CHAPTER 5
CONFINEMENT

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5. CONFINEMENT

The confinement evaluation described in this chapter is applicable to the EOS 37PTH dry shielded canister (DSC) and the EOS 89BTH DSC.

5.1 Confinement Boundary

The EOS-37PTH and EOS-89BTH DSCs are high integrity stainless steel or duplex steel welded vessels that provide confinement of radioactive materials, encapsulate the fuel in a helium atmosphere, and provide biological shielding during DSC closure and transfer and storage operations. The DSCs are designed to maintain confinement of radioactive material within the limits of 10 CFR 72.104(a), 10 CFR 72.106(b) and 10 CFR 20 under normal, off-normal, and credible accident conditions. Chapter 3 and associated appendices conclude that the design, including the helium atmosphere within the DSC, will adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage. The design ensures that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage.

The confinement boundary is shown in Figure 5-1. The DSC cylindrical shell, the inner top cover and inner bottom cover form the confinement boundary for the spent fuel. The drain port cover, vent plug and welds are also included in the confinement boundary. The outer top cover plate is an attachment to the confinement boundary that provides bearing to help support the inner top cover plate, and is therefore subject to ASME Code Subsection NB per ASME Figure NB-1132.2-3 note 6. The outer bottom cover plate is not needed to support the inner bottom cover plate under design pressure and is not in the component support path. It is thus outside ASME Code jurisdiction per ASME Figure NB-1132.2-2 note 5 and NB-2190(b). The dimensions and material descriptions for the confinement boundary assemblies and the redundantly welded barriers are discussed in Chapter 1. The components important-to-safety are identified in Chapter 2.

5.1.1 Boundary Definition/Design Features

The cylindrical shell to bottom cover plate welds are made during fabrication of the DSCs, and are fully compliant with ASME Section III, Subsection NB. The welds between the cylindrical shell and inner top cover (including drain port cover and vent plug welds) are made after fuel loading. These welds are designed, fabricated, inspected, and tested using alternatives to the ASME code specified in Section 4.4.4 of the Technical Specifications [5-3].

Stringent design and fabrication requirements ensure that the confinement function of the DSC is maintained. The cylindrical shell and inner bottom cover are pressure tested in accordance with the ASME Code, Section III, Subarticle NB-6300. This pressure test is performed after installation of the inner bottom cover at the fabricator’s facility and may be performed concurrently with the leak test, provided the requirements of NB-6300 are met.

A leak test of the shell assembly, including the inner bottom cover, is performed in accordance with ANSI N14.5 [5-1] and the ASME Code, Section V, Article 10. These tests are typically performed at the fabricator’s facility. The acceptance criteria for the test are “leaktight” as defined in ANSI N14.5.

The process for leak testing the DSC involves temporarily sealing the shell from the top end. The gas-filled envelope and evacuated envelope testing methodologies have the required nominal test sensitivity for leaktight construction and are used for leak testing. A helium mass spectrometer is used to detect any leakage as defined in ANSI N14.5. During final drying and sealing operations of the DSC, the top closure confinement welds are applied to confine radioactive materials within the cavity.

The inner top cover weld is welded to the DSC shell using automated welding equipment. Once the DSC has been vacuum dried, a pressure test is performed by backfilling the DSC cavity with helium. Following the satisfactory completion of the pressure test, the drain port cover and vent plug are welded, and a leak test is performed to verify that the weld between the DSC shell and the inner top cover, drain port cover and vent plug meet the leaktight criteria of ANSI N14.5. The outer top cover plate is also welded in place using automated welding equipment.

5.1.2 Confinement Penetrations

All penetrations in the DSC confinement boundary are welded closed. The DSC is designed to have no credible leakage as described above.

5.1.3 Seals and Welds

The welds made during fabrication of the DSC that affect the confinement boundary include the weld applied to the shell bottom, and the circumferential and longitudinal seam welds applied to the cylindrical shell. These welds are inspected (radiographic or ultrasonic inspection, and liquid penetrant inspection (PT)) according to the requirements of Subsection NB of the ASME Code.

The welds applied to the drain port cover, vent plug, and the inner top cover during closure operations define the confinement boundary at the top end of the DSC. These welds are applied using a multiple-layer technique with multi-level PT in accordance with alternatives to the ASME code as specified in Section 4.4.4 of the Technical Specifications [5-3]. This effectively eliminates any pinhole leak that might occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. Figure 5-1 provides a graphic representation of the confinement boundaries and welds.

5.1.4 Closure

Because the DSC is closed entirely by welding, there are no closure devices utilized for confinement.

5.2 Design Criteria

5.2.1 Requirements for Normal Conditions of Storage

The DSC shell is designed to prevent the leakage of radioactive materials. No discernable, undetected leakage is credible and the dose at the controlled area boundary from atmospheric release is negligible.

5.2.1.1 Release of Radioactive Material

Because the DSC is designed to have no credible leakage, Revision 1 of NUREG 1536 [5-2] does not require analyses for determining the annual dose equivalent from releases of radioactive material to an individual located at the site boundary or outside the controlled area. Analyses required for determining the annual dose equivalent based on direct radiation for normal, off-normal, and accident conditions are discussed in Chapters 11 and 12.

5.2.1.2 Pressurization of Confinement Vessel

The design provides for drying and evacuation of the DSC interior as part of the loading operations. As discussed in Chapter 4, the design is acceptable for the pressures that may be experienced during these operations. On completion of fuel loading, the gas fill of the DSC interior is at a pressure level that will maintain a non-reactive environment for at least the 80-year storage life of the DSC interior under normal, off-normal, and accident conditions.

5.2.2 Confinement Requirements for Hypothetical Accident Conditions

5.2.2.1 Fission Gas Products

The DSC confinement boundary is designed to prevent the leakage of radioactive materials. The analyses presented in Chapters 3 and 12 demonstrate that the confinement boundary is not compromised following hypothetical accident conditions. Therefore, estimating the maximum quantity of fission gas products is not necessary in accordance with Revision 1 of NUREG 1536.

5.2.2.2 Release of Contents

The DSC confinement boundary is designed to prevent the leakage of radioactive materials. The analyses presented in Chapters 3 and 12 demonstrate that the confinement boundary is not compromised following hypothetical accident conditions. End and corner drops are not considered credible events during storage and transfer. However, the DSC and EOS-TC have been evaluated for these drops to support evaluations required for postulated events under 10 CFR Part 50 and 10 CFR Part 71. The cladding integrity must be demonstrated by the user for 10 CFR Part 50 postulated end drops and will be evaluated in the 10 CFR Part 71 transport safety analysis report for hypothetical accidents during transport. Therefore, confinement analyses for the release of radioactive materials are not necessary in accordance with Revision 1 of NUREG 1536.

5.2.2.3 Confinement Monitoring Capability

The NUHOMS® EOS System is a self-contained, passive system that does not produce routine, solid, liquid or gaseous effluents. Effluent processing systems, or monitoring for airborne or liquid radioactivity, are not required to protect personnel or the environment during storage conditions. Since the DSC is closed entirely by welding, a closure monitoring system is not utilized in accordance with Revision 1 of NUREG 1536.

5.3 References

5-1 ANSI N14.5, "Leakage Tests on Packages for Shipment of Radioactive Materials," 1997.

5-2 NUREG-1536, "Standard Review Plan for Spent Fuel Dry Cask Storage Systems at a General License Facility," Revision 1, U.S. Nuclear Regulatory Commission, July 2010.

5-3 Proposed CoC 1042 Appendix A, NUHOMS® EOS System Generic Technical Specifications, Amendment 0.

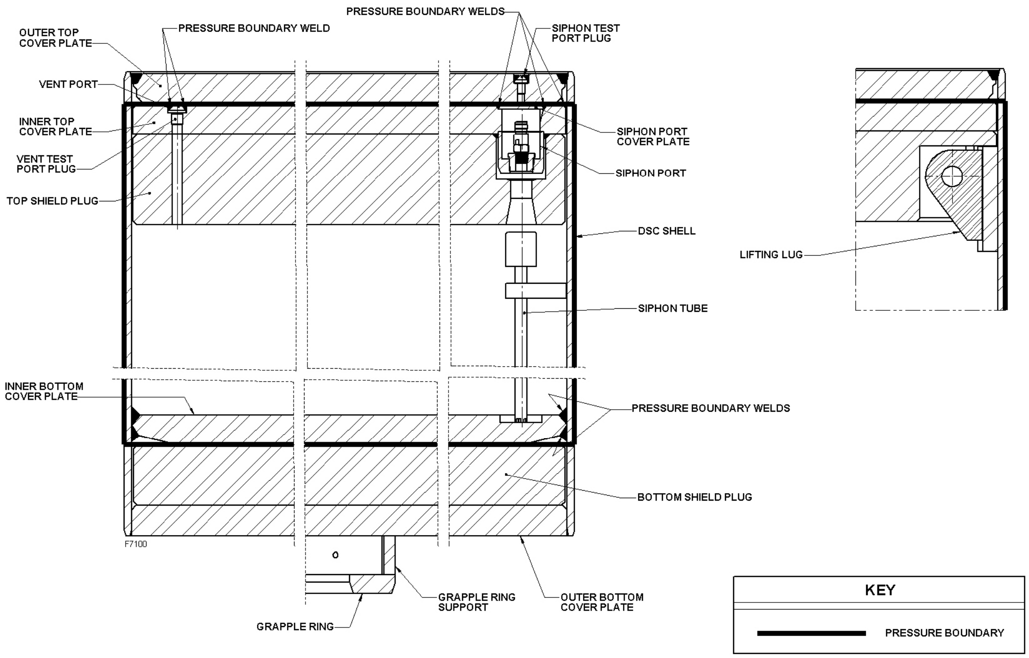

Figure 5-1
DSC Confinement Boundaries and Welds

CHAPTER 6
SHIELDING EVALUATION

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6. SHIELDING EVALUATION

The EOS system is designed to store intact pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assemblies (FAs) within the EOS-37PTH dry shielded canister (DSC) and EOS-89BTH DSC, respectively. The transfer casks (TCs) EOS-TC108 and EOS-TC125/135 are used to transfer the EOS-DSC to the EOS horizontal storage module (EOS-HSM). Normal and off-normal condition, near-field dose rates are presented in this chapter for the EOS-TC and EOS-HSM. Detailed three-dimensional dose rate calculations are performed to determine the dose rate fields around the EOS-TCs during loading, decontamination, welding, drying, and transfer operations. Detailed three-dimensional dose rate calculations are also performed to determine the dose rate fields around an EOS-HSM. These near-field dose rates are used as input to the dose assessment documented in Chapter 11, Radiation Protection.

The methodology, source terms, and dose rates presented in this chapter are developed to be reasonably bounding for general licensee implementation of the EOS System. These results may be used in lieu of near-field calculations by the general licensee, although the inputs utilized in this chapter should be evaluated for applicability by each site. Site-specific EOS-TC and EOS-HSM near-field calculations may be performed by the general licensee to modify key input parameters.

Compliance with 10 CFR 72.106 is demonstrated in this chapter for a loss of neutron shield accident for a single EOS-TC. Further, site dose calculations for an array of EOS-HSMs under normal, off-normal, and accident conditions are documented in Chapter 11, based on the near-field EOS-HSM results presented in this chapter. Because the number and arrangement of EOS-HSMs and the distance to the site boundary is site-specific, compliance with 10 CFR 72.104 and 10 CFR 72.106 for an array of EOS-HSMs can only be demonstrated using a site-specific calculation. Inputs for the site dose calculations developed in the current chapter may be directly used as input to a site-specific dose calculation by the general licensee.

6.1 Discussions and Results

The following is a summary of the methodology and results of the shielding analysis of the EOS system. More detailed information is presented in the body of the chapter.

The EOS-37PTH DSC stores up to 37 PWR FAs, while the EOS-89BTH stores up to 89 BWR FAs. Each EOS-DSC is configured into three heat load zones in order to optimize the system performance for both thermal and shielding considerations. The bounding heat load zoning configurations for fuel qualification are provided in Figure 6-1 and Figure 6-2 for PWR and BWR fuel, respectively. Fuel to be stored is limited by the decay heat and minimum cooling times provided with these figures.

Source Terms

The ORIGEN-ARP module of the Oak Ridge National Laboratory (ORNL) SCALE6.0 code package [6-1] is used to develop reasonably bounding gamma and neutron source terms. [

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Control components (CCs) are allowed to be stored within a PWR FA. Examples of CCs include burnable poison rod assemblies (BPRAs) and thimble plug assemblies. Control components typically have a Co-60 source because of its light element activation, which contributes substantially to the dose rates. The CC source term is provided in Table 6-37. CCs should be limited as follows:

- Zones 1, 2, and 3: 308 Ci Co-60 per CC in the active fuel region
- Zones 1 and 2: 63.0 Ci Co-60 per CC in the combined plenum/top region
- Zone 3: 24.3 Ci Co-60 per CC in the combined plenum/top region

BWR fuel does not include CCs other than the fuel channel, which is conservatively included in the source term. The BWR fuel channel is fabricated from zirconium alloy and does not require a Co-60 limit because the contribution to the source term from the fuel channel is negligible.

Dose Rates

The Monte Carlo transport code, MCNP5 [6-5], is used to compute dose fields around the EOS-TCs and EOS-HSM using detailed three-dimensional models for the following normal configurations:

- EOS-37PTH DSC inside the EOS-TC108
- EOS-37PTH DSC inside the EOS-TC125/135
- EOS-37PTH DSC inside the EOS-HSM-Short
- EOS-89BTH DSC inside the EOS-TC108
- EOS-89BTH DSC inside the EOS-TC125/135
- EOS-89BTH DSC inside the EOS-HSM-Medium

The EOS-TC125 and EOS-TC135 provide equivalent shielding, but accommodate different DSC lengths. The EOS-TC135 is used only with the EOS-37PTH DSC. The EOS-TC125 and EOS-TC135 designs are bounded by the same Monte Carlo N-particle (MCNP) model and are referred to in this chapter as EOS-TC125/135. The EOS-TC108 offers less shielding than the EOS-TC125/135 and features a removable neutron shield. The neutron shield is removed for fuel loading and attached subsequent to fuel loading. The neutron shield for the EOS-TC125/135 is integral to the cask and cannot be removed.

The EOS-37PTH and EOS-89BTH DSCs are custom-built for the fuel to be stored and, therefore, do not have a standard length. BWR fuel is typically longer than PWR fuel, so the EOS-89BTH DSC is longer than the EOS-37PTH DSC in the MCNP models. To accommodate the various DSC lengths, three versions of the EOS-HSM are available: short, medium, and long. In the EOS-HSM models, the EOS-37PTH DSC is paired with the EOS-HSM-Short, while the EOS-89BTH DSC is paired with the EOS-HSM-Medium, as these are the smallest EOS-HSMs that can accommodate the modeled EOS-DSCs.

All EOS-37PTH DSC calculations conservatively include both the FA and CC sources. BWR fuel does not include CC, other than the fuel channel, which is conservatively included in the source term.

Based on the near-field dose rates calculated for the EOS-TCs, a dose assessment is performed for the EOS-TC loading operation. This dose assessment is documented in Chapter 11.

The shielding effectiveness of the EOS-TC and EOS-HSM is not impacted by any off-normal events. Two accident events have been identified:

- Loss of neutron shielding for the EOS-TCs
- Loss of EOS-HSM outlet vent covers due to a tornado or missile event

MCNP cases are developed for the EOS-HSM in which the vent covers are absent. The EOS-HSM accident increases the average dose rate on the roof of the module to 7400 mrem/hr. The fluxes and dose rates on the surface of the EOS-HSM in an accident condition are used as input to an accident site dose calculation documented in Chapter 11.

6.2 Source Specification

Design basis source terms for PWR and BWR fuels are developed in this section. The source terms are developed to be reasonably bounding consistent with the limits on fuel qualification. A site-specific analysis must evaluate the site-specific used fuel to be stored and determine if the parameters utilized in the FSAR analysis are bounding and appropriate. Site-specific source terms may be different than the source terms presented herein. However, the source terms presented in this chapter were developed to bound most used fuels and will result in reasonably bounding dose rates.

Fuel types that are authorized for storage are provided in Chapter 2. These fuel types may be divided into PWR and BWR fuel types. The list of authorized fuels is summarized below.

PWR

- Westinghouse (WE) 14x14 class
- WE 15x15 class
- WE 17x17 class
- Babcock & Wilcox (B&W) 15x15 class
- Combustion Engineering (CE) 14x14 class
- CE 15x15 class
- CE 16x16 class

BWR

- 7x7 lattice array type
- 8x8 lattice array type
- 9x9 lattice array type
- 10x10 lattice array type

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6.2.1 Computer Programs

Source terms are generated using the ORIGEN-ARP module of SCALE6.0. ORIGEN-ARP is a control module for the ORIGEN-S computer program. ORIGEN-ARP allows a simplified input description that can rapidly compute source terms and decay heat compared to a full two-dimensional SCALE6.0/TRITON calculation.

Prior to using ORIGEN-ARP, detailed two-dimensional models of the design basis PWR and BWR FAs are developed in TRITON using the FA design data in Chapter 2. TRITON is used to generate ORIGEN-ARP data libraries as a function of burnup and enrichment. These libraries are collapsed from the ENDF/B-VII 238-group cross section library and are used by ORIGEN-ARP to compute the source terms.

ORIGEN-ARP uses interpolated cross section libraries to generate source terms that are essentially equivalent to the detailed TRITON runs. TRITON has been benchmarked against experimentally measured isotopes and results in excellent agreement with the measured data in ORNL/TM-2010 SCALE 5.1 [6-2]. As part of the code validation, the TRITON benchmark cases from SCALE 5.1 are rerun using the ENDF/B-VII 238-group cross section library. The isotopes important for shielding for which benchmark data are available include Cs-137/Ba-137m, Cs-134, Eu-154, Ce-144/Pr-144, Ru-106/Rh-106, Sr-90/Y-90, and Cm-244. The average ratio of the measured to calculated concentration for these nuclides is close to unity, indicating that TRITON/ORIGEN-ARP is an acceptable program for source term generation.

