protection cover	This component is not in contact with the pool water.	Components are exposed to ambient conditions. Stainless steels exhibit a native corrosion protection layer and no galvanic or other corrosion reactions have to be assumed. No chemical or other reactions are assumed.

Material / Component	*Environment during loading*	*Environment during storage*
High alloyed stainless steels: [redacted] Storage Cask	This component is not in contact with the pool water.	Some parts are exposed to ambient weather. Stainless steels exhibit a native corrosion protection layer and no galvanic or other corrosion reactions have to be assumed. No chemical or other reactions are assumed.
Constructional Steel [redacted] Storage Cask	This component is not in contact with the pool water.	This material has no contact to the outer environment. No significant chemical or galvanic reactions with ductile cast iron or other steels are assumed. No other reactions are assumed.
Steels: [redacted] Storage Cask	This component is not in contact with the pool water.	Components are exposed to atmosphere. These components are coated with a zinc layer. During the storage period, no loss of function has to be assumed. Components will be inspected prior to storage.
Silver, stainless steel and nickel based alloy Metal Gaskets	All materials of the gasket exhibit a native corrosion protection layer and no galvanic or other corrosion reactions have to be assumed. No chemical or other reactions are assumed.	All materials of the gasket exhibit a native corrosion protection layer and no galvanic or other corrosion reactions have to be assumed. No chemical or other reactions are assumed.
Organic Coating Storage Cask Transfer lock Protection cover	The coating is not in contact with the pool water.	The coating on external surfaces is exposed to ambient weather but exhibits a good long-term durability. Discoloration is not a concern. No further chemical, galvanic or other reactions are assumed.

8.3.2 Effects of Radiation on Materials

Metals are not impaired by gamma radiation. Significant radiation damages due to neutron exposure is not expected for metals at neutron fluences below 10^{18} n/cm². The expected neutron fluences of the components of the CASTOR® geo69 DSS will be significantly lower [redacted]

Elastomeric seals are exposed to gamma radiation and thus may undergo degradation. The elastomeric seals do not have safety related functions and the degradation products have a similar composition as the original molecule but different crosslinking or chain length. Thus, no harmful degradation products are expected.

The neutron shielding material polyethylene may be affected by irradiation analogously to elastomeric materials. The irradiation of the neutron shielding amounts approximately [redacted] The impairment through irradiation is insignificant and a loss of the neutron shielding ability does not have to be implied.

There is no significant extend of degradation of any important safety related component caused directly by the effect of the reactions. The same applies for the effects of reactions combined with the effects of exposure of the materials to neutron or gamma radiation.

8.4 Fuel Cladding Integrity

	Name, Function	Date	Signature
Prepared	[redacted]		
Prepared 8.4.3 only			
Reviewed			
Reviewed 8.4.3 only			

8.4.1 Spent Fuel Classification

The bounding cask array described in Section 1.4 comprises [redacted] CASTOR® geo69 storage casks and provides interim storage capacity for up to [redacted] fuel assemblies. The generic ISFSI uses CASTOR® geo69 casks for the storage of SNF.

Only intact FA complying with the limits specified in Subsection 1.2.3 are stored in the canister of CASTOR® geo69 casks. FA which have been deformed or damaged during reactor operation or which are otherwise defective in their structural integrity are not to be loaded into the cask. It is only allowed to load undamaged FA with completely filled grids into the cask. However, it is allowed to load FA with completely filled grids containing replacement fuel rods and/or replacement rods manufactured from solid material (dummy rods) used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

Damaged FA, fuel debris and associated nonfuel hardware are not to be stored in the fuel canister of CASTOR® geo69 storage casks.

Fuel qualification is based on the requirements for criticality safety, decay heat removal, radiological protection, and structural integrity.

During the operation of the NPP, fuel integrity has been, and continues to be, monitored. Through the detection of radiochemistry changes in the reactor coolant system, most fuel damage is assessed. When damaged rods are suspected, assemblies are inspected as they are removed from the core. All assemblies with positive indication of damage are again inspected in the spent fuel pool to determine amount and location of rods in the assembly that have failed cladding. If the FA is to be placed back in the reactor core, any failed rods are removed and replaced with nonfuel rods of equivalent dimensional properties. If the suspected damaged FA are at the end of their cycle, the assemblies may be stored in the spent fuel pool without repair. During this process, all known rod failures are noted and their assemblies are tracked. If the failure is visible from the exterior of the assembly, the damage may be videotaped. For assemblies that were removed from the reactor core and were not inspected at that time, inspections will be performed prior to loading these assemblies into the storage cask. This will ensure that there are no undetected failed rods in any assembly that is placed into a cask.

Under this failure detection process, inspections to date have found a limited amount of failures. Where single failed rods have been identified and removed, they were stored in the spent fuel pool and would ultimately be stored in a canister that can contain fuel debris, but not in the canister of CASTOR® geo69 storage cask.

This detection process, along with the history of plant operations and spent fuel storage in the fuel pool, provide a high level of confidence that the current SNF will meet the criteria for storage in the canister. In addition, based on the condition of the current spent fuel, the continued maintenance of the reactor coolant and spent fuel pool water chemistry requirements, and proper handling of the fuel, there is a high level of confidence that future SNF will meet the criteria for storage in the canister.

A cask-loading plan ensures that no damaged FA are loaded into the canister. If the structural integrity criterion is met, then approval for dry storage for a given assembly is issued. This qualification is documented and consequently referenced in the ISFSI operating procedures prior to loading spent FA into the canister.

The cask-loading plan provides a loading sequence based on the various characteristics of the FA being loaded. There are two main fuel-loading strategies used: uniform fuel loading and regionalized fuel loading.

Uniform fuel loading is used when the FA being loaded are all of similar burn-up rates, decay heat levels, and post-irradiation cooling times. In this case, the actual location of each assembly is less critical and assemblies can be placed at any location in the canister.

Regionalized fuel loading is used when high heat emitting FA are to be stored in the canister. This loading strategy allows these specific assemblies to be stored in locations towards the centre of the canister basket provided lower heat emitting FA are stored in the peripheral storage locations.

The following controls ensure that each FA is loaded into a known cell location within a qualified canister:

- A cask-loading plan is independently verified and approved.
- A fuel movement sequence is based upon the written loading plan. All fuel movements from any rack location are performed under controls that ensure strict, verbatim compliance with the fuel movement sequence.
- Prior to placement of the canister lid, all FA and associated nonfuel hardware, if included, is either videotaped or visually documented by other means, and independently verified, by ID number, to match the fuel movement sequence.

Finally, a third independent verification is performed to ensure that the fuel in the canister is placed in accordance with the original cask-loading plan.

Based on the qualification process of the spent fuel and the administrative controls used to ensure that each FA is loaded into the correct location within a canister, incorrect loading of a canister is not considered a credible event.

8.4.2 Cladding Mechanical Properties

As described in section 8.4.1, the FA can be assumed intact at the time of loading into the canister of the CASTOR® geo69 storage cask.

For at least one of the fuel types that shall be loaded into the canister of the CASTOR® geo69 cask, the burn-up exceeds 45 GWd/Mg_{HM} (see Section 2.1). For all fuel types the intended dry storage time is beyond 20 years. Hence according to NUREG-2224 [1] age-related uncertainties connected with the extended dry storage of HBU SNF are to be considered in the safety analyses.

The chosen approach in this SAR is to supplement the design basis with safety analyses that demonstrate the DSS can still meet the pertinent regulatory requirements by assuming hypothetical reconfiguration of the HBU fuel contents into justified geometric forms. This approach demonstrates that after 20 years of dry storage, even if reconfigured, fuel can still meet the 10 CFR Part 72 requirements for thermal, confinement, criticality safety, and shielding during normal, off-normal, and accident conditions.

Following NUREG-2224 [1], the impact of cladding failures of Category 1 with breached rods (Scenario 1(a) according to [1]) and with damaged rods (Scenario 1(b) according to [1])

- on the fuel cladding and package component temperatures is evaluated in Chapter 4,
- is considered by the external dose and dose rate evaluation in Chapter 5 and
- on the canister pressure is evaluated in Chapter 7.

Rupture of 1 percent, 10 percent, and 100 percent of the fuel rods is assumed for normal, off-normal, and accident conditions of storage, respectively.

The storage cask is designed to exclude the water leakage into the canister cavity under normal, off-normal and accident conditions of storage. Due to very low reactivity of dry fuel, the behaviour of the spent fuel as a result of accident conditions during dry storage beyond 20 years does not need to be explicitly evaluated and is bounded by the reactivity of the fully flooded cask with pure unborated water, as assumed in the bounding criticality safety model for normal conditions of storage (see Section 6.4).

No cladding mechanical properties and structural evaluation of the fuel rods under design-bases drop accident scenarios are required, once 100 percent rupture of the fuel rods and the resulting fuel reconfiguration for accident conditions of storage are evaluated as described in the safety analyses above.

8.4.3 Cover Gas

After exposure to oxidizing atmosphere, the fuel pellets may oxidize, expand and apply stress on the cladding. This potential oxidation is prevented by the measures/parameters applied during the drying procedure and the backfilling of the cavity with inert helium gas. This way it is ensured that the cladding will be protected against splitting from fuel pellet oxidation.

The canister is dried according to a specified drying procedure in accordance with the recommendations of R. W. Knoll and E. R. Gilbert [2]. Thereby, as much water as practicable is removed from the cavity and the pressure is evacuated to be less than or equal to 4.0×10^{-4} MPa. After evacuation, adequate moisture removal is verified by maintaining a constant pressure over an appropriate period without vacuum pump operation (or the vacuum pump is running but it is isolated from the cask with its suction vented to atmosphere). Moisture is removed to levels below 0.43 mole H_2O.

Potential icing of the evacuation system line during evacuation is considered and excluded through adequate measures and use of suitable devices. This way possible ice blockage of the canister evacuation path are prevented.

The canister cavity is then backfilled with helium as inert gas for applicable pressure and leak testing. The applied cover gas fulfils a defined quality specification that ensures a known maximum percentage of impurities and is additionally verified by sampling. This way, the source of potentially oxidizing impurity gases and vapours are minimized and contaminants are adequately removed from the canister. The maximum quantity of oxidizing gasses (e.g., oxygen, carbon dioxide, and carbon monoxide) are thereby limited to 1 mole per cask. This 1 mole limit reduces the amount of oxidants to below levels where cladding degradation is expected. Afterwards the inert gas atmosphere is maintained and no ingress from oxidizing atmosphere is implied.

In case the DSS confinement cavity is opened to an oxidizing atmosphere (as may occur in conjunction with remedial welding, seal repairs), the process of evacuation and re-pressurisation is repeated.

List of References

[1] NUREG-2224, Dry Storage and Transportation of High Burnup Spent Fuel
Office of Nuclear Material Safety and Safeguards, November 2020

[2] Evaluation of cover gas impurities and their effects on the dry storage of LWR spent fuel
R. W. Knoll; E. R. Gilbert (November 1987)
Prepared for the U.S. Department of Energy under Contract DE-AC06-76RL0 1830

8.5 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

With intent no items.

9 Operating Procedures

9.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

This chapter includes a description of the designated operating procedures with the CASTOR® geo69 DSS and CLU, comprising sufficient details regarding loading and unloading of the storage cask under usage of the CLU as well as preparation of the loaded storage cask and establishing the DSS configuration for long-term interim dry storage. The operations shall ensure the performance of the DSS and that it is operated in a manner consistent with the conditions assumed in the safety evaluation chapters of this SAR.

All operations shall be performed according to detailed written and approved procedures which shall comply with the content of this document, the applicable codes and standards and the CoC. The preparation of these procedures, which are site-specific, is in the responsibility of the user. The results from tests to be performed within the scope of the operations (e.g. leakage tests) shall be documented and become part of the quality documentation of the DSS.

The operational procedures have to be consistent with maintaining occupational radiation exposures as low as reasonably achievable (ALARA) as required by 10 CFR 20.1101(b). Occupational doses for operating procedures are estimated in Chapter 11.

In general, even if not explicitly mentioned, all components shall be subject to visual inspections prior to handling to ensure they are in proper condition.

9.1 Procedure for Loading the Storage Cask

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

The CASTOR® geo69 is designated as transport and storage cask. The procedure for loading the CASTOR® geo69 via the CLU is identical to the loading procedure described in Section 7.1 of the SAR (transport). Therefore, this procedure is not described in this section.