6.2.2 PWR and BWR Source Terms

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Sources are developed for a variety of different enrichments. For a particular U-235 enrichment, the uranium fuel loading is distributed according to the following relationship from the SCALE 6.0 manual:

- wt. % U-234 = 0.0089 * wt. % U-235
- wt. % U-236 = 0.0046 * wt. % U-235
- wt. % U-238 = 100 – wt. % U-234 – wt. % U-235 – wt. % U-236

The EOS-DSC baskets are zoned by heat load. Heat load zoning allows hotter FAs, which generally have larger neutron and gamma source terms, to be placed in the inner zones and be shielded by FAs in the outer zone. The heat load zoning configurations conservative for shielding analysis for the EOS-37PTH and EOS-89BTH DSCs are shown in Figure 6-1 and Figure 6-2, respectively. The EOS-TC108 and EOS-TC125/135 have different heat load zone configurations because the EOS-TC125/135 is more heavily shielded than the EOS-TC108 and can therefore be loaded with stronger sources.

Because the FAs are zoned by heat load, it is necessary to develop source terms for each zone. Candidate sources are developed for high burnup (62 GWd/MTU), medium burnup (50 GWd/MTU) and lower burnup (40 GWd/MTU) fuel. Cooling time is selected so that the decay heat meets or exceeds the heat load limit for each zone. Because the cooling time required at these burnups is generally much larger than the minimum allowed cooling time for each zone, the burnup that results in a cooling time that matches the minimum cooling time for each zone is also determined. From these four candidate burnup/cooling time combinations, a bounding source for each zone is selected.

The EOS-TC108 and EOS-TC125/135 feature reduced lead thickness next to the top nozzle region of the fuel assembly. For this reason, the maximum dose rate at the side of the EOS-TC occurs next to the top nozzle rather than the active fuel. The dose rate at this location is due almost entirely to Co-60 in the top nozzle and plenum regions of the FA. Therefore, to be conservative, the burnup/enrichment/cooling time combination that maximizes Co-60 activity is used to develop the top nozzle, plenum, and bottom nozzle sources. The computed Co-60 activity for each burnup/enrichment/cooling time is provided in the last column of Table 6-8 and Table 6-9 and represents the total Co-60 present in the FA. These Co-60 activities are only used for ranking the sources. For the active fuel region, the burnup/enrichment/cooling time combination that maximizes dose rate next to the active fuel is used to develop the active fuel region sources. The final “hybrid” sources are very conservative because the hardware is integral to the FA and the hardware and active fuel cannot be at different burnups/cooling times.

Based on EOS-TC and EOS-HSM dose rates (for the active fuel) and Co-60 activity (for the end hardware), reasonably bounding burnup/enrichment/cooling time combinations are determined. For these burnup/enrichment/cooling time combinations, the sources in the bottom nozzle, active fuel, plenum, and top nozzle are computed using the appropriate light elements from Table 6-5 and Table 6-6.

During an EOS-TC accident, it is postulated that the water in the neutron shield is lost. In this scenario, there is no hydrogenous neutron shield and the neutron dose rate dominates the primary gamma dose rate. Therefore, the highest allowed burnup (62 GWd/MTU) is used in accident calculations with no neutron shield because the neutron source is maximized for high-burnup fuel. In many cases, the normal condition and accident condition sources are the same.

PWR source terms are reported in the following tables:

- PWR sources terms for EOS-TC108: Table 6-10 through Table 6-13
- PWR source terms for EOS-TC125/135: Table 6-14 through Table 6-16
- PWR source terms for EOS-HSM: Table 6-17 through Table 6-19

BWR source terms are reported in the following tables:

- BWR sources terms for EOS-TC108: Table 6-20 through Table 6-22
- BWR source terms for EOS-TC125/135: Table 6-23 through Table 6-26
- BWR source terms for EOS-HSM: Table 6-27 through Table 6-29

In these tables, the "raw" neutron source computed by ORIGEN-ARP is provided, as well as neutron sources that include neutron peaking factors and subcritical neutron multiplication. These factors are derived in Section 6.2.3. The scaled neutron sources are used in the detailed MCNP dose rate calculations. Only the total neutron source magnitude is reported because the Cm-244 spectrum is used in all dose rate calculations for simplicity because the neutron source is almost entirely due to Cm-244 decay. For example, for the 62 GWd/MTU, 10.25 year cooled PWR source, 95% of the neutron source is due to spontaneous fission of Cm-244. Cm-244 is also the dominant neutron source for shorter cooling times. For instance, for a 36.178 GWd/MTU, three-year cooled PWR source, Cm-244 represents 97% of the total neutron source. The effect on the neutron spectrum of neutron source isotopes with shorter half-lives, such as Cm-242 and Cf-252, is negligible.

6.2.3 Axial Source Distributions and Subcritical Neutron Multiplication

ORIGEN-ARP is used to compute source terms for the average assembly burnup. However, an FA will exhibit an axial burnup profile in which the fuel is more highly burned near the axial center of the fuel assembly and less burned near the ends. This axial burnup profile must be taken into account when performing dose rate calculations, as the dose rate will typically peak near the maximum of this distribution.

The PWR axial burnup profile is taken from NUREG/CR-6801 [6-6] for fuel in the burnup range 26-30 GWd/MTU and is provided in Table 6-30. As fuel is more highly peaked for lower burnups, this distribution is more conservative than a flatter high-burnup distribution. The gamma source term varies proportionally to axial burnup, while neutron source terms vary exponentially with burnup by a power of 4.0 to 4.2 [6-7]. Therefore, the burnup profile is used as the gamma axial source distribution, while the neutron axial source distribution is derived as the burnup profile raised to the power of 4.2.

The average value of the neutron source distribution is 1.215, as shown in Table 6-30. This value has a physical meaning, as it is the ratio of the total neutron source from an FA with the given axial burnup profile to an assembly with a flat burnup profile. The neutron source term as computed by ORIGEN-ARP is for a flat burnup profile (average assembly burnup). Therefore, the "raw" PWR neutron source computed by ORIGEN-ARP is scaled by the factor 1.215 to account for the burnup profile.

For clarity, both the gamma and neutron axial source distributions are renormalized to sum to 1.0, as shown in Table 6-30. When normalized in this manner, the source distribution is the fraction of the source in each axial segment. For example, the fraction of the neutron source in axial segment 10 is 0.0781, or 7.81%.

The BWR axial burnup profile is taken from [6-8] for fuel with a burnup of 40.2 GWd/MTU and is provided in Table 6-31. This distribution is highly peaked and is conservative. The BWR gamma and neutron source distributions are derived using the same method used for the PWR source distributions. The average value of the BWR neutron source distribution is 1.232, and the "raw" neutron sources computed by ORIGEN-ARP are increased by this factor to account for the burnup profile.

ORIGEN-ARP does not account for subcritical neutron multiplication. Subcritical neutron multiplication is taken into account by multiplying the neutron source by 1/(1-k), where k is the multiplication factor for the system. When the system is dry, k is low due to the lack of moderation as well as burnup of the fuel. For dry analysis, k is assumed to be 0.40. When the system is wet, such as during decontamination of the EOS-TC, k is larger. For wet analysis, k is assumed to be 0.65. This value of k is reasonable for shielding calculations because the fuel is burned and heavily poisoned. Fresh or lightly burned fuel would have a higher value for k, but fresh or lightly burned FAs have a small neutron source. Because the neutron source increases proportional to the 4.2 power of the burnup, large neutron sources occur only at high burnups, and for such burnups a k of 0.65 is reasonable.

The effect on the neutron source of both the axial source distribution and subcritical neutron multiplication are combined, as shown in the source term tables (Table 6-10 and Table 6-29). For PWR sources, the "raw" ORIGEN-ARP neutron sources are scaled by 1.215/(1-0.40) = 2.025 for dry analysis and 1.215/(1-0.65) = 3.471 for wet analysis. For BWR sources, the "raw" ORIGEN-ARP neutron sources are scaled by 1.232/(1-0.40) = 2.053 for dry analysis and 1.232/(1-0.65) = 3.520 for wet analysis.

The only analysis that uses the wet neutron source term is the loading/decontamination stage of the EOS-TCs. After loading/decontamination the EOS-TCs are modeled as dry. No wet neutron sources are provided in the EOS-HSM source term tables because the DSC is always dry when inside the EOS-HSM.

6.2.4 Control Components

Control components may also be included with the PWR FAs. For BWR fuel, the fuel channel and associated attachment hardware is included in the BWR source presented in Section 6.2.2, so it will not be discussed in this section. While CCs do not contain fuel, these items result in a source term, primarily due to activation of the Co-59 impurity in the metal. Allowed CCs are identified in section 2.1 of the Technical Specifications [6-11].

Any other CC type is acceptable if it can be demonstrated that the source term is bounded by the source terms presented in this analysis. Also, the total as-loaded decay heat of the system, including CCs, must be less that the heat load zoning configurations defined in Figure 6-1.

Control components may be grouped into two categories: (1) those that extend into the top, plenum and active fuel regions of the fuel assembly, and (2) those that essentially extend only into the top and plenum regions of the FA. The BPRA is used as a representative CC for category (1) and the TPA is used as a representative CC for category (2). The objective is to use these representative CC types to develop Co-60 activity limits for CCs.

The BPRAs are assumed to be burned in two cycles to a total host FA burnup of 50 GWd/MTU. This represents a limiting burnup because the absorber material is completely depleted for this burnup. TPAs do not contain burnable poisons and may be used in multiple host FAs for very long burnups. A cumulative host FA burnup of 300 GWd/MTU is assumed. However, a TPA is primarily located in the top nozzle and plenum region of the core where the flux is depressed and the "effective" burnup of a TPA is significantly less.

A neutron source may be included in CCs, such as an NSA. Typically, the neutron source from an NSA is negligible compared to the neutron source from spent fuel. However, some neutron sources could have comparable source strength relative to the fuel assemblies. For this purpose, the loading of neutron sources is limited to the interior locations of the EOS-37PTH basket to maximize self shielding; i. e., neutron sources can be loaded in Zone 1 locations only (13 locations per Figure 1).

Representative BPRA hardware masses are available for three BPRA types:

- B&W 15x15
- WE 17x17 Pyrex
- WE 17x17 WABA

The BPRA hardware masses are provided in Table 6-32.

Representative TPA hardware masses are available for three TPA hardware types:

- Westinghouse 17x17
- Westinghouse 14x14 Type 1 and 2

The TPA hardware masses are provided in Table 6-33.

Elemental compositions for Zircaloy-4, Inconel-718, Inconel X-750, and 304 stainless steel are provided in Table 6-3. Note that the source term and dose rate are driven by Co-60, which arises primarily from Co-59 activation and to a much lesser extent from Ni-60 activation via an (n,p) reaction. The remaining light elements have little effect on the source term at the decay times of interest.

The poison is assumed to be Pyrex® (borosilicate glass). The choice of poison material has little effect on the source term or decay heat and is included for completeness. The elemental composition is obtained from [6-9] and is reproduced in Table 6-34.

The plenum and top regions are outside the active core and experience a reduced flux. The ratio of the flux in each region to the active fuel flux is provided in Table 6-4.

The source term and decay heat for the decay times of interest are dominated by Co-60. Co-60 primarily arises through activation of the Co-59 impurity present in the metal. Therefore, the BPRA and TPA hardware that has the largest Co-59 mass in each region is used to prepare the light element inputs. For the BPRA, B&W 15x15 is used for the top and WE 17x17 Pyrex is used for the plenum and active fuel regions. For the TPA, the WE 17x17 is used for all regions.

The source terms are computed using ORIGEN-ARP and the B&W 15x15 library. A separate ORIGEN-ARP input file is developed for each hardware type and region.

For the BPRA, the host FA is burned to 50 GWd/MTU in two cycles. The minimum enrichment is 3.1% based on Table 6-7. The FA loading is 0.492 MTU. The assembly power is 19.68 MW, the irradiation time per cycle is 625 days, and the down time between cycles is 30 days. Decay heat, Co-60 activity, and the gamma source term is requested for a decay time of 10 years.

To account for the reduced flux in the plenum and top regions, the BPRA input masses are scaled by the appropriate flux scaling factor. The ORIGEN-ARP inputs for the three BPRA regions are summarized in Table 6-35.

The methodology for TPAs is slightly different than for BPRAs. The reason is that a TPA may reside in several host FAs for a total host fuel assembly burnup of 300 GWd/MTU. ORIGEN-ARP cannot burn a single FA to such a high burnup. Therefore, rather than apply the flux scaling factors to the input masses, the true masses are input and the flux scaling factors are applied to the FA burnup. The TPA input masses are summarized in Table 6-33. These masses do not include flux scaling factors and are therefore larger than the BPRA input masses.

For the TPA plenum, the effective burnup is 300*0.2 = 60 GWd/MTU, while for the TPA top the effective burnup is 300*0.1 = 30 GWd/MTU. This reduces the cumulative burnup in each region to a value within the bounds of a typical ORIGEN-ARP model.

The TPA irradiation time is input to match the true irradiation time to properly credit Co-60 decay during the irradiation. Assuming a reactor assembly power of 19.68 MW and fuel loading of 0.492 MTU, the irradiation time to achieve a cumulative fuel assembly burnup of 300,000 GWd/MTU is 7,500 days. Because the irradiation time is fixed at 7,500 days, the FA power is selected to give the desired effective burnup in the plenum and top regions. For the top, the assembly power is 1.968 MW to achieve an effective burnup of 30 GWd/MTU. For the plenum, the assembly power is 3.936 MW to achieve an effective burnup of 60 GWd/MTU.

For simplicity of input preparation in the TPA calculation, no credit is taken for down time between cycles (typically assumed to be 30 days). Using approximately 12 cycles to achieve a burnup of 300 GWd/MTU, the conservatism of this assumption is 11*30 = 330 days of uncredited decay time.

Results for Co-60 activity and decay heat for both the BPRA and TPA are summarized in Table 6-36 for a cooling time of 10 years. It is observed that the BPRA source may be used in the active fuel region, as the TPA does not extend into this region. However, the TPA has a larger source than the BPRA in the plenum and top regions due to the high TPA burnup. Decay heat for both is small compared to SFA but must be accounted for during loading. The CC source used in the detailed PWR dose rate calculations is a hybrid CC source that combines the active fuel source of the BPRA with the top/plenum source of the TPA in Zones 1 and 2, but limits Zone 3 to the lower BPRA source. This source is provided in Table 6-37.

The CC source significantly impacts the peak dose rates on the side of the EOS-TC, due to the reduced lead thickness near the top nozzle. Site-specific calculations by the general licensee of CC source terms should limit the computed Co-60 activity to values bounded by the results of this analysis, as follows:

- Zones 1, 2, and 3: 308 Ci Co-60 per CC in the active fuel region
- Zones 1 and 2: 63.0 Ci Co-60 per CC in the combined plenum/top region
- Zone 3: 24.3 Ci Co-60 per CC in the combined plenum/top region

While the specific CC source term presented in Table 6-37 is computed for a decay time of 10 years, this is not a minimum decay time requirement for licensing purposes. The actual CC to be loaded may have a shorter decay time as long as the as-loaded Co-60 activity is less than the limits provided above, and the total EOS-DSC decay heat remains below the applicable limit.

6.2.5 Blended Low Enriched Uranium Fuel

6.2.6 Reconstituted Fuel

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6.2.7 Irradiation Gases

During irradiation in a reactor, a FA will generate gases due to fission, alpha decay, and light element activation. The moles of gas generated are needed for subsequent pressure calculations documented in Chapter 4, Section 4.7, and are computed using ORIGEN-ARP. The noble gases (He, Ne, Ar, Kr, Xe, and Rn) are of primary interest as these gases do not react with other elements. The elements H, N, F, and Cl are conservatively assumed to be present in a gaseous state, although these elements may have formed solid compounds and may not be present as a gas. Bromine and iodine are also assumed to be present as a gas because the boiling points of these elements are low. Oxygen is not treated as a gas because it is present primarily in the compound UO_2.

The quantities of irradiation gases increase with burnup. Therefore, the quantity of gas is maximized for a burnup of 62 GWd/MTU. Most of the gases generated are stable isotopes. However, due to alpha decay of actinides present in spent fuel, the quantity of helium slowly increases with time. To obtain a bounding value for helium buildup due to alpha decay, 100 years of decay is assumed.

Integral fuel burnable absorber rods (IFBA) are used in some Westinghouse PWR designs. IFBA contains B-10, which results in helium gas generation due to the reaction B-10 + n → Li-7 + He-4. While the design basis B&W 15x15 FA does not contain IFBA, the effect of an IFBA FA is conservatively included by adding 450 g boron to the PWR input file.

Control components also may result in helium gas generation, primarily due to B-10 activation. No actinides or fission products are present in the CCs, so the quantity of gas is smaller than spent fuel. Because the BPRA contains boron while the TPA does not, the BPRA bounds the TPA for gas generation. BPRA data is summarized in Table 6-32. The B&W 15x15 BPRA contains poison in the form B_4C-Al_2O_3, typically up to 5% B_4C, while the WE 17x17 Pyrex design utilizes Pyrex poison. To conservatively bound these designs and potentially other designs, the boron mass is input as 450 g.

Irradiation gases are computed for (1) the design basis PWR fuel including CCs (without IFBA), (2) the design basis PWR fuel with IFBA (without CCs), and (3) the design basis BWR fuel. The moles of each isotope of interest are reported in Table 6-40 for a decay time of 100 years. The PWR FA with CCs (without IFBA) has 59.0 moles of gas (49.4 moles from the FA and 9.6 moles from CCs). For a PWR fuel assembly with IFBA (without CCs), there are 57.7 moles of gas. Therefore, the bounding PWR value is 59.0 moles of gas. For BWR fuel, the total gas is 20.6 moles.

Due to the helium generated from B-10 activation, the following restrictions are in effect for PWR fuel in order to limit the total moles of gas generated:

- If a FA does not contain IFBA, it may include any CC type.

- If a FA contains IFBA, it cannot include CCs that had an initial boron loading (e.g., BPRAs). Control components that do not contain initial boron (e.g., TPAs) may be included with an IFBA assembly.

The quantity of fission gas generated in a FA is proportional to the fuel loading. The moles of gas for the design basis PWR FA are based on a fuel loading of 0.492 MTU. However, there are shorter FAs with smaller fuel loadings and longer FAs with larger fuel loadings. The EOS-DSC may be shorter or longer depending on the length of fuel, and thus the free volume within the DSC changes with fuel length/loading. For the pressure calculation (Section 4.7), FAs are binned into short, medium, and long groups.

The short group has an unirradiated FA length < 157 inches. The medium group has an unirradiated FA length between 157 and 190 inches, while the long group has an unirradiated FA length > 190 inches. The design basis PWR fuel has an unirradiated fuel length of 165.76 inches, which places it in the medium group.

For the pressure calculation the medium length FAs bound the long fuel assemblies. There are three short PWR FAs, CE 14x14 Fort Calhoun, CE 15x15 Palisades, and Exxon/ANF 15x15 CE. The maximum fuel loading for the three short FAs is 0.450 MTU. Therefore, the irradiation gas result for the design basis assembly (49.4 moles) may be scaled by 0.450/0.492 = 0.915. The gas from CCs (9.6 moles) is conservatively assumed to be unchanged for the shorter FAs. The bounding quantity of gas for the short PWR assemblies is then 49.4 moles*0.915 + 9.6 moles = 54.8 moles.

Note that the moles of gas presented are only gases generated due to irradiation. Both fuel and CCs will be pre-pressurized with gas (typically helium) when fabricated and the moles of this initial gas is not included.

6.3 Model Specification

MCNP5 is used to perform detailed three-dimensional near-field dose rate calculations for EOS-TCs and EOS-HSMs. All relevant details of the EOS-37PTH DSC, EOS-89BTH DSC, EOS-TC108, EOS-TC125/135, and EOS-HSM are modeled explicitly.

Separate primary gamma and neutron models are developed. The EOS-TC and EOS-HSM neutron models are run in coupled neutron-photon mode so that the secondary gamma dose rate from (n,γ) reactions may be computed. The secondary gamma dose rate from the EOS-TC arises primarily from neutron absorption in the water neutron shield. Secondary gammas from the EOS-HSM are negligible but are computed for completeness.

The treatment of subcritical neutron multiplication is suppressed in MCNP by using the NONU card. This is done because the fuel assemblies are modeled as fresh fuel and homogenized for simplicity, which would cause inaccurate treatment of subcritical neutron multiplication by MCNP. Subcritical neutron multiplication is accounted for in the neutron source magnitude, as discussed in Section 6.2.3.

6.3.1 Material Properties

Basic materials used in the models, such as 304 stainless steel, carbon steel, and concrete, are obtained from PNNL-15870 [6-9] and are summarized in Table 6-41. Not all materials are used in every model. The density of concrete has been conservatively reduced to 2.243 g/cm^3. Simple materials consisting of one element are not listed in Table 6-41. Such materials include lead, which is modeled with a reduced density of 11.18 g/cm^3. Aluminum is used in the basket plates with a density of 2.7 g/cm^3. The metal matrix composite (MMC) poison is modeled as pure aluminum (no boron) with a reduced density of 2.56 g/cm^3.

Borated polyethylene is used at the bottom of the EOS-TC for neutron shielding. Approximately 16% boric acid by weight (B_2O_3) is added to polyethylene so that the material is 5% boron by weight. The atom density of hydrogen is conservatively reduced by 15% to account for potential hydrogen loss due to aging. The borated polyethylene composition used in the EOS-TC models is provided in Table 6-42.

The FAs are homogenized for simplicity. Fuel assemblies are modeled as fresh with a U-235 enrichment of 3%. The enrichment used is arbitrary because fission has been suppressed with the NONU card. Separate homogenization is performed for the bottom nozzle, active fuel, plenum, and top nozzle regions of the FA for both wet and dry conditions. The masses used for the homogenization are obtained from Table 6-1 and Table 6-2 for PWR and BWR fuel, respectively.

Table 6-1 does not include CC masses. For PWR fuel, the CC mass is also included in the plenum and bottom nozzle homogenizations because the CC source is always included in the MCNP models (no CC mass is credited in the active fuel region). As discussed in Section 6.2.4, the CC source is based on the BPRA B&W 15x15 in the top region (Zone 3), BPRA WE 17x17 Pyrex in the plenum region (Zone 3), and TPA WE 17x17 in both the top and plenum regions (Zones 1 and 2). The additional CC mass to be homogenized with the fuel is the minimum masses when comparing these CC types. This results in an additional 2.468 kg SS304 and 0.358 kg Inconel-718 in the top nozzle and an additional 2.85 kg SS304 in the plenum.

For BWR fuel, the mass of the channel is conservatively ignored because the channel may not be present. In the wet models, water with a density of 0.958 g/cm^3 fills the void space within the FA. The homogenized PWR fuel compositions are provided in Table 6-43 and Table 6-44 for dry and wet analysis, respectively. The homogenized BWR fuel compositions are provided in Table 6-45 and Table 6-46 for dry and wet analysis, respectively.

Concrete used in the EOS-HSM is modeled without steel rebar at a conservatively low density of 140 pcf (2.243 g/cm^3).

6.3.2 MCNP Model Geometry for the EOS-TC

Detailed EOS-TC MCNP models are developed for the following four configurations:

- EOS-TC108 with EOS-37PTH DSC
- EOS-TC108 with EOS-89BTH DSC
- EOS-TC125/135 with EOS-37PTH DSC
- EOS-TC125/135 with EOS-89BTH DSC

The EOS-37PTH DSC and EOS-89BTH DSC are modeled explicitly, including the steel basket structure, aluminum plates, MMC (conservatively modeled without boron), transition rails, and shield plugs. Key dimensions used to develop the DSC models are summarized in Table 6-47, and figures illustrating the basic MCNP model are provided in Figure 6-3 through Figure 6-6. The figures illustrate the EOS-TC108 with the EOS-89BTH DSC, although the other EOS-TC and EOS-DSC combinations are similar.

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Three model configurations are used to simulate the EOS System during the various stages from loading to transfer. These configurations are loading/decontamination, welding/drying, and downending/transfer, and are described below.

- Loading/Decontamination. The shield plug is in place while the ITCP, OTCP, and top cover plate (lid) are not installed. The neutron shield is off (for the EOS-TC108) or drained (for the EOS-TC125/135), simulating the configuration when the EOS-TC is first removed from the pool. (The actual decontamination operation is performed with the neutron shield full.) Due to crane weight constraints, the EOS-TC108 is modeled with the DSC cavity drained to the top of the active fuel for the EOS-37PTH DSC and drained to the top of the plenum for the EOS-89BTH DSC. The top nozzle of BWR fuel is sufficiently long that draining the plenum region is not anticipated. For the EOS-TC125/135, the DSC cavity is completely filled with water at all times because there is no constraint on crane capacity. The annulus between the EOS-DSC and EOS-TC is filled with water.
- Welding/Drying. The ITCP is installed but the OTCP and top cover plate (lid) are not installed. The neutron shield is filled with water and the DSC is dry. The water height in the annulus is reduced by 12 inches.
- Downending/Transfer. The TC is fully assembled for the transfer operation. The OTCP and top cover plate (lid) are installed. The neutron shield is filled with water and all TC cavities are dry. The EOS-TC108 is modeled with the intermediate aluminum lid rather than the final steel lid.

These three configurations are also summarized in Table 6-49.

The EOS-TC108 has a removable neutron shield. The shield is formed of three panels that are connected with hinges on two joints and latches on the third joint. The interface joint between the three panels features 1.5 inches of aluminum, which allows limited neutron streaming through these three joints along the length of the EOS-TC108. The EOS-TC108 models do not include this streaming path, i.e., the neutron shield is modeled as continuous around the circumference. However, the neutron shield joints are modeled explicitly in a supplementary model, and the dose rates in the vicinity of the joints do not exceed the reported peak dose rates. In addition, the dose rates used in the dose assessment are essentially unchanged when the neutron shield joints are modeled. Therefore, it is acceptable to model the EOS-TC108 neutron shield as continuous around the circumference.

No temporary shielding is modeled, which would be used in practice to shield penetrations or localized areas of high dose rate. Therefore, the computed dose rates are larger than the dose rates that would be observed in actual practice.

The source terms used in the EOS-37PTH DSC models are the combined fuel and CC source terms. The CC source term from Table 6-37 is simply added to the fuel source term from Table 6-10 through Table 6-16. The CC source is added to every FA in the EOS-37PTH DSC. The EOS-89BTH DSC source terms are provided in Table 6-20 through Table 6-26. Note that the source term tables provide dry and wet neutron sources. Wet neutron sources are used only in the loading/decontamination models, while dry neutron sources are used in all other models. For the active fuel regions, an axial source distribution is applied per Table 6-30 and Table 6-31 for PWR and BWR fuel, respectively. For the top nozzle, plenum, and bottom nozzle regions, the source is evenly distributed throughout the region.

For each TC/DSC combination, dose rates are calculated on the surface, 30 cm, and 100 cm from the surfaces of the EOS-TC. Dose rates are also computed 300 cm from the side surface. All side dose rates are computed in 18 axial bins. The general tally locations are shown in Figure 6-7. In addition, for the final transfer configuration, dose rates are computed on the bottom and top surface in six radial segments (see Figure 6-8 and Figure 6-9) and on the side surface in 18 radial segments and 24 angular segments (see Figure 6-10).

Accident models are also developed for the four transfer configurations. In the accident models, the water neutron shield, neutron shield panel, and borated polyethylene bottom neutron shield are replaced with void, and the accident source terms are used. The dose rate is calculated at a distance of 100 m from the EOS-TC. Ground is modeled to account for ground scatter at large distances.

6.3.3 MCNP Model Geometry for the EOS-HSM

Detailed EOS-HSM MCNP models are developed for the following two configurations:

- EOS-HSM-Short with EOS-37PTH DSC

- EOS-HSM-Medium with EOS-89BTH DSC

The EOS-37PTH DSC and EOS-89BTH DSC models developed in Section 6.3.2 are used in the EOS-HSM models. Consistent with the EOS-DSC models, the Z-axis in the EOS-HSM models is along the length of the EOS-DSC. Because the DSC cavity has been reduced in length to match the length of the fuel, the EOS-37PTH DSC model is shorter than the EOS-89BTH DSC model. Short, medium, and long versions of the EOS-HSM may be used, depending on the length of EOS-DSC to be stored. The EOS-HSM modeled is the smallest EOS-HSM that fits the EOS-DSC. Therefore, the EOS-HSM-Short is modeled with the EOS-37PTH DSC and the EOS-HSM-Medium is modeled with the EOS-89BTH DSC.

PWR source terms (without CCs) are provided in Table 6-17 through Table 6-19, and the CC source provided in Table 6-37 is added to these PWR source terms for all FAs. BWR source terms are provided in Table 6-27 through Table 6-29. For the active fuel regions, an axial source distribution is applied per Table 6-30 and Table 6-31 for PWR and BWR fuel, respectively. For the top nozzle, plenum, and bottom nozzle regions, the source is evenly distributed throughout the region.

The EOS-HSMs are modeled explicitly, including the inlet (front) and outlet (roof) vents. Key dimensions used to develop the EOS-HSM models are summarized in Table 6-50, and figures illustrating the basic MCNP model are provided in Figure 6-11 through Figure 6-13. The figures illustrate the EOS-HSM-Medium with the EOS-89BTH DSC, although the geometry of the EOS-HSM-Short with the EOS-37PTH DSC is similar.

The EOS-HSM design consists of a base module that includes the door and 1-foot thick shield walls on the sides and rear. A 3-foot-8-inch thick roof block that matches the length and width of the base rests on the base module. The modules may be positioned either side-by-side in a single row or back-to-back in a double row. When positioned in a single row the rear of the base module is shielded by a 3-foot thick rear shield wall. An end (side) shield wall, which is also 3 feet thick, is placed beside the last module in the row. The end shield wall is comprised of two pieces mated with a Z-joint to prevent direct streaming through the joint. A corner shield wall is placed at the interface of the rear and end shield walls. When the modules are positioned back-to-back, no rear or corner shield walls are used.

Air inlet vents are located on the front and air outlet vents are located on the roof. Because little radiation directly penetrates the thick concrete shielding, essentially all of the dose rate is due to gamma radiation streaming from the vents. Radiation streaming through the outlet vents is mitigated by the use of vent covers. The vent covers feature a 1-inch thick steel plate and approximately 11 inches of concrete. The vent covers are 4 feet wide and are placed between adjacent EOS-HSMs or between an EOS-HSM and the end shield wall. Under normal and off-normal conditions the vent covers are always in place.

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The baseline MCNP model consists of an EOS-HSM with a rear shield wall. On the right side (+x direction) an end shield wall is modeled, while on the left side (-x direction) a mirror boundary is modeled. The mirror boundary simulates an adjacent EOS-HSM so that the model is effectively a 2x1 array of EOS-HSMs. Modeling a 2x1 array significantly increases the vent dose rates compared to simply modeling a single EOS-HSM because a significant source of radiation at a vent is from the EOS-DSC in an adjacent EOS-HSM. The mirror boundary is placed 0.75 inch from the left face of the module to simulate a total gap of 1.5 inches. This model is referred to as the "single reflection model." Dose rates are computed at the inlet and outlet vents and at the 1.5-inch gaps between the base module and shield walls.

The average fluxes and dose rates on the faces of the EOS-HSM are used as input to a generic site dose calculation that is documented in Chapter 11. These average fluxes and dose rates are computed on the surface of a box that envelops the EOS-HSM model, including the vent covers, door, and fabrication gaps. The average end shield wall dose rates are computed with the 2x1 EOS-HSM "single reflection" model described above. However, to capture the contribution from side-by-side or back-to-back EOS-HSMs, additional "double reflection" and "triple reflection" models are developed.

In the "double reflection" model, the end shield wall and corner shield wall are removed and a reflective boundary is added on the right side. The double reflection model simulates an EOS-HSM with an adjacent EOS-HSM on each side. Gaps are included between the modules. This model is only used to compute the average fluxes and dose rate on the rear shield wall used as input to the site dose calculation.

In the "triple reflection" model, all shield walls are removed and replaced with reflective boundaries. This model is illustrated in Figure 6-14. The triple reflection model simulates an EOS-HSM with an adjacent EOS-HSM on each side and back-to-back. The triple reflection model is used to compute the average dose rates on the front and roof used as input to the site dose calculation. The triple reflection model is also used to compute vent and gap dose rates.

In an accident condition, the vent covers are assumed to be absent. This will cause the average roof dose rate to increase substantially but will have negligible effect on the front, rear, or side dose rates. A triple reflection accident model is developed to compute the average flux and dose rate on the roof when the vent covers are removed.

A simple model of an EOS-89BTH DSC with 44 inches of cylindrical concrete shielding and no vents is used to demonstrate the bulk shielding effectiveness of the EOS-HSM in the absence of penetrations. This model is illustrated in Figure 6-15. The outer three and six inches of this model is also replaced with low-density grout (100 pcf versus the 140 pcf concrete used in the EOS-HSM model) to demonstrate that concrete that has spalled may be patched with grout with a negligible impact on the dose rates.

6.4 Shielding Analysis

6.4.1 Computer Codes

MCNP5 v1.40 is used in the shielding analysis [6-5]. MCNP5 is a Monte Carlo transport program that allows full three-dimensional modeling of the EOS-TC and EOS-HSM. Therefore, no geometrical approximations are necessary when developing the shielding models.

6.4.2 Flux-to-Dose Rate Conversion

MCNP5 is used to compute the neutron or gamma flux at the location of interest and the flux is converted to a dose rate using ANSI/ANS-6.1.1-1977 flux-to-dose rate conversion factors [6-10]. These factors are provided in Table 6-51. Results are computed in the units mrem/hr.

6.4.3 EOS-TC Dose Rates

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The results are presented in Table 6-59. With no grout, the dose rate is 1.4 mrem/hr, while with 6 inches of grout, the dose rate is 2.2 mrem/hr. Therefore, the use of lower density grout to repair concrete is acceptable because the localized dose rate remains small. For example, the average dose rate on the end shield wall of the EOS-HSM is 0.544 mrem/hr (see Table 6-55), and if 6 inches of grout were used, the dose rate would increase to only 0.544*2.2/1.4 = 0.85 mrem/hr. Dose rates are dominated by streaming from the vents because little radiation penetrates the thick concrete shield walls, and repairing the EOS-HSM with grout in localized areas will have no effect on worker exposure or site dose rates.

6.5 Supplemental Information

6.5.1 References

6-1 Oak Ridge National Laboratory, "A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/TM-2005/39, Version 6, SCALE, January 2009.

6-2 Oak Ridge National Laboratory, "Predictions of PWR Spent Nuclear Fuel Isotopic Compositions," ORNL/TM-2010/44, SCALE 5.1, March 2010.

6-3 Oak Ridge National Laboratory, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Code," ORNL/TM-11018, December 1989.

6-4 Pacific Northwest Laboratory, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal, Volume 1 – Activation Measurements and Comparison with Calculations for Spent Fuel Assembly Hardware," PNL-6906, Vol. 1, June 1989.

6-5 Oak Ridge National Laboratory, "MCNP/MCNPX – Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," CCC-730, RSICC Computer Code Collection, January 2006.

6-6 NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," March 2003.

6-7 NUREG-1536, Rev. 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," July 2010.

6-8 Design Data Document DI-81001-02, NOK Document, "Technical Specification for the Supply of Transportable Casks for the Storage of Kernkraftwerk Leibstardt (KKL) Spent Fuel in ZWILAG," TS 07/01, Rev. 1.

6-9 Pacific Northwest National Laboratory, "Compendium of Material Composition Data for Radiation Transport Modeling," PNNL-15870, Rev. 1, March 2011.

6-10 ANSI/ANS-6.1.1-1977, "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," American National Standards Institute, Inc., New York, New York.

6-11 Proposed CoC 1042 Appendix A, NUHOMS® EOS System Generic Technical Specifications, Amendment 0.