9.2 Procedure for Unloading the Storage Cask

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

The CASTOR® geo69 is designated as transport and storage cask. The procedure for unloading the CASTOR® geo69 via the CLU is identical to the unloading procedure described in Section 7.2 of the SAR (transport). Therefore, this procedure is not described in this section.

9.3 Preparation of the Storage Cask for Long-Term Interim Dry Storage

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	[redacted]
Reviewed	[redacted]	[redacted]	[redacted]

This section describes the preparation of the storage cask for long-term interim dry storage and the set-up of the CASTOR® geo69 DSS at the storage facility. The operations described in this section shall be performed subsequent to the procedures for receipt of either the package from transport carrier described in Section 7.2 of the SAR (transport) [1] or the storage cask after onsite transfer from the NPP. The initial situation comprises the storage cask (the impact limiters are removed) after visual inspection, resting on the transport frame in horizontal orientation. The cask lid system is leak tight in accordance with ANSI N14.5, dose rates are in accordance with 10 CFR 71.47 and the storage cask is free from contamination.

Item numbers refer to the design parts list of the storage cask [2].

The designated storage position of the cask must be prepared prior to performance of the operations specified in Table 9.3-1. Equipment and further components (e.g. pressure switch) necessary for establishing the CASTOR® geo69 DSS must be available.

Table 9.3-1: Operations for preparation for long-term interim dry storage

Step	*Description*	*Requirement*
1	**Transfer of the storage cask to the service station of the storage facility.**	
1.1	Load attachment on the trunnions (Item 12) of the storage cask.	ANSI N14.6 NUREG-0612
1.2	Tilting of the storage cask by application of the tilting studs and the corresponding tilting support.	
1.3	Vertical crane transfer (if necessary under usage of a supporting lock wagon) to the service station of the storage facility and positioning in the service platform.	
1.4	Closing of the service platform around the storage cask and installation of the temporary additional shielding.	
2	**Installation of the pressure switch in the cask lid (Item 55).**	
2.1	Removal of cap screws (Item 37), protection cap (Item 113) and metal gasket (Item 44). Visual inspection of the demounted components.	
2.2	Controlled pressure normalisation in the cask cavity using corresponding equipment via quick connect (Item 60).	
2.3	Removal of cap screws (Item 37), blind flange (Item 89) and metal gasket (Item 71).	
2.4	Visual inspections of • Pressure switch, • Cap screws and application of lubricant, • New metal gasket, • Sealing groove in pressure switch and surface in cask lid, • Threaded holes	
2.5	Installation of the pressure switch at the former position of the blind flange without metal gasket and cap screws in the cask lid. Tightening of the cap screws with nominal torque.	
2.6	Check for proper installation of the pressure switch and removal of the pressure switch.	

Step	Description	Requirement
2.7	Installation of the pressure switch at the former position of the blind flange with the new metal gasket and cap screws in the cask lid. Tightening of the cap screws with nominal torque. Recording of the metal gasket identification.	Cask logbook
2.8	Check for proper installation of the pressure switch.	
2.9	Functional test of the pressure switch.	
2.10	Helium backfilling of the storage cask via quick connect (Item 60) and adjustment of the internal pressure.	[redacted]
2.11	Leakage test of the metal gasket in the pressure switch.	ANSI N14.5
2.12	Visual inspections of • Protection cap, • Cap screws and application of lubricant, • New metal gasket, • Sealing groove in protection cap and surface in cask lid, • Threaded holes	
2.13	Installation of the protection cap without metal gasket and cap screws in the cask lid. Tightening of the cap screws with nominal torque.	
2.14	Check for proper installation of the protection cap and removal of the protection cap.	
2.15	Installation of the protection cap with the new metal gasket and cap screws in the cask lid. Tightening of the cap screws with nominal torque. Recording of the metal gasket identification.	Cask logbook
2.16	Check for proper installation of the protection cap.	
2.17	Leakage test of the metal gaskets in the protection cap.	ANSI N14.5
2.18	Disassembly of the temporary additional shielding.	
3	**Set-up of CASTOR® geo69 DSS**	
3.1	Control of the conservation status of the storage cask and implementation of improvements, if necessary	
3.2	Load attachment on the trunnions of the storage cask.	ANSI N14.6 NUREG-0612
3.3	Removal of the service platform around the storage cask.	
3.4	Vertical crane transfer of the storage cask to the designated storage position in the storage facility. Disassembly of the crane traverse.	
3.5	Visual inspection of the protection cover incl. attachments and load attachment points.	
3.6	Load attachment on the LAPs of the protection cover. Crane transfer of the protection cover and installation onto the storage cask.	
3.7	Installations of the cable conduit and feeding of the pressure switch cables through the cable conduit	
3.8	Connection of CASTOR® geo69 DSS to the pressure monitoring system of the storage facility and check of functionality.	
4	**Long-term interim dry storage for the designated storage period of the CASTOR® geo69 DSS.**	

List of References

[1] 1014-SR-00001, Rev. 0
Safety Analysis Report – Type B(U)F Transport Package CASTOR® geo69
Docket No.: 71-9383

[2] 1014-DPL-30934, Rev. 0
Design Parts List Cask, CASTOR® geo69

9.4 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

With intent no items.

10 Acceptance Criteria and Maintenance Program

10.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the CASTOR® geo69 DSS and the CLU to verify that the SSCs classified as important to safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this SAR. The acceptance criteria and maintenance program requirements specified in this chapter apply to each CASTOR® geo69 DSS and CLU fabricated, assembled, inspected, tested, accepted and maintained for use under the scope of the CoC issued by the NRC in accordance with the requirements of 10 CFR 72. The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters (incl. the parts lists and drawings presented in Section 1.5) ensure that the CASTOR® geo69 DSS and the CLU, if applicable, will retain the stored radioactive materials; will maintain subcriticality control; will properly transfer the decay heat of the stored radioactive materials; and that radiation doses will meet regulatory requirements.

Identification and resolution of nonconformance shall be performed in accordance with the QAP as described in Chapter 14. Nonconformance reports shall become part of the documentation of the DSS and the CLU.

10.1 Acceptance Criteria

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

This section provides the acceptance tests to be performed, demonstrating that each CASTOR® geo69 DSS and CLU is fabricated, assembled, inspected, tested, and accepted for use in accordance with the design criteria of this SAR and that the initial operation complies with the regulatory requirements. All tests applicable to DSS and CLU shall be performed in accordance with written and approved procedures (10 CFR 21.162). The results shall be documented and become part of the quality documentation.

10.1.1 Visual Inspection and Nondestructive Examination

10.1.1.1 General

All components shall be visually inspected for intactness, cleanness and non-corrosiveness.

The materials of construction shall be recipe inspected for visual and dimensional acceptability, taking the parts lists and the corresponding drawings included in Section 1.5 as a basis. A check of the manufacturing documentation (material certification) for completeness and factual correctness shall be included. The dimensions of DSS, CLU and all parts given in the manufacturing documentation shall conform to the dimensions and tolerances specified in the drawings.

Any occurring noncompliance with the parts lists and drawings leads either to the further treatment of the part to correct its dimensions, or to the manufacturing of a new exemplar of that particular part. Further treatment of DSS parts shall comply with Division 3 and further treatment of CLU parts shall comply with Division 1, Subsection NF. A verification that each part consists only of the materials specified in the part lists must be included in the manufacturing documentation. Furthermore, the storage cask must be conspicuously and durably marked according to 10 CFR 72.236(k) with a model number, a unique identification number and an empty weight.

10.1.1.2 Weld Examinations

The examination of welds shall verify that the fabrication of DSS, CLU and components confirm with the drawings in the SAR and referred to in the CoC. All weld examinations shall be performed in compliance with the appropriate Article of Section V for the particular examination method. Weld seams exist on the following parts of the DSS and the CLU:

- 1014-DPL-36855 canister (welded canister body)
- 1014-DPL-13752 protection cover (weld seams between plate and ring)
- 1015-DPL-37509 transfer cask (inner liner, [redacted] are welded to the head ring and bottom ring)
- 1015-DPL-38148 transfer lock (weld joint connections of the cylinder console to the frame)

The non-destructive examination procedures applicable depend on the category of the weld seams according to Division 3, WC-3251. Weld examination of containment welds of the canister body shall be performed in accordance with Division 3, WC-5200 with regard to the required examination method and acceptance criteria. Personnel performing non-destructive examinations shall be qualified in accordance with Division 3, WC-5520. [redacted] Division 1, NF-5210 determines the required NDE methods of the welds in the CLU components, NF-5300 the acceptance criteria and NF-5400 the personal qualification.

Table 10-1 gives an overview on the required examinations of welds on the different parts of the DSS. The required DNE procedures shall be performed in accordance with Section V [2] of the BPVC, except for the protection cover.

Table 10-1: Required examination of welds in the DSS and the CLU

Weld seam location	*Code / standard*	*Welded joint category*	*Examination method*
Canister body	Division 3 Subsection WC	A, B, C	RT + PT
Protection cover	[redacted]	[redacted]	[redacted]
Transfer cask	Division 1 Subsection NF	Primary member Secondary member	RT + PT + VT VT
Transfer lock	Division 1 Subsection NF	Secondary member	VT

10.1.1.2.1 Examination of Welds in the Canister Body

Table 10-2 summarizes the required NDE of welds in the canister body. Location, types and size of each weld must be confirmed by the required examination method before commissioning. Defects in weld metal detected by the examinations shall be eliminated and repaired when necessary or the indication shall be reduced to an acceptable limit.

Table 10-2: Required Examination of Welds in the Canister Body

Part	*Weld location*	*t [mm]*	*Weld type*	*Required NDE*
Canister body	Item 2-2 – Item 2-3	[redacted]	Single U butt weld	RT, PT
	Item 2-3 – Item 2-4			
	Item 2-2 – Item 2-6			
	Item 2-4 – Item 2-6			

* Item No. according to the respective parts list of storage cask and canister, referenced in Section 1.2

Radiographic Testing

RT shall be performed in accordance with Section V, Article 2, except that fluorescent screens are not permitted for film radiography, the geometric unsharpness shall not exceed the limits of Section V, Article 2, T-274.2, and the image quality indicators of Division 3, Table WC-5111-1 shall be used in lieu of those shown in Section V, Article 2, T-276. Indications shown on the radiographs of welds and characterized as imperfections shall meet the acceptance criteria according to Division 3, WC-5320.

Penetration Testing

PT shall be performed in accordance with Section V, Article 6. Imperfections producing indications that do not meet the acceptance criteria specified in Division 3, WC-5352 are unacceptable.

10.1.1.2.2 Examination of Welds in the CLU

Table 10-3 summarizes the required NDE of each weld in the transfer cask and the transfer lock.

Table 10-3: Required Examination of Welds in the CLU

Part	*Weld location **	*t [mm]*	*Weld type*	*Required NDE*
Transfer cask body	[redacted]	[redacted]	Bevel groove weld	RT
		[redacted]	V groove weld	VT
		[redacted]	Fillet weld	VT
		[redacted]	V groove weld	VT
		[redacted]	Fillet weld	VT
		[redacted]	Bevel groove weld	VT
		[redacted]	Fillet weld	VT
		[redacted]	V groove weld	VT
		[redacted]	V groove weld	VT
		[redacted]	Fillet weld	VT
		[redacted]	Fillet weld	VT
		[redacted]	V groove weld	VT
		[redacted]	Bevel groove weld	VT
		[redacted]	Fillet weld	VT

Part	Weld location *	t [mm]	Weld type	Required NDE
	[redacted]	■	V groove weld	VT
		■	V groove weld	VT
		■	Bevel groove weld	VT
		■	Bevel groove weld	VT
		■	Fillet weld	VT
		■	Fillet weld	VT
		■	Fillet weld	VT
		■	Bevel groove weld	VT
Transfer lock		■	Fillet weld	VT
		■	Fillet weld	VT

* Item No. according to the respective parts list of storage cask and canister, referenced in Section 1.2

Radiographic Testing

RT shall be in accordance with Section V, Article 2, except that the geometric unsharpness shall not exceed the limits of BPVC Section V, Article 2, T-274.2. Only the welded connections between head ring, liner and bottom ring in the transfer cask body require RT under consideration of the acceptance criteria specified in BPVC Section III, Division 1, Subsection NF-5320. Indications shown on the radiographs of the welds shall meet these acceptance criteria.

Visual Testing

VT shall be performed in accordance with Section V, Article 9. When VT is performed on welds in the transfer cask or the transfer lock, the acceptance criteria according to BPVC Section III, Division 1, Subsection NF-5360 shall apply.