Table 6-1
PWR (BW 15x15) Hardware Characteristics

Fuel Assembly Region and Length	Fuel Assembly Part	Material	Mass (kg)
Top Nozzle, 6.23 in.	Top nozzle/misc. steel	SS304	9.180
	Hold down spring	Inconel-718	1.800
Plenum, 8.73 in.	Upper spring	Inconel-718	4.344
	Upper end cap	Zircaloy-4	1.039
	Encompassing cladding	Zircaloy-4	5.763
	Upper end grid	Inconel-718	1.067
	Encompassing guide tube	Zircaloy-4	0.004
Active Fuel, 142.29 in.	Encompassing cladding	Zircaloy-4	101.1
	Encompassing guide tube	Zircaloy-4	6.328
	Six spacer grids	Inconel-718	4.985
	Grid Supports	Zircaloy-4	0.640
Bottom Nozzle, 8.38 in.	Lower end plug	Zircaloy-4	8.877
	Encompassing guide tube	Zircaloy-4	0.140
	Lower guide tube plugs	Zircaloy-4	1.439
	Lower end fitting	SS304	8.172
	Lower end grid	Inconel-718	1.067

Table 6-2
BWR (GE 7x7) Hardware Characteristics

Fuel Assembly Region and Length	Fuel Assembly Part	Material	Mass (kg)
Top Nozzle, 12.62 in.	Upper tie plate	SS304	2.08
	Lock tab washers and nuts	SS304	0.05
	Expansion springs	Inconel X-750	0.43
	End plugs	Zircaloy-2	1.26
Plenum, 12.93 in.	Cladding	Zircaloy-2	4.89
	Springs	SS304	1.05
Active Fuel, 144 in.	Cladding	Zircaloy-2	49.2
	Spacers	Zircaloy-2	1.95
	Spacer springs	Inconel X-750	0.36
	Channel sleeve	Zircaloy-2	37.1
	Channel spacer and rivet	SS304	0.13
	Channel fastener guard	SS304	0.46
	Channel fastener spring and bolt	Inconel X-750	0.13
Bottom Nozzle, 6.65 in.	Finger springs	Inconel X-750	0.05
	End plugs	Zircaloy-2	1.26
	Lower tie plate	SS304	4.70

Table 6-3
Fuel Assembly Material Compositions
2 Pages

Element	Zircaloy-4 (ppm)	Inconel-718 (ppm)	Inconel X-750 (ppm)	SS304 (ppm)	UO_2 (g/MTU)
Hydrogen	13	0	0	0	0
Lithium	0	0	0	0	1
Boron	0.33	0	0	0	1
Carbon	120	400	399	800	89.4
Nitrogen	80	1300	1300	1300	25
Oxygen	950	0	0	0	134454
Fluorine	0	0	0	0	10.7
Sodium	0	0	0	0	15
Magnesium	0	0	0	0	2
Aluminum	24	5992	7982	0	16.7
Silicon	0	1997	2993	10000	12.1
Phosphorous	0	0	0	450	35
Sulfur	35	70	70	300	0
Chlorine	0	0	0	0	5.3
Calcium	0	0	0	0	2
Titanium	20	7990	24943	0	1
Vanadium	20	0	0	0	3
Chromium	1250	189753	149660	190000	4
Manganese	20	1997	6984	20000	1.7
Iron	2250	179766	67846	688440	18
Cobalt	10	500	500	500	1
Nickel	20	519625	721861	89200	24
Copper	20	999	499	0	1
Zinc	0	0	0	0	40.3
Zirconium	979110	0	0	0	0
Niobium	0	55458	8980	0	0
Molybdenum	0	29961	0	0	10
Silver	0	0	0	0	0.1
Cadmium	0.25	0	0	0	25
Indium	0	0	0	0	2
Tin	16000	0	0	0	4
Gadolinium	0	0	0	0	2.5

Table 6-3
Fuel Assembly Material Compositions
2 Pages

Element	Zircaloy-4 (ppm)	Inconel-718 (ppm)	Inconel X-750 (ppm)	SS304 (ppm)	UO_2 (g/MTU)
Hafnium	78	0	0	0	0
Tungsten	20	0	0	0	2
Lead	0	0	0	0	1
Bismuth	0	0	0	0	0.4
Uranium	0.2	0	0	0	1000000

Table 6-4
Flux Scaling Factors

Region	PWR	BWR
Top Nozzle	0.1	0.1
Plenum	0.2	0.2
Active Fuel	1.0	1.0
Bottom Nozzle	0.2	0.15

Table 6-5
PWR Light Elements by Fuel Assembly Region (ORIGEN-ARP Input, grams)
2 Pages

Element	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle	Total
H	2.718E-02	1.405E+00	1.769E-02	0.000E+00	1.450E+00
Li	0.000E+00	4.919E-01	0.000E+00	0.000E+00	4.919E-01
B	6.901E-04	5.275E-01	4.492E-04	0.000E+00	5.287E-01
C	1.642E+00	5.894E+01	5.962E-01	8.055E-01	6.198E+01
N	2.567E+00	2.742E+01	1.516E+00	1.426E+00	3.293E+01
O	1.987E+00	6.623E+04	1.293E+00	0.000E+00	6.624E+04
F	0.000E+00	5.263E+00	0.000E+00	0.000E+00	5.263E+00
Na	0.000E+00	7.378E+00	0.000E+00	0.000E+00	7.378E+00
Mg	0.000E+00	9.837E-01	0.000E+00	0.000E+00	9.837E-01
Al	1.329E+00	4.068E+01	6.517E+00	1.079E+00	4.960E+01
Si	1.675E+01	1.591E+01	2.161E+00	9.528E+00	4.434E+01
P	7.345E-01	1.721E+01	0.000E+00	4.126E-01	1.836E+01
S	5.778E-01	4.132E+00	1.234E-01	2.876E-01	5.121E+00
Cl	0.000E+00	2.607E+00	0.000E+00	0.000E+00	2.607E+00
Ca	0.000E+00	9.837E-01	0.000E+00	0.000E+00	9.837E-01
Ti	1.747E+00	4.248E+01	8.674E+00	1.438E+00	5.434E+01
V	4.182E-02	3.637E+00	2.722E-02	0.000E+00	3.706E+00
Cr	3.532E+02	1.083E+03	2.071E+02	2.084E+02	1.852E+03
Mn	3.311E+01	1.295E+01	2.188E+00	1.870E+01	6.695E+01
Fe	1.167E+03	1.148E+03	1.976E+02	6.635E+02	3.176E+03
Co	9.448E-01	4.065E+00	5.547E-01	5.490E-01	6.114E+00
Ni	2.565E+02	2.604E+03	5.624E+02	1.753E+02	3.598E+03
Cu	2.550E-01	7.633E+00	1.108E+00	1.798E-01	9.177E+00
Zn	0.000E+00	1.982E+01	0.000E+00	0.000E+00	1.982E+01
Zr	2.047E+03	1.058E+05	1.333E+03	0.000E+00	1.092E+05
Nb	1.183E+01	2.765E+02	6.002E+01	9.982E+00	3.583E+02
Mo	6.394E+00	1.543E+02	3.242E+01	5.393E+00	1.985E+02
Ag	0.000E+00	4.919E-02	0.000E+00	0.000E+00	4.919E-02
Cd	5.228E-04	1.232E+01	3.403E-04	0.000E+00	1.232E+01
In	0.000E+00	9.837E-01	0.000E+00	0.000E+00	9.837E-01
Sn	3.346E+01	1.731E+03	2.178E+01	0.000E+00	1.787E+03
Gd	0.000E+00	1.230E+00	0.000E+00	0.000E+00	1.230E+00

Table 6-5
PWR Light Elements by Fuel Assembly Region (ORIGEN-ARP Input, grams)
2 Pages

Element	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle	Total
Hf	1.631E-01	8.430E+00	1.062E-01	0.000E+00	8.700E+00
W	4.182E-02	3.145E+00	2.722E-02	0.000E+00	3.214E+00
Pb	0.000E+00	4.919E-01	0.000E+00	0.000E+00	4.919E-01
Bi	0.000E+00	1.967E-01	0.000E+00	0.000E+00	1.967E-01

Table 6-6
BWR Light Elements by Fuel Assembly Region (ORIGEN-ARP Input, grams)
2 Pages

Element	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle	Total
H	2.457E-03	1.147E+00	1.271E-02	1.638E-03	1.164E+00
Li	0.000E+00	1.980E-01	0.000E+00	0.000E+00	1.980E-01
B	6.237E-05	2.271E-01	3.227E-04	4.158E-05	2.275E-01
C	5.889E-01	2.896E+01	2.851E-01	2.025E-01	3.003E+01
N	9.402E-01	1.341E+01	3.509E-01	3.425E-01	1.505E+01
O	1.795E-01	2.671E+04	9.291E-01	1.197E-01	2.671E+04
F	0.000E+00	2.119E+00	0.000E+00	0.000E+00	2.119E+00
Na	0.000E+00	2.970E+00	0.000E+00	0.000E+00	2.970E+00
Mg	0.000E+00	3.960E-01	0.000E+00	0.000E+00	3.960E-01
Al	6.440E-02	9.336E+00	2.347E-02	3.462E-01	9.770E+00
Si	7.063E+00	9.755E+00	2.097E+00	2.256E+00	2.117E+01
P	3.168E-01	7.195E+00	9.438E-02	9.573E-02	7.702E+00
S	2.184E-01	3.300E+00	9.715E-02	7.124E-02	3.686E+00
Cl	0.000E+00	1.049E+00	0.000E+00	0.000E+00	1.049E+00
Ca	0.000E+00	3.960E-01	0.000E+00	0.000E+00	3.960E-01
Ti	1.909E-01	1.418E+01	1.956E-02	1.075E+00	1.547E+01
V	3.780E-03	2.359E+00	1.956E-02	2.520E-03	2.385E+00
Cr	1.351E+02	2.964E+02	4.107E+01	4.701E+01	5.196E+02
Mn	1.414E+01	1.731E+01	4.214E+00	4.557E+00	4.022E+01
Fe	4.859E+02	6.441E+02	1.466E+02	1.500E+02	1.427E+03
Co	3.577E-01	1.620E+00	1.146E-01	1.291E-01	2.222E+00
Ni	6.822E+01	4.128E+02	1.873E+01	5.002E+01	5.498E+02
Cu	7.522E-03	2.207E+00	1.956E-02	2.398E-02	2.258E+00
Zn	0.000E+00	7.979E+00	0.000E+00	0.000E+00	7.979E+00
Zr	1.850E+02	8.640E+04	9.575E+02	1.234E+02	8.767E+04
Nb	6.735E-02	4.400E+00	0.000E+00	3.861E-01	4.854E+00
Mo	0.000E+00	1.980E+00	0.000E+00	0.000E+00	1.980E+00
Ag	0.000E+00	1.980E-02	0.000E+00	0.000E+00	1.980E-02
Cd	4.725E-05	4.972E+00	2.445E-04	3.150E-05	4.972E+00
In	0.000E+00	3.960E-01	0.000E+00	0.000E+00	3.960E-01
Sn	3.024E+00	1.413E+03	1.565E+01	2.016E+00	1.433E+03
Gd	0.000E+00	4.950E-01	0.000E+00	0.000E+00	4.950E-01

Table 6-6
BWR Light Elements by Fuel Assembly Region (ORIGEN-ARP Input, grams)
2 Pages

Element	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle	Total
Hf	1.474E-02	6.883E+00	7.628E-02	9.828E-03	6.984E+00
W	3.780E-03	2.161E+00	1.956E-02	2.520E-03	2.187E+00
Pb	0.000E+00	1.980E-01	0.000E+00	0.000E+00	1.980E-01
Bi	0.000E+00	7.920E-02	0.000E+00	0.000E+00	7.920E-02

Proprietary Information on Pages 6-43 through 6-45
Withheld Pursuant to 10 CFR 2.390

Table 6-10
PWR Source Term for the EOS-TC108, Zone 1 (Normal and Accident)

Burnup (GWd/MTU)			33.086	62	33.086	33.086
Enrichment (wt. % U-235)			2.0	3.8	2.0	2.0
Cooling Time (years)			5.00	20.13	5.00	5.00
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.168E+11	9.547E+14	1.468E+11	4.999E+10
5.00E-02	to	1.00E-01	1.735E+10	2.705E+14	1.125E+10	9.697E+09
1.00E-01	to	2.00E-01	1.626E+10	1.798E+14	1.055E+10	2.375E+09
2.00E-01	to	3.00E-01	1.067E+09	5.387E+13	6.968E+08	1.170E+08
3.00E-01	to	4.00E-01	3.136E+09	3.493E+13	2.042E+09	1.519E+08
4.00E-01	to	6.00E-01	6.461E+10	3.725E+13	4.206E+10	1.072E+07
6.00E-01	to	8.00E-01	3.483E+10	1.829E+15	2.764E+10	9.512E+08
8.00E-01	to	1.00E+00	1.144E+11	2.667E+13	2.477E+10	6.536E+10
1.00E+00	to	1.33E+00	4.803E+12	3.972E+13	3.109E+12	2.810E+12
1.33E+00	to	1.66E+00	1.356E+12	4.235E+12	8.780E+11	7.935E+11
1.66E+00	to	2.00E+00	7.184E+02	9.029E+10	1.354E+03	3.931E+02
2.00E+00	to	2.50E+00	3.245E+07	4.694E+09	2.101E+07	1.899E+07
2.50E+00	to	3.00E+00	2.773E+04	9.233E+08	1.795E+04	1.622E+04
3.00E+00	to	4.00E+00	8.044E-06	8.009E+07	3.995E-05	6.624E-06
4.00E+00	to	5.00E+00	2.247E-28	2.690E+07	1.463E-28	0.000E+00
5.00E+00	to	6.50E+00	6.475E-29	1.080E+07	4.215E-29	0.000E+00
6.50E+00	to	8.00E+00	8.236E-30	2.118E+06	5.361E-30	0.000E+00
8.00E+00	to	1.00E+01	1.099E-30	4.496E+05	7.154E-31	0.000E+00
Total Gamma, g/(sec*FA)			6.627E+12	3.431E+15	4.253E+12	3.732E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						7.848E+08
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						1.589E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						2.724E+09

Table 6-11
PWR Source Term for the EOS-TC108, Zone 2 (Normal and Accident)

Burnup (GWd/MTU)			40	62	40	40
Enrichment (wt. % U-235)			2.5	3.8	2.5	2.5
Cooling Time (years)			4.148	7.817	4.148	4.148
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.994E+11	1.461E+15	2.025E+11	5.956E+10
5.00E-02	to	1.00E-01	2.082E+10	3.943E+14	1.368E+10	1.156E+10
1.00E-01	to	2.00E-01	2.165E+10	3.065E+14	1.410E+10	2.835E+09
2.00E-01	to	3.00E-01	1.434E+09	8.619E+13	9.387E+08	1.394E+08
3.00E-01	to	4.00E-01	4.321E+09	5.399E+13	2.816E+09	1.810E+08
4.00E-01	to	6.00E-01	8.906E+10	5.984E+14	5.798E+10	1.317E+07
6.00E-01	to	8.00E-01	4.774E+10	3.023E+15	3.674E+10	1.084E+09
8.00E-01	to	1.00E+00	2.401E+11	3.007E+14	4.680E+10	1.369E+11
1.00E+00	to	1.33E+00	5.720E+12	1.305E+14	3.757E+12	3.350E+12
1.33E+00	to	1.66E+00	1.615E+12	2.943E+13	1.061E+12	9.460E+11
1.66E+00	to	2.00E+00	1.273E+04	3.558E+11	2.738E+04	8.465E+03
2.00E+00	to	2.50E+00	3.865E+07	3.138E+11	2.538E+07	2.264E+07
2.50E+00	to	3.00E+00	3.302E+04	1.992E+10	2.169E+04	1.934E+04
3.00E+00	to	4.00E+00	1.009E-05	1.910E+09	5.011E-05	8.309E-06
4.00E+00	to	5.00E+00	5.952E-28	4.275E+07	3.874E-28	0.000E+00
5.00E+00	to	6.50E+00	1.715E-28	1.716E+07	1.116E-28	0.000E+00
6.50E+00	to	8.00E+00	2.181E-29	3.366E+06	1.420E-29	0.000E+00
8.00E+00	to	1.00E+01	2.911E-30	7.146E+05	1.895E-30	0.000E+00
Total Gamma, g/(sec*FA)			8.060E+12	6.385E+15	5.193E+12	4.508E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.247E+09
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						2.525E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						4.329E+09

Table 6-12
PWR Source Term for the EOS-TC108, Zone 3 (Normal)

Burnup (GWd/MTU)			33.086	61.536	33.086	33.086
Enrichment (wt. % U-235)			2.0	3.8	2.0	2.0
Cooling Time (years)			5.00	10.00	5.00	5.00
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.168E+11	1.289E+15	1.468E+11	4.999E+10
5.00E-02	to	1.00E-01	1.735E+10	3.490E+14	1.125E+10	9.697E+09
1.00E-01	to	2.00E-01	1.626E+10	2.627E+14	1.055E+10	2.375E+09
2.00E-01	to	3.00E-01	1.067E+09	7.484E+13	6.968E+08	1.170E+08
3.00E-01	to	4.00E-01	3.136E+09	4.663E+13	2.042E+09	1.519E+08
4.00E-01	to	6.00E-01	6.461E+10	3.000E+14	4.206E+10	1.072E+07
6.00E-01	to	8.00E-01	3.483E+10	2.574E+15	2.764E+10	9.512E+08
8.00E-01	to	1.00E+00	1.144E+11	1.629E+14	2.477E+10	6.536E+10
1.00E+00	to	1.33E+00	4.803E+12	1.001E+14	3.109E+12	2.810E+12
1.33E+00	to	1.66E+00	1.356E+12	1.794E+13	8.780E+11	7.935E+11
1.66E+00	to	2.00E+00	7.184E+02	1.685E+11	1.354E+03	3.931E+02
2.00E+00	to	2.50E+00	3.245E+07	6.186E+10	2.101E+07	1.899E+07
2.50E+00	to	3.00E+00	2.773E+04	5.154E+09	1.795E+04	1.622E+04
3.00E+00	to	4.00E+00	8.044E-06	5.137E+08	3.995E-05	6.624E-06
4.00E+00	to	5.00E+00	2.247E-28	3.836E+07	1.463E-28	0.000E+00
5.00E+00	to	6.50E+00	6.475E-29	1.540E+07	4.215E-29	0.000E+00
6.50E+00	to	8.00E+00	8.236E-30	3.020E+06	5.361E-30	0.000E+00
8.00E+00	to	1.00E+01	1.099E-30	6.412E+05	7.154E-31	0.000E+00
Total Gamma, g/(sec*FA)			6.627E+12	5.177E+15	4.253E+12	3.732E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.117E+09
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						2.262E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						3.878E+09

Table 6-13
PWR Source Term for the EOS-TC108, Zone 3 (Accident)