10.1.1.2.3 Examination of Welds in the Protection Cover

For the non-destructive examination of the weld seams in the protection cover, the acceptance criteria of [redacted] apply. Each weld shall be tested via VT and PT. VT shall be performed according to ISO 17637 [3] or another applicable standard. PT shall be performed according to ISO 3452-1 [4] or another applicable standard. Defects detected by the examinations shall be eliminated and repaired when necessary or the indication shall be reduced to an acceptable limit.

10.1.2 Structural and Pressure Tests

All test shall comply with the applicable cods and standards mentioned in the following respective subsections.

10.1.2.1 Pressure Tests

Pressure tests during fabrication only apply to the DSS, as the CLU does not have pressure-bearing components. For the pressure tests, the requirements of Division 3 apply. [redacted] shall be performed with a maximum internal overpressure of at least 1.25 times the design pressure. The external design pressure of the canister is equal to the internal design pressure of the storage cask. Since this pressure exceeds the internal design pressure in the canister, the requirements of Division 3, WC-6610 apply. The canister shall be pressure tested with a test pressure equal to 1.25 times the external design pressure. The applied test pressures for the storage cask and canister are summarized in Section 3.7 and 3.9. They cover the required minimum test pressures according to Division 3. The test pressure shall be maintained for at least 10 minutes. The canister test pressure shall not be exceeded by more than 6 % during the test. Examinations after the pressure test shall be in accordance with WC-6224.

After the test period time of the [redacted] tests is exceeded, the initiation of the examination for leakage starts. LT shall be performed as described in Subsection 10.1.3 with an internal pressure equal to [redacted] of the [redacted] test pressure. All joints, connections and regions of high stress are tested for leaks.

10.1.2.2 Structural Tests

Structural tests are performed on all LAP in terms of a static load test. The applied load depends on the maximum weight that will be attached to the LAP during handling of the DSS and CLU components, multiplied with a certain load factor. The load attachment points on the canister lid and the trunnions of storage cask and transfer cask are special lifting devices and thus tested with a load factor of three in order to fulfil the provisions of ANSI N14.6 [5]. All other lifting devices are tested with a load factor of 1.5. Furthermore, a hoist factor of 1.15 according to CMAA #70 applies to all lifting devices for slow crane operation. The loads for the static load tests are listed in Table 10-4. All load tests shall be performed at room temperature and the static load shall be maintained for at least 30 minutes.

The static load tests of the trunnions and the tilting studs require the use of bearing shells as indicated in Figure 10.1-1. The requirements for the geometry of the bearing shells are specified in Subsection 10.1.2.2.1 and 10.1.2.2.2.

Table 10-4: Requirements for static load tests

Lifting device (quantity)	*Load factor*	*Max. weight attached [Mg]*	*Load [kN]*
Storage cask trunnions (2)	3 x 1.15	[redacted]	[redacted]
Transfer cask trunnions (2)	3 x 1.15	[redacted]	[redacted]
LAP on canister lid (12)	3 x 1.15	[redacted]	[redacted]
Tilting studs (2)	1.5 x 1.15	[redacted]	[redacted]
LAP on cask lid (6)	1.5 x 1.15	[redacted]	[redacted]

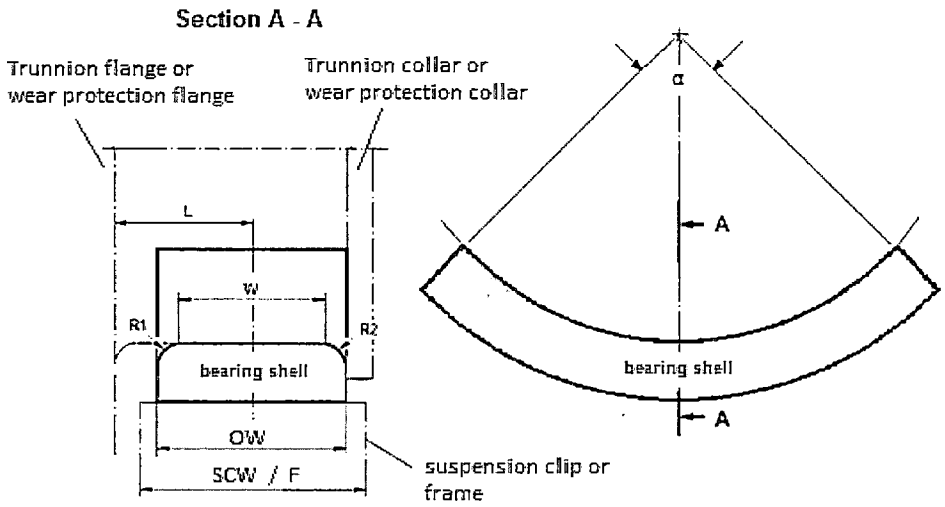

Figure 10.1-1: Geometry of load application on the trunnions / tilting studs

10.1.2.2.1 Storage Cask Trunnions

The static test load must be distributed equally on the two trunnions of the storage cask. The direction of load application must be perpendicular to the axis of the trunnions. Bearing shells [redacted] must be placed between lifting lug and the trunnion. The following requirements must be met:

[redacted]

After the load test, a visual inspection and MT or PT must be performed on the installed trunnions, as far as accessible, to verify no distortion or cracking has occurred. MT or PT shall be performed in accordance with Section V, Article 7 or Article 6, respectively, using the acceptance criteria specified in Division 3, WC-5342 or WC-5352. The trunnion deformation between the initial value prior to loading (null measurement) and after loading with the test load shall be determined via dial gauge (measuring accuracy ± [redacted]). A permanent trunnion deformation exceeding the measuring inaccuracy of the dial gauge is unacceptable. Otherwise, the deformed trunnions shall be replaced by new exemplars and the load test shall be repeated.

The cap screws that are exposed to the highest loading during the static lead test shall be removed and visually inspected after the test. Thread testing of these screws and the corresponding threaded holes in the storage cask body shall be performed. Defective cap screws shall be replaced with new exemplars. The tightening torque of all cap screws shall be checked after the load test.

10.1.2.2.2 Tilting Studs

The load test on the tilting studs of the storage cask has to be performed prior to the installation of the wear protection. The static test load must be distributed equally on the two tilting studs and must be applied perpendicular to the axis of the tilting studs. Bearing shells [redacted] must be used and the following requirements apply:

[redacted]

A visual inspection of the of the tilting studs as well as MT or PT of the whole shell surface of the tilting studs and in the area of the transition radius shall be performed after the test load is applied, to verify no distortion or cracking has occurred. The deformation of the tilting studs due to the load shall be determined via dial gauge (measuring accuracy [redacted]). A permanent deformation of the tilting studs exceeding the measuring inaccuracy of the dial gauge is unacceptable. In case of a non-permitted deformation, the cask body shall be replaced by a new exemplar.

10.1.2.2.3 Canister Lid

The static load must be distributed equally on the [redacted] threaded holes by using the lifting pintle for the canister lid. The load must be applied perpendicular to the cask surface. The test screws must be screwed into the lid by hand until the minimum required screw-in depth is reached. A visu-

al inspection and a gauge test of the threaded holes and a check of the screw-in depth on engagement must be performed before and after the load test. No adverse effects on the trueness to gauge of the threaded holes are permissible. The canister lid must be replaced by a new exemplar if the threaded holes do not pass the static load test.

10.1.2.2.4 Cask Lid

The static load must be distributed equally on the [redacted] threaded holes by using the lifting pintle for the cask lid. The load must be applied perpendicular to the cask lid surface. The test screws must be screwed into the lid by hand until the minimum required screw-in depth is reached. A visual inspection of the threaded holes and a check of the screw-in depth on engagement must be performed before and after the load test. No adverse effects on the trueness to gauge of the threaded holes are permissible. The cask lid must be replaced by a new exemplar if the threaded holes do not pass the static load test.

10.1.2.2.5 Transfer Cask Trunnions

The static test load must be distributed equally on the two trunnions of the transfer cask. The direction of load application must be perpendicular to the axis of the trunnions. Bearing shells [redacted] must be placed between lifting lug and the trunnion. The following requirements must be met:

[redacted]

After the load test, a visual inspection and MT must be performed on the installed trunnions, as far as accessible, to verify no distortion or cracking has occurred. MT shall be performed in accordance with Section V, Article 7, using the acceptance criteria specified in Division 3, NF-5342. The trunnion deformation between the initial value prior to loading (null measurement) and after loading with the test load shall be determined via dial gauge (measuring accuracy ± 0.01 mm). A permanent trunnion deformation exceeding the measuring inaccuracy of the dial gauge is unacceptable Otherwise, the deformed trunnions shall be replaced by new exemplars and the load test shall be repeated.

The cap screws that are exposed to the highest loading during the static lead test shall be removed and visually inspected after the test. Thread testing of these screws and the corresponding threaded holes in the transfer cask shall be performed. Defective cap screws shall be replaced with new exemplars. The tightening torque of all cap screws shall be checked after the load test.

10.1.3 Leak Tests

At any time the containment boundaries of storage cask or canister are completely set up (during assembly, dispatch after loading, etc.), helium LT (helium detection via mass spectrometer) shall be performed on all metal gaskets. LT is performed in accordance with Section V, Article 10 and ANSI N14.5 [6].

The leak detector shall have a sensitivity of at least 10^{-10} Pa m³/s. The maximum permissible standard helium leakage rate according to ANSI N14.5 is 10^{-8} Pa m³/s. A calibration of the leak detector is performed using a test leak before connecting the measuring setup to the test volume. Table 10-5 lists all containment boundaries with metal gaskets that require leak testing.

Table 10-5: Containment boundaries with metal gaskets requiring LT

Containment	*Sealing surface **	*Metal Gasket **	*Component **
Storage cask	Cask body (Item 2)	Item 69,	Cask lid (Item 55)
	Cask lid (Item 55)	Item 44	Protection cap (Item 113)
	Cask lid (Item 55)	Item 71	Blind flange (Item 89)
Canister	Canister body (Item 2)	Item 16	Canister lid (Item 3)
	Canister lid (Item 3)	Item 13	Tightening plug (Item 10)

* Item No. according to the respective parts list of storage cask and canister, referenced in Section 1.2

The leak tightness is considered proven when the calculated standard helium leakage rate is lower than the permissible one. In case of an unacceptable leakage rate the corresponding sealing barrier is to be depressurised and opened. The metal gasket or elastomeric seal is to be replaced, the sealing surface checked for cleanliness and damage and necessary repairs performed. Containment material may be repaired in accordance with Division 3, WC-2500 or disposed of. After the new gasket and the sealing surface are preserved, the containment boundary is to be closed again and a new leak test has to be performed.

10.1.4 Components

10.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no valves, rupture discs or fluid transport devices associated with the containment boundaries of the CASTOR® geo69 DSS. The only valve-like components in the DSS are quick connections installed in the vent and drain ports in the service orifice of canister and cask lid. After completion of drying and helium backfill operations, the service orifice in the canister lid is covered and sealed by the tightening plug and a metal gasket and the one in the cask lid by the protection cap and a metal gasket. LT to verify canister and cask containment boundaries is performed. The quick connections are thus not accessible during transport or storage unless the containment boundaries remain closed.

The pressure in the water chambers of the transfer cask is limited by means of pressure relief valves to be plugged to the quick connection in the top side of the outer water chamber. The pressure relief valves shall relieve at an overpressure of [redacted]

10.1.4.2 Gaskets

The sealing performance of metal gaskets and the properties of the corresponding materials are specified in Section 8.2. Each metal gasket and the corresponding sealing surface is visually inspected prior to installation and helium leak tested after closure of the corresponding lid or cover lid. The creep of the metal gaskets is limited to an extent that will not degrade its sealing performance during storage.

10.1.4.3 Miscellaneous

<u>FA receptacle calibre test</u>

After assembly of the basket a FA receptacle calibre test is used to simulate the loading of the canister with SNF in the nuclear facility. A replica of the intended FA is successively lowered into each receptacle of the basket to check if the fuel elements fit into the basket as requested. The calibre test is successfully completed when the replica fits into each receptacle without without jamming.

<u>Weighing of canister and transfer cask</u>

As required by Section 2.0 the total mass of the transfer cask carrying a canister loaded with SNF must not exceed the maximum crane capacity of [redacted]. However, the masses of the respective components of canister with basket and shielding elements and CLU tabulated in Section 1.2 may result into a total weight of the loaded transfer cask of more than this limit, if all tolerances and a traverse weighing [redacted] are included in the considerations. Therefore, following the respective

assembly, weighing of both, the transfer cask as well as the canister with basket and shielding elements must be performed. The results of the weighing must be assessed with the limits listed in Table 10-6. If an actual manufactured transfer cask weights less than the tabulated limit canisters to be handled may weight respectively more unless the maximum crane capacity is not exceeded.

Table 10-6: Mass limits on transfer cask and canister (including internals)

Component	*Mass limit [kg]*
Transfer cask	[redacted]
Canister (including internals)	[redacted]

10.1.4.4 Shielding Test

The performed shielding evaluation provided in Chapter 5 shows that the dose rates calculated under conservative assumptions are always considerably below the regulatory dose rate limits for NCS and ACS. The components and materials serving as radiation shield for neutron and/or gamma radiation are described in Section 1.2. Material and component tests are performed to eliminate the possibility of defects, uncontrollable voids or streaming paths in the shielding, which could lead to a deviation from the calculated radiation profile and a local violation of a dose rate limit. Significant degradation, gas release, or physical alteration is not to be assumed. Each shielding component is visually inspected to ensure homogeneity and the absence of cracks, voids, shrinkage holes, pinholes and other defects in the material. The chemical composition of the materials and their properties ascertain that they exhibit the desired shielding properties used in the calculations. Hence, the intended shielding performance can be assumed throughout all loading operations and the entire storage time including maintenance operations. No additional shielding tests are designated for acceptance of shielding components.

10.1.5 Neutron Absorber Tests

Top, centre and bottom sheets of the fuel basket are made of the Aluminium-Boron Metal Matrix Composite (Al-B_4C-MMC) [redacted], which consists of a [redacted] Al alloy containing boron carbide (B_4C). During the manufacturing process, the chemical composition of the raw materials (aluminium alloy powder, B_4C powder), the blended powder and the extrusion profile are measured. The minimum required B_4C content in the processed material is [redacted]. Furthermore, a ^{10}B content in boron of [redacted] is required. Only neutron absorber material with a boron content above the minimum required value is accepted. Requirements for other elements are listed

in Table 10-7. Compliance with the required chemical composition ensures a minimum density of [redacted] as used in the criticality safety analysis.

Table 10-7: Required chemical composition of Al-B_4C-MMC neutron absorber material

[redacted]

[redacted]

Table 10-8: Mechanical Requirements for Al-B_4C-MMC neutron absorber material

[redacted]

10.1.6 Thermal Test

The thermal evaluation (see Chapter 4) provides reliable results in terms of three-dimensional temperature distribution and temperature time dependency in the DSS during storage and in the transfer cask during short-term operations. The results indicate that the maximum temperatures during both, short-term operations as well as NCS, off-normal conditions and ACS (fire phase and subsequent cooling phase) are considerably below the limit values of each component. The materials used in the CASTOR® geo69 DSS and CLU design are applied in several other transport and storage casks with a comparable design since many years. Thereby GNS exhibits many years of experience in modelling and simulation of their thermal behavior. The acceptance tests performed during material fabrication ensure that each material is manufactured in accordance with the applicable standard, which leads to reproducible thermal properties as specified in Section 8.2. The thermal properties of each material and the heat transfer between these materials determine the thermal behavior and heat dissipation in the DSS and CLU. Based on the established experience

and the previous verification of the applied simulation models it is ensured by GNS that the simulations describe the actual thermal behavior of the DSS under all conditions of storage and of the CLU during short-term operations with adequate accuracy. Therefore, no further thermal tests are designated for acceptance.

List of References

[1] ISO 10042 (2018)
Welding – Arc-welded joints in aluminium and its alloys – Quality levels for imperfections

[2] ASME Boiler and Pressure Vessel Code (2017)
Section V – Nondestructive Examination

[3] ISO 17637 (2016)
Non-destructive testing of welds – Visual testing of fusion-welded joints

[4] ISO 3452-1 (2014)
Non-destructive testing – Penetrant testing
Part 1: General principles

[5] ANSI N14.6 (1993)
Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More

[6] ANSI N14.5 (2014)
Radioactive Materials - Leakage Tests on Packages for Shipment

10.2 Maintenance Program

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

A periodically ongoing maintenance program shall be defined and incorporated into the CASTOR® geo69 DSS and CLU operations manuals, which shall be prepared and issued prior to the delivery and first use of the DSS to each user. These documents shall delineate the detailed inspections, maintenance, tests, and parts replacement necessary to ensure continued structural, thermal, and confinement performance; radiological safety, and proper handling of the system in accordance with 10 CFR 72.136(g) regulations, the conditions in the CoC, and the design requirements and criteria contained in this SAR.

The CASTOR® geo69 DSS is passive by design. There are no active components required to assure the performance of its safety functions. The pressure switch in the cask lid, which is connected to the pressure monitoring systems, is a self-reporting component, which means that a defect of the pressure switch will be automatically detected. As a result, only minimal maintenance will be required over the CASTOR® geo69 DSS lifetime, and this maintenance would primarily include replacement routines. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces or the replacement of the pressure switch in case of a reported defect.

The number of lifting and handling operations performed with the transfer cask and storage cask shall be documented in the cask logbook. Trunnions and trunnion bolts shall be replaced when the minimum number of permissible load cycles specified in Section 3.5 is reached.

The CLU components shall be inspected prior to usage.

Any maintenance operations shall be performed according to written and approved procedures. Results of tests, repaired or replaced parts and other maintenance operations shall be recorded and become part or the quality documentation of the DSS or CLU.

10.2.1 Structural and Pressure Tests

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be performed on the lifting trunnions of the transfer cask and on the trunnions and tilting studs of the storage cask. The examination shall inspect for indications of overstress such as cracking, deformation, or wear marks. Repairs or replacement in accordance with written and approved procedures shall be required if unacceptable conditions are identified. Testing to verify continuing compliance of the transfer cask and storage cask trunnions shall be performed in accordance with ANSI N14.6 [1].

As described in Chapter 7 and 12, there are no credible normal, off-normal, or accident conditions that can cause the structural failure of the DSS. Therefore, periodic structural or pressure tests on the DSS following the initial acceptance tests are not required as part of the storage maintenance program.

10.2.2 Leak Tests

The internal pressure of the storage cask is continuously monitored during storage via the pressure switch in the cask lid, which is connected to the pressure monitoring system of the storage facility. A leak in the containment boundary of the storage cask or the canister would lead to a pressure drop in the storage cask. Such a pressure drop is automatically reported by the pressure switch. Therefore, frequent leak tests are not required as part of the storage maintenance program. Leak tests are only performed on an as-needed basis when the pressure switch reports a pressure drop in the storage cask to determine the location of the leak.

10.2.3 Subsystem Maintenance

The CASTOR® geo69 DSS is connected to a pressure monitoring system during storage. Maintenance activities shall be performed by the licensee to ensure the operational reliability of the pressure monitoring system over the complete duration of storage.

Maintenance and calibration testing will be required on the vacuum drying, helium backfill, and leakage testing systems. Cranes and lifting beams shall be inspected prior to each loading campaign to ensure that proper maintenance and continued performance is achieved. Additional temporary neutron and gamma shielding provided during loading and transfer operations with the CLU requires no maintenance.

10.2.4 Valves, Rupture Discs, and Fluid Transport Devices

The pressure relief valves used on the water jackets of the transfer cask shall be calibrated on an annual basis (or prior to the next use if the period the transfer cask is out of use exceeds one year) to ensure the pressure relief setting is [redacted], or replaced with factory-set relief valves.

List of References

[1] ANSI N14.6 (1993)
Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More

10.3 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

With intent no items.

11 Radiation Protection

11.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]		
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This chapter discusses the design considerations and operational features that are incorporated in the CASTOR® geo69 DSS and CLU design to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during operating procedures and long-term interim dry storage. Occupational exposure estimates for operating procedures are provided in Section 11.3, including canister loading, dispatch and transhipment via CLU and closure of the storage cask. An off-site dose assessment of the DSS for a generic storage facility is provided in Section 11.4. Since the determination of off-site doses is necessarily site-specific, similar dose assessments are to be prepared by the licensee, as part of implementing the CASTOR® geo69 DSS in accordance with 10 CFR 72.212.

11.1 Ensuring that Occupational Radiation Exposures are as Low as Reasonably Achievable (ALARA)

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

11.1.1 Policy Considerations

The CASTOR® geo69 DSS has been designed in accordance with 10 CFR 72. DSS as well as CLU maintain radiation exposures ALARA consistent with 10 CFR 20. Licensees using the CASTOR® geo69 DSS and CLU will utilize and apply their existing site ALARA policies, procedures and practices for onsite activities to ensure that personnel exposure requirements of 10 CFR 20 are met. Personnel performing operations on the DSS or the CLU shall be trained on the operations and be familiarized with the expected dose rates. Pre-job ALARA briefings should be held with workers and radiological protection personnel prior to work on or around the DSS and CLU. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities, will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures, users shall ensure that ALARA practices are implemented and the 10 CFR 20 standards for radiation protection are met in accordance with the site's written commitments.

Users can further reduce dose rates around the DSS and the CLU within the admissible loading patterns by preferentially loading longer-cooled and lower burnup spent fuel assemblies in the periphery receptacles of the fuel basket, while loading assemblies with shorter cooling times and higher burnups in the inner basket receptacles.

11.1.2 Design Considerations

Consistent with the design criteria defined in Subsection 2.3.5, the radiological protection criteria that limit exposure to radioactive effluents and direct radiation from the CASTOR® geo69 DSS are as follows:

- 10 CFR 72.104 requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 0.25 mSv to the whole body, 0.75 mSv to the thyroid, and 0.25 mSv to any other critical organ. This dose would be a result of planned discharges, direct radiation from the DSS/CLU, and any other radiation in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements.

- 10 CFR 72.106 requires that any individual located on or beyond the nearest owner-controlled area boundary may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv. The sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye)

shall not exceed 0.5 Sv. The lens dose equivalent shall not exceed 0.15 Sv and the shallow dose equivalent to skin or to any extremity shall not exceed 0.5 Sv. The licensee is responsible for demonstrating site-specific compliance with this requirement.

- 10 CFR 20, Subpart C specifies occupational dose limits for adults. The annual total effective dose equivalent shall not exceed 0.05 Sv. The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye shall not exceed 0.5 Sv in a year. Additionally, an annual limit of 0.15 Sv for the lens dose equivalent and an annual limit of 0.5 Sv for the shallow-dose equivalent to the skin of the whole body or to the skin of any extremity shall not be exceeded. The licensee is responsible for demonstrating site-specific compliance with this requirement.

- 10 CFR 20, Subpart D; specifies dose limits for individual members of the public. The total effective dose equivalent from licensed operations shall not exceed 1 mSv in a year. The dose rate in any unrestricted area from external sources shall not exceed 0.02 mSv/h. The licensee is responsible for demonstrating site-specific compliance with this requirement.

11.1.3 Operational Considerations

The following operational considerations that most directly influence occupational exposures have been incorporated into the design of the CASTOR® geo69 DSS and CLU:

- Use of a DPC for transport and storage, which requires no unloading/transfer of the canister from transport into a separate storage cask and vice versa at the storage facility and thus reduces on-site radiation exposure;
- A totally-passive DSS design requiring minimum maintenance and monitoring (other than security monitoring) during storage;
- A self-reporting pressure switch in the cask lid that is connected to a pressure monitoring system at the storage facility during long-term interim dry storage to allow remote monitoring of the storage cask internal pressure;
- Mostly passive CLU designs requiring minimum human interactions in the vicinity;
- Remotely operated transfer lock, lifting gear, etc.;
- Use of remote handling equipment, where practical
- Use of e.g. quick connection couplings in the service orifices of the lids
- Use of additional temporary neutron and gamma shielding during canister dispatch for loading into the CASTOR® geo69 storage cask;
- Inspections and function tests prior to actual loading

- A sequence of operations based on ALARA considerations.
- Dry run trainings for the workers prior to actual FA handling

Operating controls and limits that are necessary for compliance with regulatory requirements and ALARA objectives are specified in Chapter 13.

11.1.4 Auxiliary/Temporary Shielding

To minimize occupational dose during loading and unloading operations, a specially designed set of auxiliary shielding (part of multi-equipment) will be used during loading and unloading operations performed with the CASTOR® geo69 CLU. The auxiliary shielding comprises additional temporary shielding (e.g. lead blankets, PE plates) for the lid area of transfer cask and storage cask and a shielding plate for the canister lid. Each auxiliary shield is described in Table 11.1-1 and the procedures for utilization are provided in Chapter 9. Table 11.1-1 provides the minimum requirements for the use of temporary additional shielding. Users shall evaluate the need for additional auxiliary and temporary shielding and use of special tooling to reduce the overall exposure based on an ALARA review of cask loading operations and the loaded contents.