Burnup (GWd/MTU)			33.086	62	33.086	33.086
Enrichment (wt. % U-235)			2.0	3.8	2.0	2.0
Cooling Time (years)			5.00	10.25	5.00	5.00
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.168E+11	1.283E+15	1.468E+11	4.999E+10
5.00E-02	to	1.00E-01	1.735E+10	3.475E+14	1.125E+10	9.697E+09
1.00E-01	to	2.00E-01	1.626E+10	2.608E+14	1.055E+10	2.375E+09
2.00E-01	to	3.00E-01	1.067E+09	7.440E+13	6.968E+08	1.170E+08
3.00E-01	to	4.00E-01	3.136E+09	4.635E+13	2.042E+09	1.519E+08
4.00E-01	to	6.00E-01	6.461E+10	2.813E+14	4.206E+10	1.072E+07
6.00E-01	to	8.00E-01	3.483E+10	2.557E+15	2.764E+10	9.512E+08
8.00E-01	to	1.00E+00	1.144E+11	1.542E+14	2.477E+10	6.536E+10
1.00E+00	to	1.33E+00	4.803E+12	9.823E+13	3.109E+12	2.810E+12
1.33E+00	to	1.66E+00	1.356E+12	1.721E+13	8.780E+11	7.935E+11
1.66E+00	to	2.00E+00	7.184E+02	1.605E+11	1.354E+03	3.931E+02
2.00E+00	to	2.50E+00	3.245E+07	5.228E+10	2.101E+07	1.899E+07
2.50E+00	to	3.00E+00	2.773E+04	4.525E+09	1.795E+04	1.622E+04
3.00E+00	to	4.00E+00	8.044E-06	4.548E+08	3.995E-05	6.624E-06
4.00E+00	to	5.00E+00	2.247E-28	3.895E+07	1.463E-28	0.000E+00
5.00E+00	to	6.50E+00	6.475E-29	1.563E+07	4.215E-29	0.000E+00
6.50E+00	to	8.00E+00	8.236E-30	3.067E+06	5.361E-30	0.000E+00
8.00E+00	to	1.00E+01	1.099E-30	6.511E+05	7.154E-31	0.000E+00
Total Gamma, g/(sec*FA)			6.627E+12	5.120E+15	4.253E+12	3.732E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.135E+09
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						2.298E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						3.940E+09

Table 6-14
PWR Source Term for the EOS-TC125/135, Zone 1 (Normal and Accident)

Burnup (GWd/MTU)			33.086	62	33.086	33.086
Enrichment (wt. % U-235)			2.0	3.8	2.0	2.0
Cooling Time (years)			5.00	20.13	5.00	5.00
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.168E+11	9.547E+14	1.468E+11	4.999E+10
5.00E-02	to	1.00E-01	1.735E+10	2.705E+14	1.125E+10	9.697E+09
1.00E-01	to	2.00E-01	1.626E+10	1.798E+14	1.055E+10	2.375E+09
2.00E-01	to	3.00E-01	1.067E+09	5.387E+13	6.968E+08	1.170E+08
3.00E-01	to	4.00E-01	3.136E+09	3.493E+13	2.042E+09	1.519E+08
4.00E-01	to	6.00E-01	6.461E+10	3.725E+13	4.206E+10	1.072E+07
6.00E-01	to	8.00E-01	3.483E+10	1.829E+15	2.764E+10	9.512E+08
8.00E-01	to	1.00E+00	1.144E+11	2.667E+13	2.477E+10	6.536E+10
1.00E+00	to	1.33E+00	4.803E+12	3.972E+13	3.109E+12	2.810E+12
1.33E+00	to	1.66E+00	1.356E+12	4.235E+12	8.780E+11	7.935E+11
1.66E+00	to	2.00E+00	7.184E+02	9.029E+10	1.354E+03	3.931E+02
2.00E+00	to	2.50E+00	3.245E+07	4.694E+09	2.101E+07	1.899E+07
2.50E+00	to	3.00E+00	2.773E+04	9.233E+08	1.795E+04	1.622E+04
3.00E+00	to	4.00E+00	8.044E-06	8.009E+07	3.995E-05	6.624E-06
4.00E+00	to	5.00E+00	2.247E-28	2.690E+07	1.463E-28	0.000E+00
5.00E+00	to	6.50E+00	6.475E-29	1.080E+07	4.215E-29	0.000E+00
6.50E+00	to	8.00E+00	8.236E-30	2.118E+06	5.361E-30	0.000E+00
8.00E+00	to	1.00E+01	1.099E-30	4.496E+05	7.154E-31	0.000E+00
Total Gamma, g/(sec*FA)			6.627E+12	3.431E+15	4.253E+12	3.732E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						7.848E+08
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						1.589E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						2.724E+09

Table 6-15
PWR Source Term for the EOS-TC125/135, Zone 2 (Normal and Accident)

Burnup (GWd/MTU)			50	62	50	50
Enrichment (wt. % U-235)			3.1	3.8	3.1	3.1
Cooling Time (years)			3.959	5.131	3.959	3.959
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	3.521E+11	2.057E+15	2.398E+11	6.714E+10
5.00E-02	to	1.00E-01	2.348E+10	5.785E+14	1.574E+10	1.302E+10
1.00E-01	to	2.00E-01	2.546E+10	4.777E+14	1.666E+10	3.190E+09
2.00E-01	to	3.00E-01	1.693E+09	1.332E+14	1.112E+09	1.569E+08
3.00E-01	to	4.00E-01	5.145E+09	8.924E+13	3.358E+09	2.038E+08
4.00E-01	to	6.00E-01	1.061E+11	1.492E+15	6.904E+10	1.529E+07
6.00E-01	to	8.00E-01	5.685E+10	4.078E+15	4.371E+10	1.283E+09
8.00E-01	to	1.00E+00	2.984E+11	6.795E+14	5.780E+10	1.701E+11
1.00E+00	to	1.33E+00	6.430E+12	1.973E+14	4.315E+12	3.772E+12
1.33E+00	to	1.66E+00	1.816E+12	6.062E+13	1.219E+12	1.065E+12
1.66E+00	to	2.00E+00	2.582E+04	1.630E+12	5.554E+04	1.717E+04
2.00E+00	to	2.50E+00	4.345E+07	2.680E+12	2.916E+07	2.549E+07
2.50E+00	to	3.00E+00	3.712E+04	1.205E+11	2.491E+04	2.177E+04
3.00E+00	to	4.00E+00	1.313E-05	1.126E+10	6.518E-05	1.081E-05
4.00E+00	to	5.00E+00	2.027E-27	4.746E+07	1.319E-27	0.000E+00
5.00E+00	to	6.50E+00	5.840E-28	1.905E+07	3.801E-28	0.000E+00
6.50E+00	to	8.00E+00	7.427E-29	3.737E+06	4.834E-29	0.000E+00
8.00E+00	to	1.00E+01	9.912E-30	7.934E+05	6.452E-30	0.000E+00
Total Gamma, g/(sec*FA)			9.115E+12	9.848E+15	5.981E+12	5.092E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.386E+09
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						2.807E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						4.811E+09

Table 6-16
PWR Source Term for the EOS-TC125/135, Zone 3 (Normal and Accident)

Burnup (GWd/MTU)			40	62	40	40
Enrichment (wt. % U-235)			2.5	3.8	2.5	2.5
Cooling Time (years)			4.698	10.25	4.698	4.698
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.553E+11	1.283E+15	1.736E+11	5.548E+10
5.00E-02	to	1.00E-01	1.929E+10	3.475E+14	1.268E+10	1.075E+10
1.00E-01	to	2.00E-01	1.912E+10	2.608E+14	1.246E+10	2.633E+09
2.00E-01	to	3.00E-01	1.260E+09	7.440E+13	8.254E+08	1.297E+08
3.00E-01	to	4.00E-01	3.742E+09	4.635E+13	2.440E+09	1.684E+08
4.00E-01	to	6.00E-01	7.744E+10	2.813E+14	5.041E+10	1.187E+07
6.00E-01	to	8.00E-01	4.168E+10	2.557E+15	3.279E+10	1.084E+09
8.00E-01	to	1.00E+00	1.542E+11	1.542E+14	3.225E+10	8.801E+10
1.00E+00	to	1.33E+00	5.321E+12	9.823E+13	3.494E+12	3.116E+12
1.33E+00	to	1.66E+00	1.503E+12	1.721E+13	9.868E+11	8.800E+11
1.66E+00	to	2.00E+00	1.926E+03	1.605E+11	3.930E+03	1.187E+03
2.00E+00	to	2.50E+00	3.595E+07	5.228E+10	2.361E+07	2.106E+07
2.50E+00	to	3.00E+00	3.072E+04	4.525E+09	2.017E+04	1.799E+04
3.00E+00	to	4.00E+00	9.973E-06	4.548E+08	4.953E-05	8.212E-06
4.00E+00	to	5.00E+00	5.952E-28	3.895E+07	3.874E-28	0.000E+00
5.00E+00	to	6.50E+00	1.715E-28	1.563E+07	1.116E-28	0.000E+00
6.50E+00	to	8.00E+00	2.181E-29	3.067E+06	1.420E-29	0.000E+00
8.00E+00	to	1.00E+01	2.911E-30	6.511E+05	1.895E-30	0.000E+00
Total Gamma, g/(sec*FA)			7.395E+12	5.120E+15	4.799E+12	4.154E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.135E+09
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						2.298E+09
Treated with peaking factor 1.215 and k_{eff}=0.65 (wet)						3.940E+09

Table 6-17
PWR Source Term for the EOS-HSM, Zone 1 (Normal and Accident)

Burnup (GWd/MTU)			33.086	33.086	33.086	33.086
Enrichment (wt. % U-235)			2.0	2.0	2.0	2.0
Cooling Time (years)			5.00	5.00	5.00	5.00
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.168E+11	1.271E+15	1.468E+11	4.999E+10
5.00E-02	to	1.00E-01	1.735E+10	3.698E+14	1.125E+10	9.697E+09
1.00E-01	to	2.00E-01	1.626E+10	3.082E+14	1.055E+10	2.375E+09
2.00E-01	to	3.00E-01	1.067E+09	8.618E+13	6.968E+08	1.170E+08
3.00E-01	to	4.00E-01	3.136E+09	6.061E+13	2.042E+09	1.519E+08
4.00E-01	to	6.00E-01	6.461E+10	7.503E+14	4.206E+10	1.072E+07
6.00E-01	to	8.00E-01	3.483E+10	2.131E+15	2.764E+10	9.512E+08
8.00E-01	to	1.00E+00	1.144E+11	3.162E+14	2.477E+10	6.536E+10
1.00E+00	to	1.33E+00	4.803E+12	1.083E+14	3.109E+12	2.810E+12
1.33E+00	to	1.66E+00	1.356E+12	3.206E+13	8.780E+11	7.935E+11
1.66E+00	to	2.00E+00	7.184E+02	1.371E+12	1.354E+03	3.931E+02
2.00E+00	to	2.50E+00	3.245E+07	2.541E+12	2.101E+07	1.899E+07
2.50E+00	to	3.00E+00	2.773E+04	1.030E+11	1.795E+04	1.622E+04
3.00E+00	to	4.00E+00	8.044E-06	9.562E+09	3.995E-05	6.624E-06
4.00E+00	to	5.00E+00	2.247E-28	1.253E+07	1.463E-28	0.000E+00
5.00E+00	to	6.50E+00	6.475E-29	5.029E+06	4.215E-29	0.000E+00
6.50E+00	to	8.00E+00	8.236E-30	9.866E+05	5.361E-30	0.000E+00
8.00E+00	to	1.00E+01	1.099E-30	2.095E+05	7.154E-31	0.000E+00
Total Gamma, g/(sec*FA)			6.627E+12	5.438E+15	4.253E+12	3.732E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						3.599E+08
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						7.288E+08

Table 6-18
PWR Source Term for the EOS-HSM, Zone 2 (Normal and Accident)

Burnup (GWd/MTU)			50	50	50	50
Enrichment (wt. % U-235)			3.1	3.1	3.1	3.1
Cooling Time (years)			3.959	3.959	3.959	3.959
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	3.521E+11	2.449E+15	2.398E+11	6.714E+10
5.00E-02	to	1.00E-01	2.348E+10	7.302E+14	1.574E+10	1.302E+10
1.00E-01	to	2.00E-01	2.546E+10	6.328E+14	1.666E+10	3.190E+09
2.00E-01	to	3.00E-01	1.693E+09	1.754E+14	1.112E+09	1.569E+08
3.00E-01	to	4.00E-01	5.145E+09	1.257E+14	3.358E+09	2.038E+08
4.00E-01	to	6.00E-01	1.061E+11	1.808E+15	6.904E+10	1.529E+07
6.00E-01	to	8.00E-01	5.685E+10	3.918E+15	4.371E+10	1.283E+09
8.00E-01	to	1.00E+00	2.984E+11	7.609E+14	5.780E+10	1.701E+11
1.00E+00	to	1.33E+00	6.430E+12	2.058E+14	4.315E+12	3.772E+12
1.33E+00	to	1.66E+00	1.816E+12	7.017E+13	1.219E+12	1.065E+12
1.66E+00	to	2.00E+00	2.582E+04	3.289E+12	5.554E+04	1.717E+04
2.00E+00	to	2.50E+00	4.345E+07	6.769E+12	2.916E+07	2.549E+07
2.50E+00	to	3.00E+00	3.712E+04	2.480E+11	2.491E+04	2.177E+04
3.00E+00	to	4.00E+00	1.313E-05	2.297E+10	6.518E-05	1.081E-05
4.00E+00	to	5.00E+00	2.027E-27	3.151E+07	1.319E-27	0.000E+00
5.00E+00	to	6.50E+00	5.840E-28	1.265E+07	3.801E-28	0.000E+00
6.50E+00	to	8.00E+00	7.427E-29	2.481E+06	4.834E-29	0.000E+00
8.00E+00	to	1.00E+01	9.912E-30	5.268E+05	6.452E-30	0.000E+00
Total Gamma, g/(sec*FA)			9.115E+12	1.089E+16	5.981E+12	5.092E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						9.107E+08
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						1.844E+09

Table 6-19
PWR Source Term for the EOS-HSM, Zone 3 (Normal and Accident)

Burnup (GWd/MTU)			40	40	40	40
Enrichment (wt. % U-235)			2.5	2.5	2.5	2.5
Cooling Time (years)			4.698	4.698	4.698	4.698
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.553E+11	1.621E+15	1.736E+11	5.548E+10
5.00E-02	to	1.00E-01	1.929E+10	4.734E+14	1.268E+10	1.075E+10
1.00E-01	to	2.00E-01	1.912E+10	3.991E+14	1.246E+10	2.633E+09
2.00E-01	to	3.00E-01	1.260E+09	1.112E+14	8.254E+08	1.297E+08
3.00E-01	to	4.00E-01	3.742E+09	7.827E+13	2.440E+09	1.684E+08
4.00E-01	to	6.00E-01	7.744E+10	1.052E+15	5.041E+10	1.187E+07
6.00E-01	to	8.00E-01	4.168E+10	2.730E+15	3.279E+10	1.084E+09
8.00E-01	to	1.00E+00	1.542E+11	4.477E+14	3.225E+10	8.801E+10
1.00E+00	to	1.33E+00	5.321E+12	1.393E+14	3.494E+12	3.116E+12
1.33E+00	to	1.66E+00	1.503E+12	4.315E+13	9.868E+11	8.800E+11
1.66E+00	to	2.00E+00	1.926E+03	1.817E+12	3.930E+03	1.187E+03
2.00E+00	to	2.50E+00	3.595E+07	3.468E+12	2.361E+07	2.106E+07
2.50E+00	to	3.00E+00	3.072E+04	1.364E+11	2.017E+04	1.799E+04
3.00E+00	to	4.00E+00	9.973E-06	1.266E+10	4.953E-05	8.212E-06
4.00E+00	to	5.00E+00	5.952E-28	1.869E+07	3.874E-28	0.000E+00
5.00E+00	to	6.50E+00	1.715E-28	7.501E+06	1.116E-28	0.000E+00
6.50E+00	to	8.00E+00	2.181E-29	1.472E+06	1.420E-29	0.000E+00
8.00E+00	to	1.00E+01	2.911E-30	3.124E+05	1.895E-30	0.000E+00
Total Gamma, g/(sec*FA)			7.395E+12	7.100E+15	4.799E+12	4.154E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						5.376E+08
Treated with peaking factor 1.215 and k_{eff}=0.4 (dry)						1.089E+09

Table 6-20
BWR Source Term for the EOS-TC108, Zone 1 (Normal and Accident)

Burnup (GWD/MTU)			17.07	62	17.07	17.07
Enrichment (wt. % U-235)			0.9	3.8	0.9	0.9
Cooling Time (years)			3.000	16.490	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	5.478E+10	4.235E+14	1.045E+11	2.552E+10
5.00E-02	to	1.00E-01	7.401E+09	1.171E+14	2.736E+09	2.729E+09
1.00E-01	to	2.00E-01	2.931E+09	8.105E+13	6.338E+09	1.404E+09
2.00E-01	to	3.00E-01	1.691E+08	2.424E+13	4.480E+08	8.636E+07
3.00E-01	to	4.00E-01	4.376E+08	1.589E+13	1.711E+09	2.576E+08
4.00E-01	to	6.00E-01	5.899E+09	2.407E+13	3.045E+10	3.932E+09
6.00E-01	to	8.00E-01	3.122E+09	8.094E+14	1.613E+10	2.101E+09
8.00E-01	to	1.00E+00	1.515E+11	1.462E+13	4.572E+10	4.680E+10
1.00E+00	to	1.33E+00	2.111E+12	1.787E+13	6.749E+11	7.732E+11
1.33E+00	to	1.66E+00	5.961E+11	2.287E+12	1.906E+11	2.184E+11
1.66E+00	to	2.00E+00	1.388E+05	4.114E+10	4.210E+04	1.017E+05
2.00E+00	to	2.50E+00	1.426E+07	2.301E+09	4.561E+06	5.225E+06
2.50E+00	to	3.00E+00	1.219E+04	3.129E+08	3.897E+03	4.464E+03
3.00E+00	to	4.00E+00	3.040E-07	3.411E+07	3.117E-08	1.713E-06
4.00E+00	to	5.00E+00	4.928E-31	1.087E+07	2.553E-30	3.289E-31
5.00E+00	to	6.50E+00	1.420E-31	4.362E+06	7.356E-31	9.477E-32
6.50E+00	to	8.00E+00	1.806E-32	8.557E+05	9.356E-32	1.205E-32
8.00E+00	to	1.00E+01	2.410E-33	1.817E+05	1.249E-32	1.609E-33
Total Gamma, g/(sec*FA)			2.933E+12	1.530E+15	1.074E+12	1.074E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						3.168E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						6.505E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						1.115E+09

Table 6-21
BWR Source Term for the EOS-TC108, Zone 2 (Normal and Accident)