Table 11.1-1: CASTOR® geo69 CLU auxiliary and temporary shields

Temporary Shield	*Description*	*Utilization*
Transfer cask	Temporary additional neutron and gamma shielding (e.g. lead blankets, PE plates) for the lid area of the transfer cask.	Installation in the reactor hall after placement of transfer cask (including the loaded canister) in the service platform. Removal before transfer to the CASTOR® geo69 storage cask.
Storage cask	Temporary additional neutron and gamma shielding (e.g. lead blankets, PE plates) for the lid area of the storage cask.	Installation in the truck lock after loading of the canister into the CASTOR® geo69 storage cask. Removal after closure of the storage cask and in case of any maintenance work during storage.
Canister	Shielding plate for the canister lid.	Installation before vacuum drying of the canister cavity in the reactor hall. Removal before transfer to the CASTOR® geo69 storage cask.

The use of temporary additional shielding is required at the lid areas of canister, transfer cask and storage cask. The lateral area of the transfer cask provides sufficient shielding during loading and unloading operations due to the integrated lead shield and the two water chambers. During canister transhipment in the truck lock, the contact area between transfer cask and storage cask is sufficiently shielded by the transfer lock. Therefore, the use of additional temporary shielding at the lateral area of transfer cask and storage cask is optional.

11.2 Radiation Protection Design Features

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

The development of both, the CASTOR® geo69 DSS as well as of the CLU have focused on design provisions to address the considerations summarized in Subsection 11.1.2 and 11.1.3. The intent has been to combine a canister-based system with the proven CASTOR® dual-purpose cask design for transport and storage requiring no additional loading or unloading procedures after closure of the cask in the reactor building. Canister and storage cask form a double-containment system around the loaded SNF. Both closure systems are re-openable and provide sufficient activity retention. The design thus combines the preferred features of canister-based and metal cask systems. This approach reduces overall radiation levels and the need for performing operating procedures and maintenance. The following specific design features of DSS and CLU ensure a high degree of containment integrity and radiation protection:

- Two independent containment barriers of the DSS
- Reduction of streaming paths to a minimum
- DSS equipped with
    - Thick monolithic DCI wall and bottom of the storage cask and a thick stainless steel canister and cask lid system;
    - UHMW-PE rods in the wall and plates in the cask bottom and below the cask lid;
    - Shielding elements in the canister;

  to reduce the surface dose rates;
- CLU equipped with
    - Thick stainless steel bottom lid as well as [redacted] of the transfer cask;
    - Thick [redacted] of the transfer lock;

  to reduce the surface dose rates;
- Material selection and smooth surface preparation (e.g. coating, almost no crud traps) of DSS and CLU to enable easy decontamination;

[redacted]

- A totally passive design requiring a minimum of maintenance during handling and storage.

11.3 Occupational Exposures

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading, surveillance and maintenance operations using the CASTOR® geo69 DSS and the CLU. This section uses the shielding evaluation provided in Chapter 5 and the operating procedures provided in Chapter 9 to develop a dose assessment. The dose rates from the CASTOR® geo69 storage cask and the CLU, each housing the canister, are calculated at various positions (height from ground and distance from lateral area) around the casks lateral and lid area (Figure 11.3-1) to determine the cumulative dose to personnel performing loading, unloading and transfer operations. The positions indicated in Figure 11.3-1 apply to the transfer cask (CLU) and the CASTOR® geo69 storage cask. The positions are enumerated with letters from A to Z. Each position is a possible inhabitancy for personell during the operating procedures described in Chapter 9.

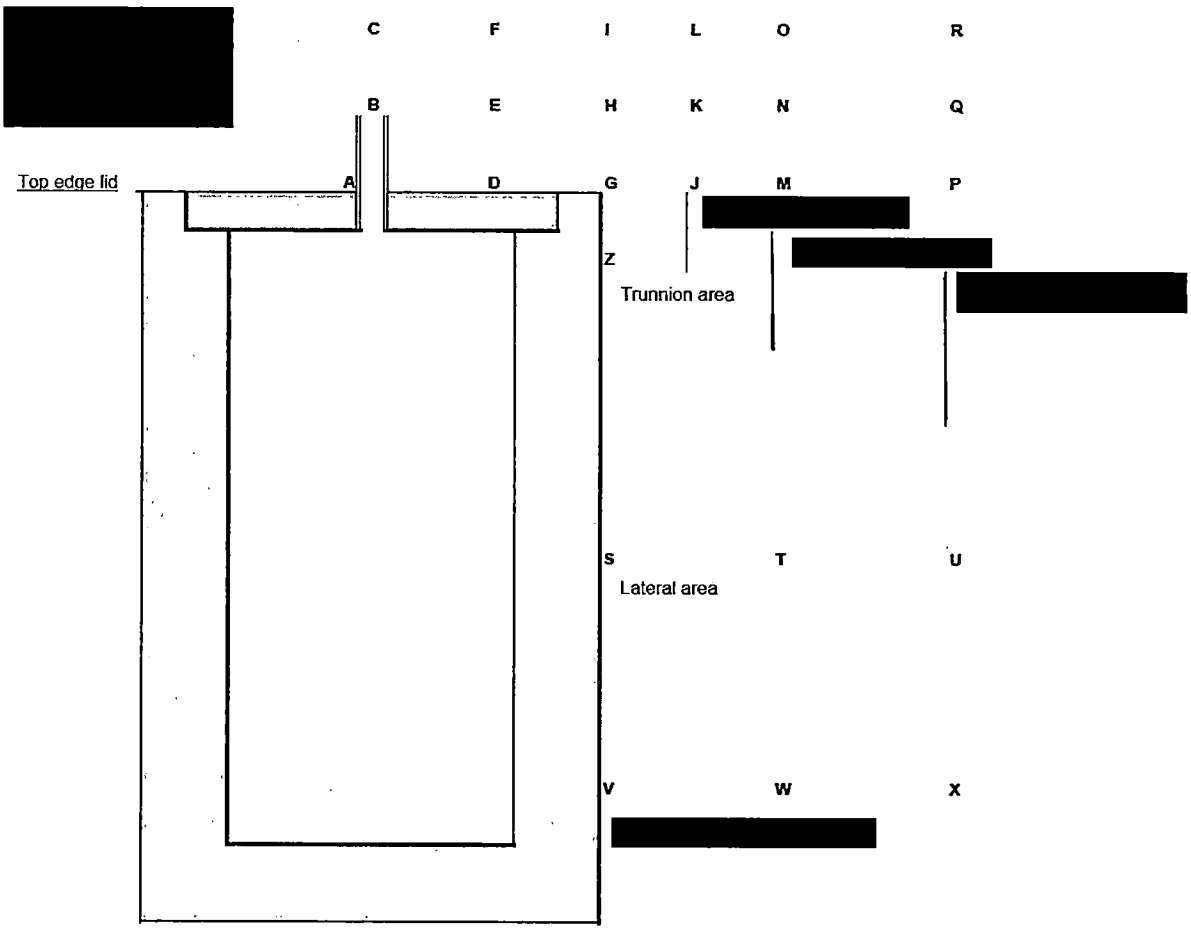

Figure 11.3-1: Relevant positions at the transfer/storage cask surface

There are no anticipated occurrences (off-normal conditions) during loading or unloading of the cask that lead to deviations from the dose rates specified in the following subsection or that require

actions to be performed to recover from the anticipated occurrence, leading to different occupational exposures than those specified in the following subsection.

11.3.1 Estimated Exposures for Loading the Cask

The cumulative exposure to personnel performing the procedure for loading the cask in accordance with Section 9.1 is calculated for a homogeneous content loading of the cask in accordance with loading pattern TR1 (see Chapter 5). Influences of other radiation sources, such as background radiation, are not taken into account. The calculation is performed for the loading procedure performed with and without the temporary additional shielding specified in Subsection 11.1.4. [redacted] The shielding effect of the service station in the reactor hall, which is required for the loading procedure, is neglected.

MCNP [1] is used for the dose rate calculation for all handling steps with relevant contribution to the cumulative dose to the staff comprising the lifting of the loaded transfer cask out of the SNF pool until the complete dispatch of the CASTOR® geo69 storage cask. Radiation from the active zone of the FAs, the head and foot pieces and from the structural parts are considered as neutron and gamma radiation sources.

The following four shielding configurations are modeled for the calculation of the cumulative dose to personell:

1. Location: Service station next to the SNF pool.
   Configuration: [redacted]
2. Location: Service station in the truck lock.
   Configuration: Canister completely dispatched, transfer cask positioned on top of the transfer lock atop the CASTOR® geo69 storage cask, [redacted]
3. Location: Service station in the truck lock.
   Configuration: Loaded CASTOR® geo69 storage cask without cask lid, transfer cask and transfer lock removed.
4. Location: Service station in the truck lock.
   Configuration: Loaded CASTOR® geo69 storage cask with cask lid.

The four shielding configurations are shown in Figure 11.3-2, Figure 11.3-3 and Figure 11.3-4 and the results of the occupational dose evaluations are listed in Table 11.3-1. For each relevant sub-step of each operation, the required time, the number of persons involved, the position of the persons (position according to Figure 11.3-1), the dose rate (without additional temporary shielding) at this position, the cumulative dose and the reduced dose with additional temporary shielding (if applicable) is listed. The dose is calculated by multiplying the dose rate with the time required for the operational sub-step and the number of persons involved in the operation. Some sub-steps require the abidance of personnel at more than one position for different duration times. For these sub-steps, the doses at each position are listed separately.

The total occupational dose (with and without additional temporary shielding) according to Table 11.3-1 is distributed to several individuals because many operational sub-steps require more than one person. When an even distribution of the dose to the involved personnel is assumed for every sub-step, the total occupational dose per person amounts to [redacted] without and [redacted] with temporary additional shielding.

Table 11.3-1 does not include every sub-step included in the procedure for loading the cask according to Section 9.1. Some steps are performed far away from the transfer cask or the storage cask and are thus not relevant for the total occupational dose. An example is the installation of the blind flange (with metal gasket) in the cask lid and the check for proper installation, which is performed before the cask lid is positioned and installed on the CASTOR® geo69 storage cask. The sequence of steps according to Table 11.3-1 is in accordance with Section 9.1.

[redacted]

Figure 11.3-2: Shielding configuration 1 – Loaded transfer cask

[redacted]

Figure 11.3-3: Shielding configuration 2 – Canister transhipment

[redacted]

Figure 11.3-4: Shielding configurations 3 and 4 – Loaded storage cask without/with cask lid

Table 11.3-1: Occupational doses for CASTOR® geo69 loading

Operation	*Sub-steps*	*Time [min]*	*Pers.*	*Pos.*	*Dose rate [mSv/h]*	*Dose [mSv]*	*Reduced dose [mSv]*
Shielding configuration 1							
Crane transfer of the loaded transfer cask to the service station next to SNF pool	[redacted]						
	[redacted]						
	[redacted]						
	[redacted]						
	Transfer to service station	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
	Removal of traverse	[redacted]					
Dispatch of canister and transfer cask at the service station	Removal of lifting pintle	[redacted]					
	Removal of guide bolts	[redacted]					
	Dewatering and drying of annulus between canister and transfer cask cavity	[redacted]					
	Drying of canister lid area	[redacted]					
	Installation of additional temporary shielding	[redacted]					
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Check for proper installation of canister lid	[redacted]					
	Vacuum drying of canister cavity	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Installation of blind plug and quick connect	[redacted]					
	Evacuation and helium filling of canister cavity	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Installation of tightening plug with metal gasket	[redacted]					
	Fastening of pressure nut	[redacted]					
	Check for proper installation of pressure nut	[redacted]					
	Leakage test of canister lid system	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Removal of additional temporary shielding	[redacted]					
	Installation of lifting pintle	[redacted]					

Operation	*Sub-steps*	*Time [min]*	*Pers.*	*Pos.*	*Dose rate [mSv/h]*	*Dose [mSv]*	*Reduced dose [mSv]*
Transfer of transfer cask to the CASTOR® geo69 storage cask	Attachment of traverse on trunnions	[redacted]					
	Crane transfer to truck lock	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
	Positioning of transfer cask in transfer lock on storage cask	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
Shielding configuration 2							
Canister transhipment	[redacted]	[redacted]					
	[redacted]						
	[redacted]	[redacted]					
	[redacted]						
	[redacted]						
	[redacted]	[redacted]					
Shielding configuration 3							
Removal of CLU components and canister lifting pintle	[redacted]						
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
	[redacted]						
Installation of retention ring in the CASTOR® geo69 storage cask	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
	[redacted]		[redacted]	[redacted]	[redacted]	[redacted]	
	[redacted]						
	[redacted]						
	[redacted]						
Dispatch of CASTOR® geo69 storage cask	Removal of sealing surface protection	[redacted]					
	Installation of guide bolts	[redacted]					
	Positioning of cask lid (load attachment on lifting pintle)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
	Shielding configuration 4						
	Installation of additional temporary shielding	[redacted]					

Operation	Sub-steps	Time [min]	Pers.	Pos.	Dose rate [mSv/h]	Dose [mSv]	Reduced dose [mSv]
	Installation of cask lid bolting						
	Check for proper installation of cask lid						
	Removal of lifting pintle						
	Installation of blind flange or pressure switch with metal gasket and cap screws in cask lid and check for proper installation						
	Vacuum drying of cask cavity						
	Helium back filling of cask cavity and adjustment of pressure						
	Installation of protection cap with metal gasket and cap screws in cask lid and check for proper installation						
	Leakage test of protection cap						
	Leakage test of blind flange						
	Leakage test of cask lid						
	Removal of additional temporary shielding						
	Preparation for transport of the storage cask out of the building						
Total occupational dose							

11.3.2 Estimated Exposures for Unloading the Cask

The occupational dose for the procedure for unloading the cask in accordance with Section 9.2 is expected to be smaller than the occupational dose for loading the cask. Most sub-steps of the unloading procedure, which are relevant for occupational exposure, are consistent with sub-steps of the loading procedure, but are performed in reverse order. The four shielding situations described in Subsection 11.3.1 appear in reverse order during the unloading procedure.

Unless there are no reasons for intervention, unloading of the CASTOR® geo69 storage cask is expected to take place after the intended storage period of the DSS. During long-term interim dry storage, the total activity of the loaded contents decreases exponentially, leading to lower radiation source terms compared to those specified in Section 5.2. Radiation resulting from cask activation as a consequence of neutron irradiation during storage can be neglected according to Section 2.4. Therefore, occupational dose rates during unloading of the cask are lower than during loading of the cask.

11.3.3 Estimated Exposures for Surveillance and Maintenance

An estimation of the occupational exposure required for security surveillance and maintenance during long-term interim dry storage of the CASTOR® geo69 DSS is necessarily site specific.

Security surveillance time is based on a daily security patrol around the controlled area boundary of the storage facility. According to the results of the shielding evaluation presented in Section 5.1, the CASTOR® geo69 DSS complies with 10 CFR 72.104 and the maximum permitted annual dose equivalent of 0.25 mSv at the controlled area boundary for normal operations and anticipated occurrences, assuming 100 % occupancy (8766 hours) and a bounding array of [redacted]. Therefore, the occupational annual dose due to security surveillance is only a small fraction of this limit value. A duration of [redacted] for a daily security patrol around the controlled area boundary leads to a maximum annual dose equivalent of approximately [redacted] This complies with the occupational dose limits specified in 10 CFR 20, Subpart C.

Occupational exposure for maintenance operations is reduced to a minimum due to the passive design of the CASTOR® geo69 DSS. Only minimum maintenance will be required over the lifetime of the DSS. An ongoing maintenance program, including specified maintenance procedures and intervals, will be defined and incorporated into the CASTOR® geo69 DSS operations manual, prior to the delivery and first use of the DSS. The maintenance procedures and intervals shall comply with the requirements of 10 CFR 20, Subpart C.

Maintenance operations on the CLU components are most likely performed while the CLU is not in use and thus without any radiation exposure from radioactive contents. Radiation exposure for maintenance of CLU components is negligible since the transfer cask is decontaminated after loading in the SNF pool and the exposure time (neutron irradiation during loading of the cask) is too short to lead to significant radiation resulting from activation of CLU components.

List of References

[1] C.J. Werner (ed.), MCNP User's Manual – Code Version 6.2, LA-UR-17-29981, 2017

11.4 Exposures at or Beyond the Controlled Area Boundary

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

11.4.1 Controlled Area Boundary Dose for Normal Conditions of Storage

10 CFR 72.104 limits the annual dose equivalent to any real individual at or beyond the controlled area boundary to a maximum of 0.25 mSv to the whole body, 0.75 mSv to the thyroid, and 0.25 mSv for any other critical organ. This includes contributions from all uranium fuel cycle operations in the region, including CLU operations. Radiation exposure in connection with handling operations involving the CLU contributes to the controlled are boundary dose for NCS.

Dose rates and doses at the controlled area boundary result from the direct neutron and gamma radiation stemming from the loaded CASTOR® geo69 DSS. The structural integrity and the redundant containments are not impaired during normal, off-normal and accident conditions of storage, as demonstrated in Chapter 7. Therefore, no radioactive material is released from the DSS that could contribute to the controlled area boundary dose.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of the storage facility and the number and configuration of storage casks are necessarily site-specific. In order to compare the performance of the CASTOR® geo69 DSS with the regulatory requirements, a bounding array of storage casks is analysed in Chapter 5. Users are required to perform a site specific dose analysis for their particular situation in accordance with 10 CFR 72.212. The analysis must account for size, configuration and FA specifics of the storage installation and any other radiation from uranium fuel cycle operations within the region.

Section 5.1 presents dose rates and annual doses at the storage site for an array [redacted] as a function of distance. 100 % occupancy (8766 hours) is conservatively assumed. 10 CFR 72.106 specifies that the minimum distance from the storage facility to the nearest boundary of the controlled area must be at least 100 m. For open-air storage without additional shielding provided by a storage building, the minimum distance needed to meet the annual dose requirements of 10 CFR 72.104 is approx. [redacted] away from the centre of the long side of the DSS array specified in Section 5.1. [redacted]

These results demonstrate the compliance of the CASTOR® geo69 DSS design with the requirements listed in Subsection 11.1.2. Users are required to perform a site-specific analysis to demonstrate compliance with 10 CFR 72 104 and 10 CFR 20. Neither the distances nor the array configurations specified in this subsection become part of the technical specifications of the DSS or the ISFSI.

11.4.2 Controlled Area Boundary Dose for Off-normal Conditions of Storage

As demonstrated in the shielding evaluation in Chapter 5, there are no factors influencing the shielding performance of the storage cask under off-normal conditions of storage. None of the off-normal conditions analysed have an impact on the shielding analysis. The only significant difference between off-normal conditions and NCS is that 10 % fuel rod failure is assumed instead of 3 %. Fuel rod failure under normal or off-normal conditions of storage does not lead to an increase of the dose rate. Therefore, the dose at the controlled area boundary from direct radiation for off-normal conditions is equal to that for NCS.

11.4.3 Controlled Area Boundary Dose for Accident Conditions of Storage

10 CFR 72.106 specifies the maximum allowed doses to any individual at the controlled area boundary from any design basis accident. In addition, it is specified that the distance to the controlled area boundary must be at least 100 m.

For accident conditions of storage, it is conservatively assumed that all moderator material in the DSS is lost. It is also assumed that no storage building is left around to provide additional shielding for the DSS array. These assumptions are bounding for all design basis accidents specified in Subsection 2.2.3. The [redacted] dose at a distance of 100 m remains safely under the maximum total effective dose equivalent of 0.05 Sv specified in 10 CFR 72.106, as demonstrated in Section 5.1. The DSS thus fulfils the shielding requirements for ACS at the controlled area boundary for the worst case shielding consequence.

11.5 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]		
Reviewed			

With intent no items.

12 Accident Analyses

12.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

This chapter presents the evaluation of the CASTOR® geo69 DSS for the effects of off-normal and accident conditions of storage. The design basis off-normal and accident conditions of storage, including events resulting from mechanistic and non-mechanistic causes as well as those caused by natural phenomena, are identified in Section 2.2. For each postulated event, the event cause, means of detection, consequences, and corrective action are discussed and evaluated. As applicable, the evaluation of consequences includes structural, thermal, shielding, criticality, containment, and radiation protection evaluations for the effects of each design event.

The structural, thermal, shielding, criticality, and containment features and performance of the CASTOR® geo69 DSS are discussed in Chapter 3, 4, 5, 6 and 7, respectively. The evaluations provided in this chapter are based on the design features and evaluations described therein.

According to Chapter 2, off-normal and accident conditions during canister handling via CLU are not to be assumed. The evaluations provided in this chapter exclude the CLU components.

12.1 Off-Normal Conditions

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

Off-normal conditions, as defined in accordance with ANSI/ANS-57.9 [1], are those conditions, which, although not occurring regularly, are expected to occur no more than once a year. In this section, expected off-normal conditions according to Subsection 2.2.2 are considered.

The results of the evaluations demonstrate that the CASTOR® geo69 DSS withstands the effects of off-normal conditions without affecting any of its safety functions. The following subsections present the evaluation of the DSS for the design basis off-normal conditions. Based on this evaluation, it is concluded that off-normal conditions do not affect the safe operation of the CASTOR® geo69 DSS. Compliance with the requirements of 10 CFR 72.122 10 CFR 72.104(a) and 10 CFR 20 is demonstrated.

12.1.1 Off-normal Pressure

12.1.1.1 Postulated Cause of Event

The off-normal pressure in the canister bounds the cumulative effects of the maximum fill gas volume, off-normal environmental temperatures, the maximum SNF heat load, and an assumed 10 % of the fuel rods ruptured with 100 % of the fill gas and 15 % of the fission gases released due to a cladding breach in accordance with NUREG-2224 [2].

After FA loading, the canister is drained, dried, and backfilled with helium to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods during dry storage is low. Nonetheless, the event is postulated and evaluated.

The off-normal pressure in the storage cask is unchanged to NCS according to Section 7.3.

12.1.1.2 Detection of Event

The CASTOR® geo69 is designed to withstand off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement for detection of off-normal pressure. Therefore, no monitoring is required.

12.1.1.3 Analysis of Effects and Consequences

The off-normal internal pressure of the canister and the corresponding boundary conditions are reported in Section 7.3. The applied pressure values for storage cask and canister are summarized in the appendix of Chapter 3.

Structural

The structural evaluation of the DSS for off-normal internal pressure conditions is discussed in Chapter 3. Internal overpressure acts in hot temperature conditions, while external overpressure is considered in cold temperature conditions. The stress results indicate that the DSS structure can withstand the applied loadings due to off-normal pressure.

Thermal

The temperatures under off-normal conditions are evaluated in Section 4.5 and for a failure of 10 % of the fuel rods after 20 years of storage in Section 4.8. The evaluation of the results show that all calculated maximum temperatures of the DSS components and the content are far below the maximum admissible values with large safety margins.

Shielding

Off-normal pressure conditions have no effect on the shielding performance of the DSS.

Criticality

Off-normal pressure conditions have no effect on the criticality control of the DSS.

Containment

Off-normal pressure conditions have no effect on the containment function of the DSS. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring containment boundary integrity.

Radiation Protection

Since there is no degradation in shielding or containment capabilities as discussed above, off-normal pressure conditions have no effect on occupational or public exposures.

12.1.1.4 Corrective Actions

No corrective action is required for off-normal pressure conditions.

12.1.2 Off-normal Temperatures

12.1.2.1 Postulated Cause of Event

Off-normal environmental temperatures are a consequence of seasonal variations. The DSS is standing inside a storage hall without direct influence by the environmental temperature at the storage facility site and by insolation. The ambient temperatures inside the storage hall during NCS already consider seasonal variations (see Section 4.5) so that off-normal environmental temperatures are bounded by NCS.

Off-normal temperatures as a consequence of fuel rod failure after 20 years of storage are calculated in Section 4.8.

12.1.2.2 Detection of Event

The CASTOR® geo69 DSS is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures.

12.1.2.3 Analysis of Effects and Consequences

The event off-normal temperatures is bounded by NCS, which consider an air temperature of [redacted] in the storage hall. This is the maximum air temperature between the storage casks and is based on the highest mean value over [redacted] of the environmental temperature at the storage facility site recorded over the last years. The minimum air temperature inside the storage hall of [redacted] is sufficiently low to consider off-normal seasonal variations.