Burnup (GWD/MTU)			23.4	62	23.4	23.4
Enrichment (wt. % U-235)			1.2	3.8	1.2	1.2
Cooling Time (years)			3.000	8.515	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	6.519E+10	5.613E+14	1.271E+11	3.062E+10
5.00E-02	to	1.00E-01	8.701E+09	1.516E+14	3.243E+09	3.225E+09
1.00E-01	to	2.00E-01	3.546E+09	1.136E+14	7.976E+09	1.721E+09
2.00E-01	to	3.00E-01	2.057E+08	3.258E+13	5.628E+08	1.063E+08
3.00E-01	to	4.00E-01	5.307E+08	2.101E+13	2.097E+09	3.140E+08
4.00E-01	to	6.00E-01	7.473E+09	1.819E+14	3.859E+10	4.981E+09
6.00E-01	to	8.00E-01	3.945E+09	1.137E+15	2.038E+10	2.655E+09
8.00E-01	to	1.00E+00	1.803E+11	8.994E+13	5.441E+10	5.569E+10
1.00E+00	to	1.33E+00	2.480E+12	4.038E+13	7.924E+11	9.128E+11
1.33E+00	to	1.66E+00	7.005E+11	9.234E+12	2.238E+11	2.578E+11
1.66E+00	to	2.00E+00	1.463E+05	1.069E+11	4.623E+04	1.071E+05
2.00E+00	to	2.50E+00	1.676E+07	7.362E+10	5.355E+06	6.167E+06
2.50E+00	to	3.00E+00	1.432E+04	4.972E+09	4.575E+03	5.270E+03
3.00E+00	to	4.00E+00	4.768E-07	4.831E+08	4.888E-08	2.687E-06
4.00E+00	to	5.00E+00	3.049E-30	1.469E+07	1.579E-29	2.035E-30
5.00E+00	to	6.50E+00	8.785E-31	5.896E+06	4.551E-30	5.863E-31
6.50E+00	to	8.00E+00	1.117E-31	1.157E+06	5.788E-31	7.458E-32
8.00E+00	to	1.00E+01	1.491E-32	2.456E+05	7.725E-32	9.953E-33
Total Gamma, g/(sec*FA)			3.451E+12	2.339E+15	1.270E+12	1.270E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						4.285E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						8.799E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						1.508E+09

Table 6-22
BWR Source Term for the EOS-TC108, Zone 3 (Normal and Accident)

Burnup (GWD/MTU)			62	62	62	62
Enrichment (wt. % U-235)			3.8	3.8	3.8	3.8
Cooling Time (years)			9.699	9.699	9.699	9.699
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	2.884E+10	5.285E+14	2.939E+10	1.222E+10
5.00E-02	to	1.00E-01	4.888E+09	1.430E+14	1.692E+09	1.848E+09
1.00E-01	to	2.00E-01	1.638E+09	1.056E+14	2.718E+09	7.473E+08
2.00E-01	to	3.00E-01	9.060E+07	3.051E+13	1.849E+08	4.356E+07
3.00E-01	to	4.00E-01	1.826E+08	1.963E+13	5.747E+08	9.977E+07
4.00E-01	to	6.00E-01	2.401E+09	1.263E+14	1.240E+10	1.600E+09
6.00E-01	to	8.00E-01	1.262E+09	1.053E+15	6.468E+09	8.900E+08
8.00E-01	to	1.00E+00	1.158E+09	6.453E+13	3.478E+08	4.128E+08
1.00E+00	to	1.33E+00	1.411E+12	3.500E+13	4.489E+11	5.314E+11
1.33E+00	to	1.66E+00	3.985E+11	7.091E+12	1.268E+11	1.501E+11
1.66E+00	to	2.00E+00	2.084E+01	7.384E+10	1.078E+02	1.390E+01
2.00E+00	to	2.50E+00	9.535E+06	3.080E+10	3.033E+06	3.591E+06
2.50E+00	to	3.00E+00	8.147E+03	2.376E+09	2.592E+03	3.068E+03
3.00E+00	to	4.00E+00	1.264E-06	2.379E+08	1.295E-07	7.119E-06
4.00E+00	to	5.00E+00	4.257E-28	1.404E+07	2.205E-27	2.840E-28
5.00E+00	to	6.50E+00	1.227E-28	5.634E+06	6.352E-28	8.184E-29
6.50E+00	to	8.00E+00	1.560E-29	1.105E+06	8.080E-29	1.041E-29
8.00E+00	to	1.00E+01	2.082E-30	2.347E+05	1.078E-29	1.389E-30
Total Gamma, g/(sec*FA)			1.850E+12	2.113E+15	6.295E+11	6.993E+11
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						4.094E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						8.406E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						1.441E+09

Table 6-23
BWR Source Term for the EOS-TC125/135, Zone 1 (Normal and Accident)

Burnup (GWD/MTU)			17.07	62	17.07	17.07
Enrichment (wt. % U-235)			0.9	3.8	0.9	0.9
Cooling Time (years)			3.000	16.490	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	5.478E+10	4.235E+14	1.045E+11	2.552E+10
5.00E-02	to	1.00E-01	7.401E+09	1.171E+14	2.736E+09	2.729E+09
1.00E-01	to	2.00E-01	2.931E+09	8.105E+13	6.338E+09	1.404E+09
2.00E-01	to	3.00E-01	1.691E+08	2.424E+13	4.480E+08	8.636E+07
3.00E-01	to	4.00E-01	4.376E+08	1.589E+13	1.711E+09	2.576E+08
4.00E-01	to	6.00E-01	5.899E+09	2.407E+13	3.045E+10	3.932E+09
6.00E-01	to	8.00E-01	3.122E+09	8.094E+14	1.613E+10	2.101E+09
8.00E-01	to	1.00E+00	1.515E+11	1.462E+13	4.572E+10	4.680E+10
1.00E+00	to	1.33E+00	2.111E+12	1.787E+13	6.749E+11	7.732E+11
1.33E+00	to	1.66E+00	5.961E+11	2.287E+12	1.906E+11	2.184E+11
1.66E+00	to	2.00E+00	1.388E+05	4.114E+10	4.210E+04	1.017E+05
2.00E+00	to	2.50E+00	1.426E+07	2.301E+09	4.561E+06	5.225E+06
2.50E+00	to	3.00E+00	1.219E+04	3.129E+08	3.897E+03	4.464E+03
3.00E+00	to	4.00E+00	3.040E-07	3.411E+07	3.117E-08	1.713E-06
4.00E+00	to	5.00E+00	4.928E-31	1.087E+07	2.553E-30	3.289E-31
5.00E+00	to	6.50E+00	1.420E-31	4.362E+06	7.356E-31	9.477E-32
6.50E+00	to	8.00E+00	1.806E-32	8.557E+05	9.356E-32	1.205E-32
8.00E+00	to	1.00E+01	2.410E-33	1.817E+05	1.249E-32	1.609E-33
Total Gamma, g/(sec*FA)			2.933E+12	1.530E+15	1.074E+12	1.074E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						3.168E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						6.505E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						1.115E+09

Table 6-24
BWR Source Term for the EOS-TC125/135, Zone 2 (Normal and Accident)

Burnup (GWD/MTU)			26.59	62	26.59	26.59
Enrichment (wt. % U-235)			1.3	3.8	1.3	1.3
Cooling Time (years)			3.000	6.971	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	6.951E+10	6.304E+14	1.360E+11	3.271E+10
5.00E-02	to	1.00E-01	9.256E+09	1.714E+14	3.458E+09	3.438E+09
1.00E-01	to	2.00E-01	3.805E+09	1.315E+14	8.658E+09	1.855E+09
2.00E-01	to	3.00E-01	2.210E+08	3.742E+13	6.104E+08	1.147E+08
3.00E-01	to	4.00E-01	5.690E+08	2.449E+13	2.254E+09	3.371E+08
4.00E-01	to	6.00E-01	8.127E+09	3.004E+14	4.197E+10	5.417E+09
6.00E-01	to	8.00E-01	4.287E+09	1.294E+15	2.214E+10	2.885E+09
8.00E-01	to	1.00E+00	1.905E+11	1.421E+14	5.750E+10	5.885E+10
1.00E+00	to	1.33E+00	2.638E+12	4.996E+13	8.425E+11	9.729E+11
1.33E+00	to	1.66E+00	7.451E+11	1.353E+13	2.379E+11	2.747E+11
1.66E+00	to	2.00E+00	1.487E+05	2.171E+11	4.786E+04	1.089E+05
2.00E+00	to	2.50E+00	1.783E+07	2.445E+11	5.694E+06	6.574E+06
2.50E+00	to	3.00E+00	1.523E+04	1.377E+10	4.865E+03	5.617E+03
3.00E+00	to	4.00E+00	5.566E-07	1.306E+09	5.706E-08	3.136E-06
4.00E+00	to	5.00E+00	6.020E-30	1.560E+07	3.119E-29	4.018E-30
5.00E+00	to	6.50E+00	1.735E-30	6.260E+06	8.986E-30	1.158E-30
6.50E+00	to	8.00E+00	2.206E-31	1.228E+06	1.143E-30	1.473E-31
8.00E+00	to	1.00E+01	2.945E-32	2.608E+05	1.525E-31	1.965E-32
Total Gamma, g/(sec*FA)			3.670E+12	2.796E+15	1.353E+12	1.353E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						4.553E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						9.349E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						1.603E+09

Table 6-25
BWR Source Term for the EOS-TC125/135, Zone 3 (Normal)

Burnup (GWD/MTU)			21.72	21.72	21.72	21.72
Enrichment (wt. % U-235)			1.1	1.1	1.1	1.1
Cooling Time (years)			3.000	3.000	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	6.284E+10	8.933E+14	1.217E+11	2.945E+10
5.00E-02	to	1.00E-01	8.418E+09	2.854E+14	3.130E+09	3.116E+09
1.00E-01	to	2.00E-01	3.404E+09	2.626E+14	7.575E+09	1.646E+09
2.00E-01	to	3.00E-01	1.971E+08	7.166E+13	5.346E+08	1.016E+08
3.00E-01	to	4.00E-01	5.085E+08	5.544E+13	2.003E+09	3.004E+08
4.00E-01	to	6.00E-01	7.084E+09	4.152E+14	3.658E+10	4.722E+09
6.00E-01	to	8.00E-01	3.742E+09	7.410E+14	1.933E+10	2.518E+09
8.00E-01	to	1.00E+00	1.736E+11	1.317E+14	5.239E+10	5.363E+10
1.00E+00	to	1.33E+00	2.400E+12	4.950E+13	7.669E+11	8.822E+11
1.33E+00	to	1.66E+00	6.778E+11	1.676E+13	2.166E+11	2.491E+11
1.66E+00	to	2.00E+00	1.443E+05	1.788E+12	4.516E+04	1.057E+05
2.00E+00	to	2.50E+00	1.622E+07	4.468E+12	5.183E+06	5.961E+06
2.50E+00	to	3.00E+00	1.386E+04	1.335E+11	4.428E+03	5.093E+03
3.00E+00	to	4.00E+00	4.305E-07	1.226E+10	4.413E-08	2.426E-06
4.00E+00	to	5.00E+00	2.020E-30	2.004E+06	1.046E-29	1.348E-30
5.00E+00	to	6.50E+00	5.820E-31	8.044E+05	3.015E-30	3.884E-31
6.50E+00	to	8.00E+00	7.403E-32	1.578E+05	3.835E-31	4.941E-32
8.00E+00	to	1.00E+01	9.879E-33	3.350E+04	5.118E-32	6.594E-33
Total Gamma, g/(sec*FA)			3.338E+12	2.929E+15	1.227E+12	1.227E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						5.743E+07
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						1.179E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						2.022E+08

Table 6-26
BWR Source Term for the EOS-TC125/135, Zone 3 (Accident)

Burnup (GWD/MTU)			21.72	62	21.72	21.72
Enrichment (wt. % U-235)			1.1	3.8	1.1	1.1
Cooling Time (years)			3.000	9.699	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	6.284E+10	5.285E+14	1.217E+11	2.945E+10
5.00E-02	to	1.00E-01	8.418E+09	1.430E+14	3.130E+09	3.116E+09
1.00E-01	to	2.00E-01	3.404E+09	1.056E+14	7.575E+09	1.646E+09
2.00E-01	to	3.00E-01	1.971E+08	3.051E+13	5.346E+08	1.016E+08
3.00E-01	to	4.00E-01	5.085E+08	1.963E+13	2.003E+09	3.004E+08
4.00E-01	to	6.00E-01	7.084E+09	1.263E+14	3.658E+10	4.722E+09
6.00E-01	to	8.00E-01	3.742E+09	1.053E+15	1.933E+10	2.518E+09
8.00E-01	to	1.00E+00	1.736E+11	6.453E+13	5.239E+10	5.363E+10
1.00E+00	to	1.33E+00	2.400E+12	3.500E+13	7.669E+11	8.822E+11
1.33E+00	to	1.66E+00	6.778E+11	7.091E+12	2.166E+11	2.491E+11
1.66E+00	to	2.00E+00	1.443E+05	7.384E+10	4.516E+04	1.057E+05
2.00E+00	to	2.50E+00	1.622E+07	3.080E+10	5.183E+06	5.961E+06
2.50E+00	to	3.00E+00	1.386E+04	2.376E+09	4.428E+03	5.093E+03
3.00E+00	to	4.00E+00	4.305E-07	2.379E+08	4.413E-08	2.426E-06
4.00E+00	to	5.00E+00	2.020E-30	1.404E+07	1.046E-29	1.348E-30
5.00E+00	to	6.50E+00	5.820E-31	5.634E+06	3.015E-30	3.884E-31
6.50E+00	to	8.00E+00	7.403E-32	1.105E+06	3.835E-31	4.941E-32
8.00E+00	to	1.00E+01	9.879E-33	2.347E+05	5.118E-32	6.594E-33
Total Gamma, g/(sec*FA)			3.338E+12	2.113E+15	1.227E+12	1.227E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						4.094E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						8.406E+08
Treated with peaking factor 1.232 and k_{eff}=0.65 (wet)						1.441E+09

Table 6-27
BWR Source Term for the EOS-HSM, Zone 1 (Normal and Accident)

Burnup (GWD/MTU)			17.07	40	17.07	17.07
Enrichment (wt. % U-235)			0.9	2.5	0.9	0.9
Cooling Time (years)			3.000	5.955	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	5.478E+10	4.862E+14	1.045E+11	2.552E+10
5.00E-02	to	1.00E-01	7.401E+09	1.362E+14	2.736E+09	2.729E+09
1.00E-01	to	2.00E-01	2.931E+09	1.074E+14	6.338E+09	1.404E+09
2.00E-01	to	3.00E-01	1.691E+08	3.041E+13	4.480E+08	8.636E+07
3.00E-01	to	4.00E-01	4.376E+08	2.076E+13	1.711E+09	2.576E+08
4.00E-01	to	6.00E-01	5.899E+09	2.518E+14	3.045E+10	3.932E+09
6.00E-01	to	8.00E-01	3.122E+09	9.138E+14	1.613E+10	2.101E+09
8.00E-01	to	1.00E+00	1.515E+11	1.114E+14	4.572E+10	4.680E+10
1.00E+00	to	1.33E+00	2.111E+12	3.914E+13	6.749E+11	7.732E+11
1.33E+00	to	1.66E+00	5.961E+11	1.125E+13	1.906E+11	2.184E+11
1.66E+00	to	2.00E+00	1.388E+05	3.161E+11	4.210E+04	1.017E+05
2.00E+00	to	2.50E+00	1.426E+07	4.963E+11	4.561E+06	5.225E+06
2.50E+00	to	3.00E+00	1.219E+04	2.265E+10	3.897E+03	4.464E+03
3.00E+00	to	4.00E+00	3.040E-07	2.113E+09	3.117E-08	1.713E-06
4.00E+00	to	5.00E+00	4.928E-31	6.042E+06	2.553E-30	3.289E-31
5.00E+00	to	6.50E+00	1.420E-31	2.425E+06	7.356E-31	9.477E-32
6.50E+00	to	8.00E+00	1.806E-32	4.757E+05	9.356E-32	1.205E-32
8.00E+00	to	1.00E+01	2.410E-33	1.010E+05	1.249E-32	1.609E-33
Total Gamma, g/(sec*FA)			2.933E+12	2.109E+15	1.074E+12	1.074E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.738E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						3.569E+08

Table 6-28
BWR Source Term for the EOS-HSM, Zone 2 (Normal and Accident)

Burnup (GWD/MTU)			26.59	50	26.59	26.59
Enrichment (wt. % U-235)			1.3	3.1	1.3	1.3
Cooling Time (years)			3.000	5.031	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	6.951E+10	7.110E+14	1.360E+11	3.271E+10
5.00E-02	to	1.00E-01	9.256E+09	2.029E+14	3.458E+09	3.438E+09
1.00E-01	to	2.00E-01	3.805E+09	1.652E+14	8.658E+09	1.855E+09
2.00E-01	to	3.00E-01	2.210E+08	4.652E+13	6.104E+08	1.147E+08
3.00E-01	to	4.00E-01	5.690E+08	3.244E+13	2.254E+09	3.371E+08
4.00E-01	to	6.00E-01	8.127E+09	4.596E+14	4.197E+10	5.417E+09
6.00E-01	to	8.00E-01	4.287E+09	1.298E+15	2.214E+10	2.885E+09
8.00E-01	to	1.00E+00	1.905E+11	1.997E+14	5.750E+10	5.885E+10
1.00E+00	to	1.33E+00	2.638E+12	5.772E+13	8.425E+11	9.729E+11
1.33E+00	to	1.66E+00	7.451E+11	1.872E+13	2.379E+11	2.747E+11
1.66E+00	to	2.00E+00	1.487E+05	6.343E+11	4.786E+04	1.089E+05
2.00E+00	to	2.50E+00	1.783E+07	1.122E+12	5.694E+06	6.574E+06
2.50E+00	to	3.00E+00	1.523E+04	4.695E+10	4.865E+03	5.617E+03
3.00E+00	to	4.00E+00	5.566E-07	4.371E+09	5.706E-08	3.136E-06
4.00E+00	to	5.00E+00	6.020E-30	1.044E+07	3.119E-29	4.018E-30
5.00E+00	to	6.50E+00	1.735E-30	4.191E+06	8.986E-30	1.158E-30
6.50E+00	to	8.00E+00	2.206E-31	8.222E+05	1.143E-30	1.473E-31
8.00E+00	to	1.00E+01	2.945E-32	1.746E+05	1.525E-31	1.965E-32
Total Gamma, g/(sec*FA)			3.670E+12	3.194E+15	1.353E+12	1.353E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						3.018E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						6.197E+08

Table 6-29
BWR Source Term for the EOS-HSM, Zone 3 (Normal and Accident)