Structural

The structural evaluation of the canister for off-normal conditions of storage in hot and cold condition is provided in Chapter 3. The off-normal conditions of storage provides for the DSS components a thermal equilibrium at their respective maximum temperatures for hot conditions and at [redacted] for cold conditions. Thermal stresses occur due to differential thermal expansion. The stress results indicate that the DSS structure withstands the applied loadings due to off-normal temperatures.

Thermal

The off-normal temperature condition is bounded by NCS. It is demonstrated in Section 4.4 that the CASTOR® geo69 DSS fulfils all requirements for NCS with regard to thermal aspects.

Off-normal temperatures as a consequence of fuel rod failure after 20 years of storage are calculated in Section 4.8. The maximum temperatures of the DSS components and the content are far below the maximum admissible values with large safety margins.

Shielding

Off-normal temperature conditions have no effect on the shielding performance of the DSS.

Criticality

Off-normal temperature conditions have no effect on the criticality control of the DSS.

Containment

Off-normal temperature conditions have no effect on the containment function of the DSS. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring containment boundary integrity.

Radiation Protection

Since there is no degradation in shielding or containment capabilities as discussed above, off-normal temperature conditions have no effect on occupational or public exposures.

12.1.2.4 Corrective Actions

No corrective action is required for off-normal temperature conditions.

List of References

[1] ANSI/ANS-57.9-1992
Design Criteria For An Independent Spent Fuel Storage Installation (Dry Type)
American National Standards Institute

[2] NUREG-2224, November 2020
Dry Storage and Transportation of High Burnup Spent Nuclear Fuel – Final Report
U.S. Nuclear Regulatory Commission

12.2 Accidents

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

Accidents, in accordance with ANSI/ANS-57.9 [1], are either infrequent events that could reasonably be expected to occur during the lifetime of the CASTOR® geo69 DSS or events postulated because their consequences may affect the public health and safety. Section 2.2 defines the design basis accidents considered. By analysing for these design basis events, safety margins inherently provided in the CASTOR® geo69 DSS design can be quantified.

The results of the evaluations performed herein demonstrate that the CASTOR® geo69 DSS withstands the effects of all credible and hypothetical accident conditions and natural phenomena without affecting its safety function. The following subsections present the evaluation of the design basis accident and natural phenomena, which demonstrate that the requirements of 10 CFR 72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10 CFR 72.106(b) and 10 CFR 20.

12.2.1 Fire Accident

12.2.1.1 Cause of Fire Accident

The possibility of a fire accident at or near the storage site is considered to be extremely remote due to the absence of significant combustible materials. The only credible source for a fire is the fuel tank of the transport vehicle, which is used to move the storage cask to the storage position in the storage hall. [redacted]

12.2.1.2 Fire Accident Analysis

To demonstrate the fuel cladding and pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded CASTOR® geo69 DSS is performed. During the postulated fire accident, the storage cask is completely engulfed by an 800°C hot fire. The surface of the storage cask receives an incident radiation and forced convection heat flux from the fire.

Structural

The structural integrity and the redundant containments of the CASTOR® geo69 DSS are not impaired by the fire accident (see Chapter 3). The resulting thermal stresses are within the allowable values.

Thermal

The fire duration is approximately [redacted] according to Section 4.6. Due to the thermal inertia of the DSS, many components reach the maximum temperature during the subsequent cooling

phase. The evaluation of the results show that all calculated maximum temperatures of DSS components and content are far below the maximum admissible values for ACS with large safety margins.

Shielding

The assumed loss of all moderator material in the CASTOR® geo69 DSS results in an increase in the radiation dose rates. However, the shielding evaluation presented in Chapter 5 demonstrates that the requirements of 10 CFR 72.106 are not exceeded at any time.

Criticality

The fire accident has no effect on the criticality control of the DSS.

Containment

The fire accident has no effect on the containment function of the DSS. Both containment barriers remain leak-tight. The internal pressure in the canister and the cask cavity is reported in Section 7.4. The temperatures of the components of the containment boundary do not exceed the short-term allowable temperature limits (see Subsection 4.6.2).

Radiation Protection

There is no degradation in containment of the CASTOR® geo69 DSS, as discussed above. Increase in the dose rate due to the loss of moderator material is evaluated in Chapter 5.

12.2.1.3 Fire Accident Dose Calculations

The complete loss of the CASTOR® geo69 DSS neutron moderator (UHMW PE rods and plates) material is assumed in the shielding analysis for the post-accident analysis of the loaded storage cask in Chapter 5 and bounds the determined fire accident consequences. The loaded CASTOR® geo69 storage cask following a fire accident meets the accident dose rate requirement of 10 CFR 72.106(b). The [redacted]s-dose remains safely under the dose limit of 0.05 Sv at 100 m distance

12.2.1.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded CASTOR® geo69 DSS, the ISFSI operator shall take the appropriate immediate actions necessary to extinguish the fire. Firefighting personnel should take appropriate radiological precautions. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed. As appropriate, temporary shielding shall be installed around the DSS.

If damage to the CASTOR® geo69 DSS as the result of a fire event is widespread and/or as radiological conditions require, the canister shall be removed from the storage cask in accordance with the procedure specified in Section 9.2. However, the thermal analysis described herein demonstrates that the DSS components remain below the accident temperature limits. The DSS may be returned to service if there is no significant increase in the measured dose rates and if the visual inspection is satisfactory.

12.2.2 Tip-Over

12.2.2.1 Cause of Tip-Over

The analysis of the CASTOR® geo69 DSS has shown that the cask does not tip over as a result of the postulated and analysed design basis accidents. It is highly unlikely that the storage cask will tip-over during on-site transfer because of the required use of a lifting device designed in accordance with ANSI N14.6 [2] and NUREG-0612 [3] as specified in Section 2.0. The tip-over accident is stipulated as a non-mechanistic accident.

12.2.2.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded DSS tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Chapter 3.

Structural

A storage cask drop analysis with the longitudinal axis horizontal (side drop) together with a storage cask drop analysis with the longitudinal axis vertical (bottom end drop) are assumed as a bounding accident combination for the non-mechanistic tip-over analysis in Chapter 3. The structural evaluation demonstrates that under the calculated [redacted] loading the stresses are within the allowable values.

Thermal

The thermal analysis of the CASTOR® geo69 DSS is based on the vertical storage configuration. The thermal consequences after the tip-over, when the DSS is in the horizontal orientation, are bounded by the thermal analysis for fuel rod failure for ACS impact in Section 4.8, which considers a maximum amount of fission gas and leads to maximum temperatures in the canister. The evaluation of all results for ACS impact (scenario I and scenario II as specified in Subsection 4.8.6) show that the calculated maximum temperatures of DSS components and content are far below the maximum admissible values with large safety margins.

Shielding

The bottom of the storage cask, which is normally facing the ground, will face the horizon after the tip-over. This orientation does not lead to dose rates exceeding the total effective dose limits specified in 10 CFR 72.106. Containment and shielding material are not impaired by the storage cask tip-over.

Criticality

The tip-over accident has no effect on the criticality control of the DSS. The displacement of the loaded FA as a result of the horizontal orientation of the DSS after tip-over is bounded by the model for criticality evaluation in Chapter 6.

Containment

The tip-over accident has no effect on the containment function of the DSS. As discussed in the structural evaluation above, all stresses remain within allowable values, which suggests containment boundary integrity.

Radiation Protection

There is no degradation in containment of the CASTOR® geo69 DSS as a result of the tip-over accident. Based on a minimum distance to the controlled area boundary of 100 meters, the [redacted] dose at the controlled area boundary (see Section 5.1) does not exceed the 10 CFR 72.106 dose requirements, even for a complete loss of the moderator material.

12.2.2.3 Tip-Over Dose Calculations

The analysis of the tip-over accident has shown that the DSS containment barriers will not be compromised and, therefore, there will be no release of radioactivity or increase in site-boundary dose rates from release of radioactivity.

The tip-over accident could cause localized minor damage to the outer shell of the storage cask (e.g. cooling fins). However, due to the massive design of the DCI cask body, containment and shielding material are not impaired. There will be no significant increase in the on-site or the boundary dose rate as a result of the localized damage.

12.2.2.4 Tip-Over Corrective Actions

Following a tip-over accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the DSS. Special handling procedures, including the use of temporary shielding, will be developed to restore the vertical orientation of the DSS. Likewise, the CASTOR® geo69 DSS shall be thoroughly inspected and a determination shall be made if repairs are required and will enable the DSS to return to service. Subsequent to the repairs, the

equipment shall be inspected and appropriate tests shall be performed to certify the DSS for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

12.2.3 Earthquake

12.2.3.1 Cause of Earthquake

It is possible that, during the intended storage period of the CASTOR® geo69 DSS, the ISFSI may experiences an earthquake.

12.2.3.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded CASTOR® geo69 DSS. The objective is to determine the stability limits of the DSS. Based on a static stability criteria, it is shown in Chapter 3 that the CASTOR® geo69 DSS is qualified to seismic activity less than or equal to the values specified in Section 13.2. The analyses in Chapter 3 show that the CASTOR® geo69 DSS will not tip over under these conditions and that incipient sliding occurs before incipient tipping. The space between neighbouring storage casks prevents possible collisions.

Some storage sites may have design basis earthquakes that exceed the seismic activity limits specified in Section 13.2. These high-seismic sites require an additional evaluation of the DSS design with regard to the structural consequences of such an earthquake. Additional safety measures (e.g. anchoring of the DSS on the storage pad) must be implemented to prevent a tip-over of the DSS as a result of the earthquake.

Structural

The sole structural effect of the earthquake is an inertial loading of less than [redacted]. This loading is bounded by the non-mechanistic tip-over analysis presented in Chapter 3. The calculated steady state deceleration of the CASTOR® geo69 DSS is [redacted]. The stress results indicate that the DSS structure can withstand the applied impact loadings.

Thermal

The earthquake has no effect on the thermal performance of the DSS.

Shielding

The earthquake has no effect on the shielding performance of the DSS.

Criticality

The earthquake has no effect on the criticality control features of the DSS.

Containment

The earthquake has no effect on the containment function of the DSS.

Radiation Protection

Since there is no degradation in shielding or containment capabilities as discussed above, there is no effect on occupational or public exposures as a result of the earthquake.

12.2.3.3 Earthquake Dose Calculation

The structural analysis of the earthquake accident shows that the loaded DSS will not tip over as a result of the specified seismic activity, hence, there is no increase in radiation dose rates or release of radioactivity to be expected.

12.2.3.4 Earthquake Corrective Actions

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the DSS to determine if any of the storage casks have slipped or tipped-over. In the unlikely event and against all expectations of a tip-over, the corrective actions shall be in accordance with Subsection 12.2.2.4.

12.2.4 100 % Fuel Rod Failure

12.2.4.1 Cause of 100 % Fuel Rod Failure

This accident event postulates that all fuel rods rupture and that the appropriate quantities of fission product gases and fill gas are released from the fuel rods into the canister cavity. Through all credible ACS, the CASTOR® geo69 DSS maintains the SNF in an inert environment while maintaining the peak fuel cladding temperature below the required temperature limits, thereby providing assurance of fuel cladding integrity. There is no credible cause for 100 % fuel rod failure. This accident is postulated in accordance with NUREG-2224 [4].

12.2.4.2 100 % Fuel Rod Failure Analysis

According to NUREG-2224, the two separate cases accident fire conditions (HAC-fire) and accident impact conditions (HAC-impact) are to be analysed. The fraction of fission gas released is 15 % for HAC-fire and 35 % for HAC-impact, [redacted]

[redacted] The amount of fuel rod filling gas released is calculated in the appendix of Chapter 7.

Structural

The pressurisation of the containment as a result of 100 % fuel rod failure is analyzed in Section 7.4. The maximum absolute pressure is [redacted] in the canister (HAC-impact) and [redacted] in the storage cask (HAC-fire). The applied pressure values in Chapter 3 bound the pressure calculated assuming 100% fuel rod rupture. Structural integrity of the canister and the storage cask are neither impaired for HAC-fire nor for HAC-impact.

Thermal

The 100 % fuel rod failure has no effect on the thermal performance of the DSS. The maximum temperatures of the DSS components and the content are far below the maximum admissible values with large safety margins (see Section 4.8).

Shielding

The 100 % fuel rod failure has no effect on the shielding performance of the DSS. According to Section 5.1, the ACS dose rates without fuel damage is bounding and generates the highest dose rate at the site boundary.

Criticality

The 100 % fuel rod failure has no effect on the criticality control features of the DSS.

Containment

The 100 % fuel rod failure has no effect on the containment function of the DSS. Storage cask and canister containment are not impaired for HAC-fire and HAC-impact (see Chapter 3) and both containment barriers remain leak-tight.

Radiation Protection

Since there is no degradation in shielding or containment capabilities as discussed above, there is no effect on occupational or public exposures as a result of 100 % fuel rod failure.

12.2.4.3 100 % Fuel Rod Failure Dose Calculations

The storage cask and canister containment boundaries maintain their integrity. There is no effect on the shielding effectiveness. Fuel rod failure under ACS leads to a local increase of the dose rate at the DSS surface according to Section 5.1. However, this effect vanishes with increasing distance from the DSS.

12.2.4.4 100 % Fuel Rod Failure Corrective Actions

As shown in the analysis for 100 % fuel rod failure, neither storage cask nor canister containment boundaries are damaged. No corrective actions are required.

List of References

[1] ANSI/ANS-57.9-1992
Design Criteria For An Independent Spent Fuel Storage Installation (Dry Type)
American National Standards Institute

[2] ANSI N14.6 – 1993
Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More

[3] NUREG-0612, July 1980
Control of Heavy Loads at Nuclear Power Plants
U.S. Nuclear Regulatory Commission, Office for Nuclear Reactor Regulation

[4] NUREG-2224, November 2020
Dry Storage and Transportation of High Burnup Spent Nuclear Fuel – Final Report
U.S. Nuclear Regulatory Commission

12.3 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

With intent no items.

13 Operating Controls and Limits

13.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

The CASTOR® geo69 DSS provides passive dry storage of SNF in a canister with a re-openable closure system. Canister loading under water and handling inside the reactor building is conducted via the CLU. During long-term interim storage the loaded canister is enclosed in a DPC. This chapter defines the operating controls and limits (i.e., technical specifications) including their supporting bases for loading and storage of a CASTOR® geo69 DSS and for handling operations with the CLU. The information provided in this chapter complies with the standard format and content provided in NUREG-1745 [1].

List of References

[1] NUREG-1745 (June 2001)
Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance
U.S. Nuclear Regulatory Commission

13.1 Proposed Operating Controls and Limits

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

13.1.1 Content of Operating Controls and Limits

This chapter establishes the commitments regarding the CASTOR® geo69 DSS and the CLU, as applicable, and their use. Other 10 CFR 72 and 10 CFR 20 requirements in addition to the Technical Specifications may apply. The licensee shall meet the conditions for a general license holder found in 10 CFR 72.212 prior to loading SNF into the CASTOR® geo69 DSS. The general license conditions governed by 10 CFR 72 are not repeated in this chapter. Licensees are required to comply with all commitments and requirements.

13.1.2 Bases for Operating Controls and Limits

The operating controls and limits are primarily established to maintain subcriticality, containment boundary and FA cladding integrity, shielding and radiation protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions (DSS) as well as during handling operations (CLU). Table 13.1-1 gives an overview of the conditions to be controlled, the required technical specifications and the chapters where the basis for a technical specification is discussed.

Table 13.1-1: Conditions to be controlled in the CASTOR® geo69 DSS and CLU

Condition to be controlled	*Required technical specification*	*Chapter*
Subcriticality	• Fuel spacers • Fuel cell pitch • Minimum neutron absorber ^{10}B loading • Maximum amount of fissile material (maximum initial fuel enrichment)	6
Containment boundary and FA cladding integrity	• Canister containment requirements • Inert atmosphere in canister (quantity of helium and residual moisture) • Helium leak rate limit	7
Shielding and radiation protection	• Transfer cask water chamber filling • CLU sequence of operations • Surface contamination limits • Fuel cool-down	5, 11
Heat removal	• Maximum heat designed to be dissipated • Storage cask array and spacing limits • Fuel cool-down	4
Structural integrity	• Fabrication and design codes • Design basis earthquake • Handling and lifting devices	2, 3

13.2 Development of Operating Controls and Limits

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

13.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits specified in this subsection apply to operating parameters and conditions that are observable, detectable, and/or measurable. The CASTOR® geo69 DSS and the CLU are passively cooled. During long-term interim dry storage, the DSS is equipped with a pressure switch that is connected to a pressure monitoring system. Other monitoring instruments are not required. The transfer cask is essentially passive during normal canister handling operations regarding its safety functions (structural integrity, shielding and heat removal). However, remote handling is foreseen for flooding and unflooding the cavity and the water chambers of the transfer cask and for opening and closure of the bottom lid of the transfer cask via the transfer lock.

13.2.2 Limiting Conditions for Operation

Limiting conditions for operation specify the minimum capability or level of performance that is required to assure that CASTOR® geo69 DSS and CLU can entirely fulfil their safety functions.

13.2.2.1 Equipment

The CASTOR® geo69 DSS and its components have been analysed for specified normal, off-normal, and accident conditions of storage, including extreme environmental conditions. The CLU and its components have been analysed for handling operations. Analysis has shown in this SAR that no credible condition or event prevents the CASTOR® geo69 DSS or the CLU from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analysed event. When all equipment is loaded, tested, and placed into storage in accordance with the procedures specified in this SAR, no failure of the system to perform its safety function is expected to occur.

13.2.2.2 Technical Conditions and Characteristics

Contents shall be limited to SNF as described in Section 2.1. A change of the fuel parameters as listed in Section 2.1 requires NRC approval. The loading plan shall ensure that the maximum permitted total thermal power is not exceeded. Therefore, the content shall comply with the design features for loading of contents as summarized in Subsection 1.2.3. The height of the FA shoes shall be adjusted to the FA length to maintain the axial FA position under all credible conditions.

The CASTOR® geo69 DSS shall be stored with a minimum centre-to-centre spacing between neighbouring storage casks (pitch) of [redacted] to ensure sufficient heat removal.

13.2.3 Surveillance Specifications

Surveillance requirements during loading, unloading, and storage operations are provided in this Subsection. Operation procedures shall be performed in compliance with Chapter 9.

13.2.3.1 Water Temperature in the Closed Canister

The water in the canister shall not boil after loading in the FA storage pool. Therefore, dewatering of the canister cavity shall start not later than [redacted] after the canister lid is mounted.

Frequency: Once, during loading of the cask (procedure according to Section 9.1)

Condition		*Required Action*	*Completion Time*
Loaded canister in SNF pool; canister lid is mounted.	A.1	Lift up the transfer cask until the canister lid is accessible. Decontaminate the canister lid. AND	Immediately
	A.2	[redacted]	[redacted]
Dewatering of the canister cannot be initiated within the associated completion time.	B.1	Lower the canister back into the FA storage pool and subsequently remove the canister lid to cool the loaded FA.	Immediately

13.2.3.2 Fuel Integrity during Drying

To limit fuel cladding temperatures during vacuum drying of the canister cavity and subsequent proof of drying criteria and helium backfilling shall be completed within [redacted] after drying.

Frequency: Once, during loading of the cask (procedure according to Section 9.1)

Condition		*Required Action*	*Completion Time*
Multi-equipment installed on canister lid for drying, proof of drying criteria and helium backfilling.	A.1	Start vacuum drying operation. AND	Immediately
	A.2	Complete vacuum drying and subsequent proof of drying criteria and helium backfilling of the canister cavity.	[redacted]
Vacuum drying, proof of drying criteria and helium backfilling not completed within the associated completion time.	B.1	Establish suitable safety measures (e.g. helium flushing or water flooding of the cask cavity) to protect the system against unallowable temperatures.	Immediately

13.2.3.3 Storage Cask Cavity Pressure

The storage cask minimum internal pressure shall be maintained at the filling pressure specified in Chapter 9. The cask lid is equipped with a pressure switch, which replaces the blind flange during long-term interim dry storage. The pressure switch is connected to the pressure monitoring system of the storage facility to detect a pressure drop in the storage cask.

The reason for a pressure switch alert is either a defect of the pressure switch (e.g. rupture of membrane) or a leakage in the containment boundary of the canister or the storage cask, which both lead to a pressure drop in the storage cask.

Frequency: Permanently during storage.

Condition	***Required Action***		***Completion Time***
Pressure switch signals a pressure drop.	A.1	Inform the supervisory authority to coordinate the following procedure.	Immediately
	A.2.1 AND A.2.2	Remove the protection cover from the storage cask. AND Check leak tightness of the cask lid system.	The time schedule for the procedures following the indicated pressure drop in the storage cask shall be coordinated with the supervisory authority.
Leakage in the cask lid system.	B.1	Restore leak-tightness of cask lid system. Restore pressure and helium atmosphere in the cask.	
Cask lid system is leak-tight in accordance with ANSI N14.5.	C.1 AND C.2.1 OR C.2.2	Check pressure in the reference chamber of the pressure switch to determine whether there is a leak in the pressure switch membrane or in the containment boundary of the canister. AND Replace the defect pressure switch with a new exemplar. OR Coordinate further procedures with the supervisory authority. The storage cask internal pressure may be determined as a decision support.	

The procedure for replacement of a defect pressure switch is identical to the procedure for installation of the pressure switch according to Section 9.3.

13.2.4 Design Features

13.2.4.1 Design Features Significant to Safety

Design features significant to safety are those, which would have a significant effect on safety if altered or modified. These features require design, fabrication and operational controls. The following design features have a significant impact on subcriticality:

- Distance between neighbouring FA positions: ≥ [redacted] (thickness of fuel basket sheets)
- ^{10}B loading in the criticality control material: ≥ [redacted]
- Maximum heavy metal mass: [redacted] (fully loaded DSS)

[redacted]

[redacted] Compliance with these design features ensures subcriticality for the content described in Subsection 1.2.3.

13.2.4.2 Codes and Standards

Section 2.0 provides information on the governing codes for SSCs of the CASTOR® geo69 DSS and CLU design. A detailed listing of component items, including the material specification, is given in Section 1.2 and the appendix of Chapter 1. The applicable codes and standards for materials are given in Section 8.1. Alternative materials are specified in the parts lists of the DSS and the CLU. No other materials are permitted for the SSCs of the CASTOR® geo69 DSS and CLU.

13.2.4.3 Structural Performance

The CASTOR® geo69 DSS is designed to withstand the effects of an earthquake at the storage site without tipping. The design earthquake ground motion on the top surface of the storage facility pad shall not exceed the following combination of horizontal peak acceleration (in each of the two orthogonal acceleration directions) and vertical peak acceleration:

Horizontal acceleration: [redacted]
Vertical acceleration: [redacted]

The use of the CASTOR® DSS is limited to sites that are bounded by these peak acceleration values. Site-specific design basis earthquakes exceeding the listed permissible peak acceleration values require an additional evaluation of the DSS design with regard to the structural consequences of such an earthquake. Additional safety measures (e.g. anchoring of the DSS on the storage pad) shall be implemented to ensure that no tip-over will result from the design basis earthquake.

A maximum permitted lifting height (above ground) is not established for the CASTOR® geo69 DSS and CLU. Storage cask, transfer cask and canister shall be handled via single-failure proof handling devices, which satisfy the enhanced safety criteria of NUREG-0612 [1] and are designed in accordance with ANSI N14.6 [2].

The CASTOR® geo69 DSS is designed for dry storage in an intermediate storage facility building. The storage building shall be designed to withstand collapse from the effects of flood, fire and explosion, lightning, earthquake, tornado and tornado-generated missiles, and accidents at nearby sites.

List of References

[1] NUREG-0612, July 1980
Control of Heavy Loads at Nuclear Power Plants
U.S. Nuclear Regulatory Commission, Office for Nuclear Reactor Regulation

[2] ANSI N14.6 – 1993
Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More

13.3 Appendix

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

With intent no items.

14 Quality Assurance

14.0 Overview

	Name, Function	Date	Signature
Prepared	[redacted]	[redacted]	
Reviewed	[redacted]		

All activities related to the design, fabrication and deployment of the CASTOR® geo69 cask are performed under GNS Quality Assurance Program (QAP). An associated Quality Assurance Program Description (GNS-QAPD-001) is submitted to the NRC for approval under Docket-No. 71-0967.

The GNS QAP mainly consists of:

1. The Quality Assurance Manual (QAM II)
2. Subordinated Quality Assurance Procedures (QAM II-P)
3. Project specific Quality Project Manuals (QPM).

Activities performed by suppliers or subcontractors on behalf of GNS are governed by the suppliers / subcontractor approved quality assurance program or GNS QAP is extended to the supplier / subcontractor. The amount and type of QA oversight depends on the importance to safety of the item or service to be procured and is based on the Graded Approach, which is described in GNS QAP.