Burnup (GWD/MTU)			21.72	40	21.72	21.72
Enrichment (wt. % U-235)			1.1	2.5	1.1	1.1
Cooling Time (years)			3.000	4.715	3.000	3.000
Gamma Source Term, γ/(sec*FA)						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
1.00E-02	to	5.00E-02	6.284E+10	6.488E+14	1.217E+11	2.945E+10
5.00E-02	to	1.00E-01	8.418E+09	1.891E+14	3.130E+09	3.116E+09
1.00E-01	to	2.00E-01	3.404E+09	1.571E+14	7.575E+09	1.646E+09
2.00E-01	to	3.00E-01	1.971E+08	4.404E+13	5.346E+08	1.016E+08
3.00E-01	to	4.00E-01	5.085E+08	3.146E+13	2.003E+09	3.004E+08
4.00E-01	to	6.00E-01	7.084E+09	3.971E+14	3.658E+10	4.722E+09
6.00E-01	to	8.00E-01	3.742E+09	1.069E+15	1.933E+10	2.518E+09
8.00E-01	to	1.00E+00	1.736E+11	1.654E+14	5.239E+10	5.363E+10
1.00E+00	to	1.33E+00	2.400E+12	5.020E+13	7.669E+11	8.822E+11
1.33E+00	to	1.66E+00	6.778E+11	1.629E+13	2.166E+11	2.491E+11
1.66E+00	to	2.00E+00	1.443E+05	7.080E+11	4.516E+04	1.057E+05
2.00E+00	to	2.50E+00	1.622E+07	1.375E+12	5.183E+06	5.961E+06
2.50E+00	to	3.00E+00	1.386E+04	5.282E+10	4.428E+03	5.093E+03
3.00E+00	to	4.00E+00	4.305E-07	4.897E+09	4.413E-08	2.426E-06
4.00E+00	to	5.00E+00	2.020E-30	6.334E+06	1.046E-29	1.348E-30
5.00E+00	to	6.50E+00	5.820E-31	2.542E+06	3.015E-30	3.884E-31
6.50E+00	to	8.00E+00	7.403E-32	4.987E+05	3.835E-31	4.941E-32
8.00E+00	to	1.00E+01	9.879E-33	1.059E+05	5.118E-32	6.594E-33
Total Gamma, g/(sec*FA)			3.338E+12	2.771E+15	1.227E+12	1.227E+12
Total Neutron Source Term, n/(sec*FA)						
Raw ORIGEN-ARP source for uniform burnup						1.821E+08
Treated with peaking factor 1.232 and k_{eff}=0.4 (dry)						3.739E+08

Table 6-30
PWR Axial Source Distributions

Axial Segment	Percentage of In-core Height (top of segment)	Burnup Profile (Gamma Profile)	Normalized Gamma Profile(1)	Neutron Profile(2)	Normalized Neutron Profile(3)
1	5.6	0.619	0.0344	0.133	0.0061
2	11.1	0.924	0.0513	0.718	0.0328
3	16.7	1.056	0.0587	1.257	0.0575
4	22.2	1.097	0.0609	1.475	0.0675
5	27.8	1.103	0.0613	1.509	0.0690
6	33.3	1.101	0.0612	1.498	0.0685
7	38.9	1.103	0.0613	1.509	0.0690
8	44.4	1.112	0.0618	1.562	0.0714
9	50.0	1.125	0.0625	1.640	0.0750
10	55.6	1.136	0.0631	1.708	0.0781
11	61.1	1.143	0.0635	1.753	0.0802
12	66.7	1.143	0.0635	1.753	0.0802
13	72.2	1.136	0.0631	1.708	0.0781
14	77.8	1.115	0.0619	1.580	0.0722
15	83.3	1.047	0.0582	1.213	0.0555
16	88.9	0.882	0.0490	0.590	0.0270
17	94.4	0.701	0.0389	0.225	0.0103
18	100.0	0.456	0.0253	0.037	0.0017
-	Sum	17.999	1.000	21.869	1.000
-	Average	1.0	-	1.215	-

Notes:

(1) The normalized gamma profile is the gamma profile divided by the sum of the gamma profile (sum=17.999).

(2) The neutron profile is the burnup profile raised to the 4.2 power.

(3) The normalized neutron profile is the neutron profile divided by the sum of the neutron profile (sum=21.869).

Table 6-31
BWR Axial Source Distributions

Axial Segment	Percentage of In-core Height (top of segment)	Burnup Profile (Gamma Profile)	Normalized Gamma Profile(1)	Neutron Profile(2)	Normalized Neutron Profile(3)
1	4	0.7144	0.0288	0.244	0.0079
2	8	1.0993	0.0443	1.488	0.0483
3	12	1.2185	0.0491	2.293	0.0745
4	16	1.2312	0.0496	2.395	0.0778
5	20	1.2200	0.0492	2.305	0.0748
6	24	1.1799	0.0475	2.003	0.0650
7	28	1.1528	0.0465	1.817	0.0590
8	32	1.1306	0.0456	1.675	0.0544
9	36	1.1140	0.0449	1.574	0.0511
10	40	1.1192	0.0451	1.605	0.0521
11	44	1.1070	0.0446	1.533	0.0498
12	48	1.0807	0.0436	1.385	0.0450
13	52	1.0722	0.0432	1.340	0.0435
14	56	1.0579	0.0426	1.267	0.0411
15	60	1.0313	0.0416	1.138	0.0370
16	64	1.0180	0.0410	1.078	0.0350
17	68	1.0180	0.0410	1.078	0.0350
18	72	1.0212	0.0412	1.092	0.0355
19	76	1.0140	0.0409	1.060	0.0344
20	80	0.9789	0.0394	0.914	0.0297
21	84	0.9169	0.0370	0.695	0.0226
22	88	0.8430	0.0340	0.488	0.0158
23	92	0.7185	0.0290	0.249	0.0081
24	96	0.5540	0.0223	0.084	0.0027
25	100	0.2025	0.0082	0.001	0.0000
-	Sum	24.814	1.000	30.801	1.000
-	Average	1.0	-	1.232	-

Notes:

(1) The normalized gamma profile is the gamma profile divided by the sum of the gamma profile (sum=24.814).

(2) The neutron profile is the burnup profile raised to the 4.2 power.

(3) The normalized neutron profile is the neutron profile divided by the sum of the neutron profile (sum=30.801).

Table 6-32
BPRA Hardware Masses

Region	SS304 (kg)	Inconel X-750 (kg)	Inconel 718 (kg)	Poison (kg)	Zr-4 (kg)
BW 15x15					
Top	3.602	0.058	0	0	0
Plenum	1.068	0	0	0.724	1.197
Core	2.468	0	0	9.146	11.98
WE 17x17 Pyrex					
Top	2.62	0	0.42	0	0
Plenum	2.85	0	0	0	0
Core	11.9	0	0	5.08	0
WE 17x17 WABA					
Top	2.95	0	0	0	0
Plenum	2.76	0	0	0	2.61
Core	0	0	0	2.5	14.8

Table 6-33
TPA Hardware Masses

Region	SS304 (kg)	Inconel X-750 (kg)	Inconel 718 (kg)	Poison (kg)	Zr-4 (kg)
WE 17x17					
Top	2.468	0	0.358	0	0
Plenum	3.266	0	0	0	0
WE 14x14 Type 1					
Top	2.03	0	0.417	0	0
Plenum	2.69	0	0	0	0
WE 14x14 Type 2					
Top	2.03	0.363	0	0	0
Plenum	2.69	0	0	0	0

Table 6-34
Elemental Constituents of Pyrex Poison

Element	Pyrex (wt. fraction)
B	0.040064
O	0.539562
Na	0.028191
Al	0.011644
Si	0.377220
K	0.003321

Table 6-35
CC ORIGEN-ARP Input Mass

Element	BPRA			TPA	
	Active Fuel (g)	Plenum (g)	Top (g)	Plenum (g)	Top (g)
B	203.525	0.000	0.000	0.000	0.000
C	9.508	0.455	0.290	2.609	2.115
N	15.450	0.740	0.475	4.240	3.670
O	2740.969	0.000	0.000	0.000	0.000
Na	143.210	0.000	0.000	0.000	0.000
Al	59.151	0.000	0.046	0.000	2.145
Si	2035.120	5.693	3.615	32.618	25.363
P	5.348	0.256	0.162	1.468	1.109
S	3.565	0.171	0.108	0.979	0.765
K	16.871	0.000	0.000	0.000	0.000
Ti	0.000	0.000	0.145	0.000	2.860
V	0.000	0.000	0.000	0.000	0.000
Cr	2258.087	108.160	69.218	619.741	536.247
Mn	237.693	11.385	7.235	65.236	50.011
Fe	8181.881	391.905	248.050	2245.548	1761.237
Co	5.950	0.285	0.183	1.633	1.413
Ni	1060.112	50.778	36.275	290.952	405.887
Cu	0.000	0.000	0.003	0.000	0.358
Zr	0.000	0.000	0.000	0.000	0.000
Nb	0.000	0.000	0.052	0.000	19.854
Mo	0.000	0.000	0.000	0.000	10.726

Note: For the BPRA ORIGEN-ARP inputs, the flux scaling factors are applied to the constituent masses. For the TPA inputs, the true masses are used in the inputs and the flux scaling factors are applied to the burnup.

Table 6-36
CC Co-60 Activity and Decay Heat

Parameter	BPRA			TPA	
	Active Fuel	**Plenum**	**Top**	**Plenum**	**Top**
Co-60 (Ci)	308	14.8	9.5	44.1	18.9
Decay Heat (watts)	4.8	0.2	0.1	0.7	0.3

Table 6-37
CC Source Term

E_{min} (MeV)	Zone	Zones 1, 2, 3	Zones 1, 2	Zone 3	Zones 1, 2	Zone 3
	Co-60 (Ci)	308	44.1	14.8	18.9	9.5
	E_{max} (MeV)	Active Fuel (γ/s-CC)	Plenum (γ/s-CC)	Plenum (γ/s-CC)	Top (γ/s-CC)	Top (γ/s-CC)
1.00E-02	5.00E-02	3.133E+11	4.540E+10	1.502E+10	2.061E+10	9.702E+09
5.00E-02	1.00E-01	6.183E+10	8.841E+09	2.965E+09	3.795E+09	1.911E+09
1.00E-01	2.00E-01	1.504E+10	2.145E+09	7.211E+08	9.226E+08	4.648E+08
2.00E-01	3.00E-01	7.435E+08	1.068E+08	3.565E+07	4.682E+07	2.300E+07
3.00E-01	4.00E-01	9.718E+08	1.397E+08	4.660E+07	6.024E+07	3.005E+07
4.00E-01	6.00E-01	7.218E+07	1.113E+07	3.288E+06	5.563E+06	2.144E+06
6.00E-01	8.00E-01	2.770E+07	5.117E+06	1.328E+06	1.497E+09	7.527E+06
8.00E-01	1.00E+00	1.640E+10	1.172E+09	7.865E+08	1.849E+09	5.047E+08
1.00E+00	1.33E+00	1.798E+13	2.573E+12	8.624E+11	1.100E+12	5.559E+11
1.33E+00	1.66E+00	5.078E+12	7.267E+11	2.435E+11	3.106E+11	1.570E+11
1.66E+00	2.00E+00	3.510E+00	3.100E-05	1.936E-05	2.716E-03	2.344E-04
2.00E+00	2.50E+00	1.215E+08	1.739E+07	5.827E+06	7.432E+06	3.756E+06
2.50E+00	3.00E+00	1.038E+05	1.486E+04	4.979E+03	6.350E+03	3.210E+03
3.00E+00	4.00E+00	1.322E-02	3.488E-11	1.102E-11	1.051E-06	9.764E-07
4.00E+00	5.00E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
5.00E+00	6.50E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
6.50E+00	8.00E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
8.00E+00	1.00E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
	Total	2.347E+13	3.358E+12	1.125E+12	1.439E+12	7.256E+11

Proprietary Information on Pages 6-73 through 6-74
Withheld Pursuant to 10 CFR 2.390

Table 6-40
Irradiation Gases

Element	PWR with CC (no IFBA) (moles)(1)	PWR with IFBA (no CC) (moles)(1)	BWR (moles)(1)
H	1.63E+00	1.58E+00	1.19E+00
He	1.17E+01	1.16E+01	1.08E+00
N	3.52E+00	2.33E+00	1.06E+00
F	2.77E-01	2.77E-01	1.12E-01
Ne	2.27E-04	1.63E-04	7.03E-05
Cl	7.35E-02	7.34E-02	2.96E-02
Ar	6.52E-04	2.60E-05	1.38E-05
Br	2.15E-01	2.15E-01	8.80E-02
Kr	3.25E+00	3.25E+00	1.34E+00
I	1.51E+00	1.51E+00	5.90E-01
Xe	3.69E+01	3.68E+01	1.51E+01
Rn	6.15E-13	6.14E-13	2.21E-13
Total	59.0(2)(3)	57.7	20.6

Notes:

(1) These gases represent gas generated by irradiation and do not include any gas present when the fuel or CC was originally fabricated.

(2) The 59.0 moles of gas includes 49.4 moles from the PWR FA and 9.6 moles from CC.

(3) For fuel with a total unirradiated fuel length < 157 inches, this value is 54.8 moles.

Table 6-41
MCNP Material Compositions (wt. %)

Element	Carbon Steel	Stainless Steel	Dry Air	Water	Regular Concrete	Soil
Hydrogen	-	-	-	11.1894	1.0	-
Helium	-	-	-	-	-	-
Boron	-	-	-	-	-	-
Carbon	0.5	0.04	0.0124	-	-	-
Nitrogen	-	-	75.5268	-	-	-
Oxygen	-	-	23.1781	88.8106	53.2	51.37
Sodium	-	-	-	-	2.9	0.614
Magnesium	-	-	-	-	-	1.33
Aluminum	-	-	-	-	3.4	6.856
Silicon	-	0.5	-	-	33.7	27.118
Phosphorous	-	0.023	-	-	-	-
Sulfur	-	0.015	-	-	-	-
Argon	-	-	1.2827	-	-	-
Potassium	-	-	-	-	-	1.433
Calcium	-	-	-	-	4.4	5.117
Titanium	-	-	-	-	-	0.461
Chromium	-	19	-	-	-	-
Manganese	-	1	-	-	-	0.0716
Iron	99.5	70.173	-	-	1.4	5.629
Cobalt	-	-	-	-	-	-
Nickel	-	9.25	-	-	-	-
Copper	-	-	-	-	-	-
Zirconium	-	-	-	-	-	-
Niobium	-	-	-	-	-	-
Molybdenum	-	-	-	-	-	-
Tin	-	-	-	-	-	-
Lead	-	-	-	-	-	-
Density (g/cm^3)	7.82	8.00	0.001205	(1)	2.243	1.52

Note: (1) 0.958 g/cm^3 inside the DSC and 0.9982 g/cm^3 inside the neutron shield.

Table 6-42
MCNP Borated Polyethylene Composition

Element	Atom Density (atom/b-cm)
Hydrogen	6.1848E-02
Boron-10	5.5978E-04
Boron-11	2.2532E-03
Carbon	3.6381E-02
Oxygen	4.2195E-03
Total	1.0526E-01

Table 6-43
MCNP PWR Dry Fuel Compositions (wt. fraction)

Material	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle
Boron	2.7088E-06	3.7133E-07	1.7956E-05	7.8154E-06
Carbon	2.0552E-04	5.4215E-06	3.3783E-04	4.5158E-04
Oxygen	6.3495E-04	9.8758E-02	5.4025E-04	-
Aluminum	2.7088E-04	3.7133E-05	1.7956E-03	7.8154E-04
Silicon	2.2469E-03	2.3617E-05	2.0878E-03	4.7155E-03
Phosphorous	1.0302E-04	1.0397E-06	9.3784E-05	2.1593E-04
Sulfur	6.9824E-05	1.0397E-06	7.8651E-05	1.4844E-04
Titanium	4.8759E-04	6.6840E-05	3.2322E-03	1.4068E-03
Chromium	8.9659E-02	1.5716E-03	1.0462E-01	1.9000E-01
Manganese	4.3216E-03	2.3617E-05	3.0336E-03	8.9340E-03
Iron	3.0144E-01	1.5836E-03	1.9469E-01	6.1862E-01
Cobalt	4.9300E-04	6.7583E-05	3.2681E-03	1.4224E-03
Nickel	6.6661E-02	3.8767E-03	2.0496E-01	1.5963E-01
Copper	1.4790E-04	2.0275E-05	9.8042E-04	4.2672E-04
Zirconium	5.2126E-01	1.5811E-01	4.4352E-01	-
Niobium	2.7765E-03	3.8062E-04	1.8405E-02	8.0108E-03
Molybdenum	1.6524E-03	2.2651E-04	1.0953E-02	4.7674E-03
Tin	7.4087E-03	2.2471E-03	6.3037E-03	-
U-234	-	1.9571E-04	-	-
U-235	-	2.1989E-02	-	-
U-236	-	1.0115E-04	-	-
U-238	-	7.1069E-01	-	-
Total	1.0	1.0	1.0	1.0
Density (g/cm^3)	1.9683	3.9508	1.4454	1.8560

Table 6-44
MCNP PWR Wet Fuel Compositions (wt. fraction)

Material	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle
Hydrogen	2.9202E-02	1.4216E-02	3.8863E-02	3.1790E-02
Boron	2.0019E-06	3.2415E-07	1.1720E-05	5.5950E-06
Carbon	1.5188E-04	4.7326E-06	2.2049E-04	3.2328E-04
Oxygen	2.3225E-01	1.9905E-01	3.0881E-01	2.5232E-01
Aluminum	2.0019E-04	3.2415E-05	1.1720E-03	5.5950E-04
Silicon	1.6605E-03	2.0616E-05	1.3627E-03	3.3758E-03
Phosphorous	7.6132E-05	9.0763E-07	6.1211E-05	1.5458E-04
Sulfur	5.1601E-05	9.0763E-07	5.1334E-05	1.0626E-04
Titanium	3.6033E-04	5.8348E-05	2.1096E-03	1.0071E-03
Chromium	6.6260E-02	1.3719E-03	6.8286E-02	1.3602E-01
Manganese	3.1937E-03	2.0616E-05	1.9800E-03	6.3958E-03
Iron	2.2277E-01	1.3824E-03	1.2707E-01	4.4286E-01
Cobalt	3.6434E-04	5.8996E-05	2.1330E-03	1.0183E-03
Nickel	4.9263E-02	3.3842E-03	1.3377E-01	1.1428E-01
Copper	1.0930E-04	1.7699E-05	6.3990E-04	3.0549E-04
Zirconium	3.8522E-01	1.3802E-01	2.8948E-01	-
Niobium	2.0519E-03	3.3226E-04	1.2013E-02	5.7349E-03
Molybdenum	1.2211E-03	1.9773E-04	7.1491E-03	3.4130E-03
Tin	5.4751E-03	1.9616E-03	4.1143E-03	-
U-234	-	1.7084E-04	-	-
U-235	-	1.9196E-02	-	-
U-236	-	8.8300E-05	-	-
U-238	-	6.2040E-01	-	-
Total	1.0	1.0	1.0	1.0
Density (g/cm^3)	2.6635	4.5258	2.2146	2.5925

Table 6-45
MCNP BWR Dry Fuel Compositions (wt. fraction)

Material	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle
Boron	-	-	-	6.0000E-06
Carbon	3.1900E-04	9.6424E-02	7.1000E-05	3.0500E-04
Oxygen	2.5100E-04	2.2200E-04	9.8500E-04	3.9400E-04
Aluminum	4.2000E-05	7.0000E-06	-	5.6300E-04
Silicon	3.9370E-03	4.0000E-06	8.8400E-04	3.1460E-03
Phosphorus	1.8100E-04	-	4.1000E-05	1.4400E-04
Sulfur	1.1800E-04	-	2.7000E-05	9.9000E-05
Titanium	7.5000E-05	1.2000E-05	-	1.0130E-03
Chromium	1.5037E-01	4.3200E-04	3.4406E-02	1.2766E-01
Manganese	7.8470E-03	4.0000E-06	1.7680E-03	5.9340E-03
Iron	5.5060E-01	5.9100E-04	1.2568E-01	4.1107E-01
Cobalt	7.6000E-05	1.2000E-05	-	1.0240E-03
Nickel	7.6705E-02	6.8400E-04	1.6351E-02	1.1067E-01
Copper	2.3000E-05	4.0000E-06	-	3.0700E-04
Zirconium	2.0585E-01	1.8186E-01	8.0830E-01	3.2386E-01
Niobium	4.2600E-04	6.7000E-05	-	5.7690E-03
Molybdenum	2.5400E-04	4.0000E-05	-	3.4330E-03
Tin	2.9260E-03	2.5850E-03	1.1488E-02	4.6030E-03
U-234	-	1.9100E-04	-	-
U-235	-	2.1511E-02	-	-
U-236	-	9.9000E-05	-	-
U-238	-	6.9525E-01	-	-
Total	1.0	1.0	1.0	1.0
Density (g/cm^3)	1.980	4.201	1.006	0.663

Table 6-46
MCNP BWR Wet Fuel Compositions (wt. fraction)

Material	Bottom Nozzle	Active Fuel	Plenum	Top Nozzle
Hydrogen	2.9538E-02	1.2850E-02	5.0093E-02	6.3595E-02
Boron	-	-	-	2.0000E-06
Carbon	2.3500E-04	8.5351E-02	3.9000E-05	1.3200E-04
Oxygen	2.3463E-01	1.0219E-01	3.9814E-01	5.0492E-01
Aluminum	3.1000E-05	6.0000E-06	-	2.4300E-04
Silicon	2.8970E-03	4.0000E-06	4.8800E-04	1.3580E-03
Phosphorus	1.3300E-04	-	2.2000E-05	6.2000E-05
Sulfur	8.7000E-05	-	1.5000E-05	4.3000E-05
Titanium	5.5000E-05	1.0000E-05	-	4.3700E-04
Chromium	1.1068E-01	3.8300E-04	1.9003E-02	5.5104E-02
Manganese	5.7750E-03	4.0000E-06	9.7600E-04	2.5610E-03
Iron	4.0525E-01	5.2300E-04	6.9417E-02	1.7744E-01
Cobalt	5.6000E-05	1.1000E-05	-	4.4200E-04
Nickel	5.6456E-02	6.0600E-04	9.0310E-03	4.7773E-02
Copper	1.7000E-05	3.0000E-06	-	1.3300E-04
Zirconium	1.5151E-01	1.6098E-01	4.4643E-01	1.3980E-01
Niobium	3.1400E-04	5.9000E-05	0.0000E+00	2.4900E-03
Molybdenum	1.8700E-04	3.5000E-05	0.0000E+00	1.4820E-03
Tin	2.1530E-03	2.2880E-03	6.3450E-03	1.9870E-03
U-234	-	1.6900E-04	-	-
U-235	-	1.9041E-02	-	-
U-236	-	8.8000E-05	-	-
U-238	-	6.1541E-01	-	-
Total	1.0	1.0	1.0	1.0
Density (g/cm^3)	2.690	4.746	1.822	1.536

Table 6-47
EOS-37PTH DSC and EOS-89BTH DSC Key As-Modeled Dimensions
(Inches)

Item	EOS-37PTH DSC	EOS-89BTH DSC
Carbon steel shield plug thickness	6.0	6.0
Stainless steel inner top cover plate thickness	2.0	2.0
Stainless steel outer top cover plate thickness	2.0	2.0
Stainless steel shell thickness	0.5	0.5
Stainless steel shell outer diameter	75.5	75.5
Stainless steel outer bottom cover thickness	2.0	2.0
Carbon steel inner bottom shield thickness	4.0	4.0
Stainless steel inner bottom cover thickness	2.0	2.0
Basket compartment inner width	8.86	5.85
Carbon steel basket plate thickness(1)	0.281	0.1875
Aluminum basket plate thickness	0.250	0
MMC basket plate thickness	0.164	0.175

Note:

(1) The EOS-89BTH DSC carbon steel basket plates are modeled as stainless steel. This has no effect on the results.

Table 6-48
EOS-TC108 and EOS-TC125/135 Key As-Modeled Dimensions (Inches)

Item	EOS-TC108	EOS-TC125/135
Cask inner diameter	76.25	76.25
Carbon steel inner shell thickness	0.75	0.75
Stainless steel side lead minimum thickness	2.4	3.51
Carbon steel outer shell thickness	1.0	1.0
Neutron shield water thickness	2.5	4.0
Ram port inner diameter	22.0	22.0
Carbon steel ram plate thickness	1.25	1.25
Carbon steel bottom panel thickness	0.75	0.75
Borated polyethylene thickness	2.13	2.13
Carbon steel bottom end plate thickness	2.0	2.0
Carbon steel bottom ring height	9.0	9.0
Carbon steel top ring height	16.25	16.25
Top ring lead height	5.65	5.65
Top ring lead minimum thickness	0.85	0.85
Top ring thickness (at top nozzle)	4.63	5.38
Top cover plate (lid) thickness	2.0 (Aluminum)	3.2 (Carbon Steel)

Table 6-49
EOS-TC Model Configurations

EOS-TC108							
Configuration	**Water Fill**	**Shield Plug**	**ITCP**	**OTCP**	**Annulus Water Fill**	**Neutron Shield**	**Top Cover Plate (Lid)**
Decontamination	Partial(1)	On	Off	Off	Full	Off	Off
Welding/Drying	Empty	On	On	Off	-12 inches	On/Full	Off
Downending/Transfer	Empty	On	On	On	Empty	On/Full	On
EOS-TC125/135							
Configuration	**Water Fill**	**Shield Plug**	**ITCP**	**OTCP**	**Annulus Water Fill**	**Neutron Shield**	**Top Cover Plate (Lid)**
Decontamination	Full	On	Off	Off	Full	Empty	Off
Welding/Drying	Empty	On	On	Off	-12 inches	Full	Off
Downending/Transfer	Empty	On	On	On	Empty	Full	On

Note: (1) Top nozzle and plenum are dry for EOS-37PTH DSC, top nozzle is dry for EOS-89BTH DSC.

Table 6-50
EOS-HSM Key As-Modeled Dimensions (Inches)

Item	EOS-HSM-Short	EOS-HSM-Medium
Base/roof width	116	116
Base height	178	178
Base/roof length	228	248
Base upper side wall thickness	12	12
Base upper rear wall thickness	12	12
Roof thickness	44	44
Rear/end (side) shield wall thickness	36	36
Rear/end (side)/corner shield wall height	222	222
End (side) shield wall length	117	127
Corner shield wall width (square)	36	36
Door inner steel plate thickness	1	1
Door concrete thickness (centerline)	30.5	30.5
Door outer concrete width (square)	100.375	100.375
Inlet vent height	30	30
Inlet vent width	10	10
Base outlet vent height	8	8
Base outlet vent length	128	148
Roof outlet vent opening width	8	8
Roof outlet vent opening length	132 max./116 min. (trapezoidal)	152 max./136 min. (trapezoidal)
Outlet vent cover width	48	48
Outlet vent cover length	156	176
Outlet vent cover concrete thickness	10.625	10.625
Outlet vent cover steel plate thickness	1	1
Outlet vent cover opening height	6.25	6.25
Outlet vent cover opening length	132	152
Gaps between base and shield walls	1.5	1.5

Table 6-51
ANSI/ANS-6.1.1-1977 Flux-to-Dose-Rate Conversion Factors

E (MeV)	Neutron Factors (mrem/hr)/(n/cm^2-s)	E (MeV)	Neutron Factors (mrem/hr)/(n/cm^2-s)
2.50E-08	3.67E-03	0.5	9.26E-02
1.00E-07	3.67E-03	1.0	1.32E-01
1.00E-06	4.46E-03	2.5	1.25E-01
1.00E-05	4.54E-03	5.0	1.56E-01
1.00E-04	4.18E-03	7.0	1.47E-01
0.001	3.76E-03	10.0	1.47E-01
0.01	3.56E-03	14.0	2.08E-01
0.1	2.17E-02	20.0	2.27E-01
E (MeV)	**Gamma Factors (mrem/hr)/(γ/cm^2-s)**	**E (MeV)**	**Gamma Factors (mrem/hr)/(γ/cm^2-s)**
0.01	3.96E-03	1.4	2.51E-03
0.03	5.82E-04	1.8	2.99E-03
0.05	2.90E-04	2.2	3.42E-03
0.07	2.58E-04	2.6	3.82E-03
0.1	2.83E-04	2.8	4.01E-03
0.15	3.79E-04	3.25	4.41E-03
0.2	5.01E-04	3.75	4.83E-03
0.25	6.31E-04	4.25	5.23E-03
0.3	7.59E-04	4.75	5.60E-03
0.35	8.78E-04	5.0	5.80E-03
0.4	9.85E-04	5.25	6.01E-03
0.45	1.08E-03	5.75	6.37E-03
0.5	1.17E-03	6.25	6.74E-03
0.55	1.27E-03	6.75	7.11E-03
0.6	1.36E-03	7.5	7.66E-03
0.65	1.44E-03	9.0	8.77E-03
0.7	1.52E-03	11.0	1.03E-02
0.8	1.68E-03	13.0	1.18E-02
1.0	1.98E-03	15.0	1.33E-02

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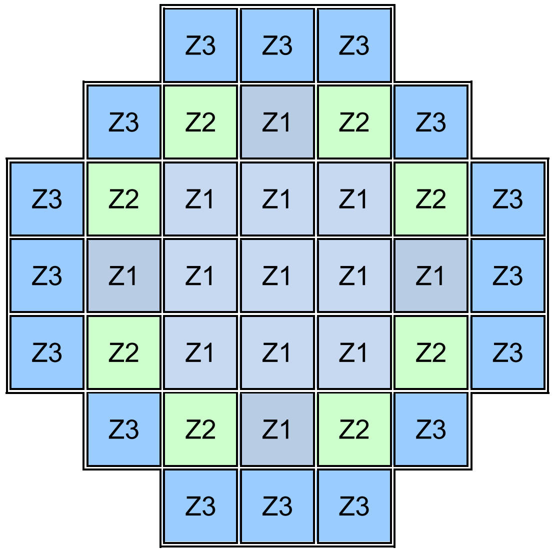
Table 6-58
EOS-HSM Neutron Average Fluxes and Dose Rates, EOS-89BTH DSC

Table 6-59
EOS-HSM Grout Study Results, EOS-89BTH DSC

Configuration	Side Gamma Dose Rate (mrem/hr)
No grout	1.40
3 in. grout	1.78
6 in. grout	2.20



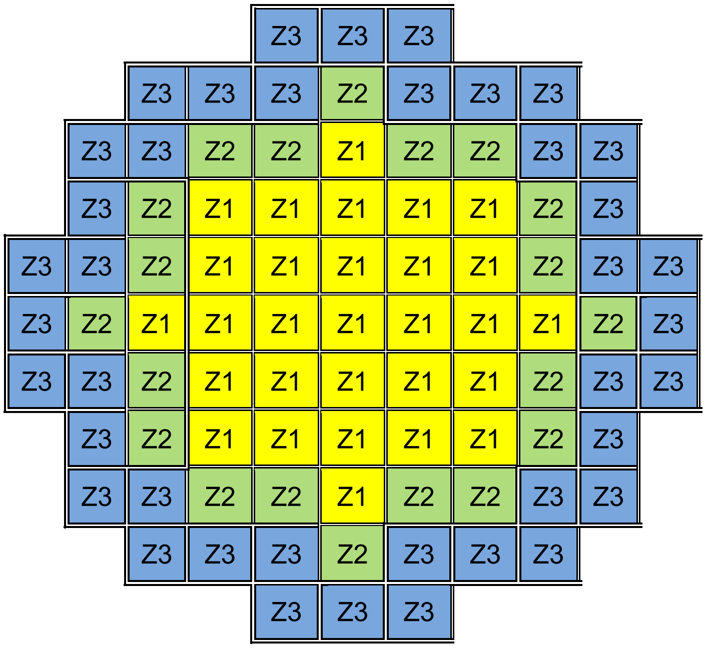
Heat Load Zone Configuration for the EOS-37PTH in the EOS-TC125/135

Zone No.	1	2	3
Maximum Decay Heat (kW/FA plus CCs if any)	1.0	2.0	1.3125
Minimum Cooling Time, Standard (years)	3.0	3.0	3.0
Minimum Cooling Time, SS304 Reconstituted (years)	15.0	15.0	15.0
Number of Fuel Assemblies	13	8	16
Maximum Decay Heat per DSC (kW)	50.0		

Heat Load Zone Configuration for the EOS-37PTH in the EOS-TC108

Zone No.	1	2	3	
Maximum Decay Heat (kW/FA plus CCs if any)	1.0	1.5	≤ 1.0	1.0 < H ≤ 1.05
Minimum Cooling Time, Standard (years)	3.0	3.0	5.0	8.0
Minimum Cooling Time, SS304 Reconstituted (years)	15.0	15.0	15.0	15.0
Number of Fuel Assemblies	13	8	16	16
Maximum Decay Heat per DSC (kW)	41.8			

Figure 6-1
EOS-37PTH Heat Load Zone Configurations



Heat Load Zone Configuration for the EOS-89BTH in the EOS-TC125

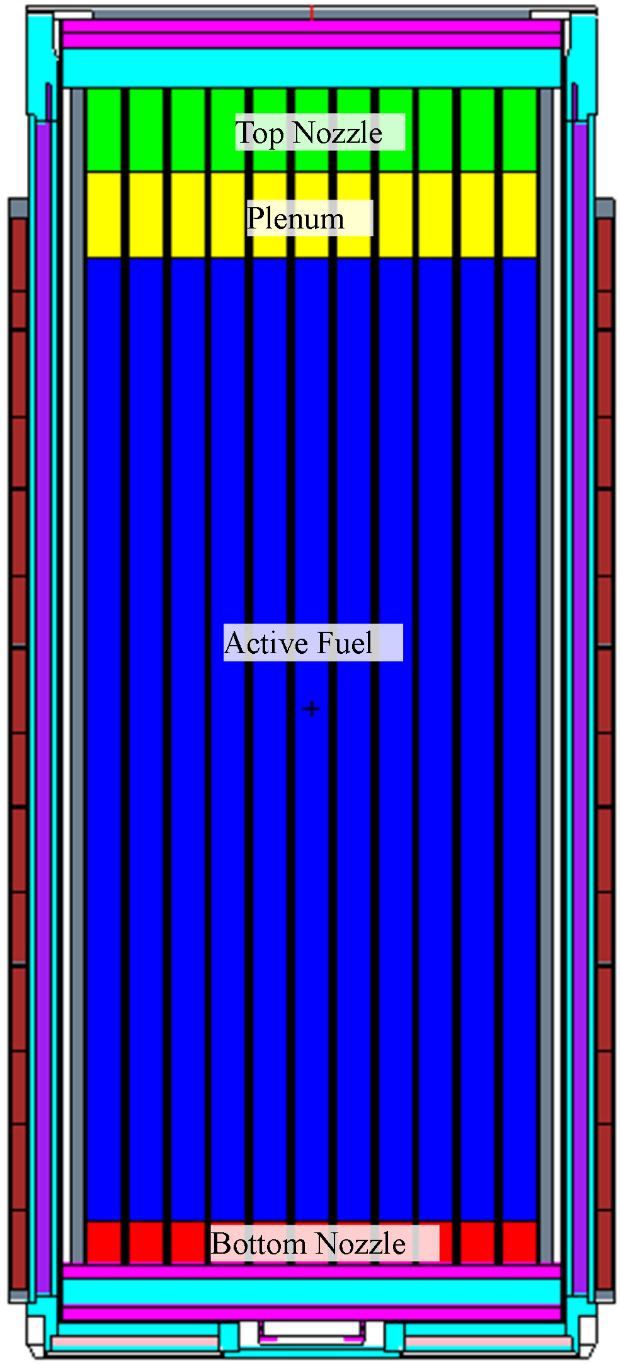
Zone Number	1	2	3
Minimum Cooling Time, Standard (years)	3.0	3.0	3.0
Minimum Cooling Time, SS304 Reconstituted (years)	15.0	15.0	15.0
Maximum Decay Heat per SFA plus channel (kW)	0.4	0.6	0.5
Number of Fuel Assemblies	29	20	40
Maximum Decay Heat per DSC (kW)	43.6		

Heat Load Zone Configuration for the EOS-89BTH in the EOS-TC108

Zone Number	1	2	3
Minimum Cooling Time, Standard (years)	3.0	3.0	9.7
Minimum Cooling Time, SS304 Reconstituted (years)	15.0	15.0	15.0
Maximum Decay Heat per SFA plus channel (kW)	0.4	0.5	0.5
Number of Fuel Assemblies	29	20	40
Maximum Decay Heat per DSC (kW)	41.6		

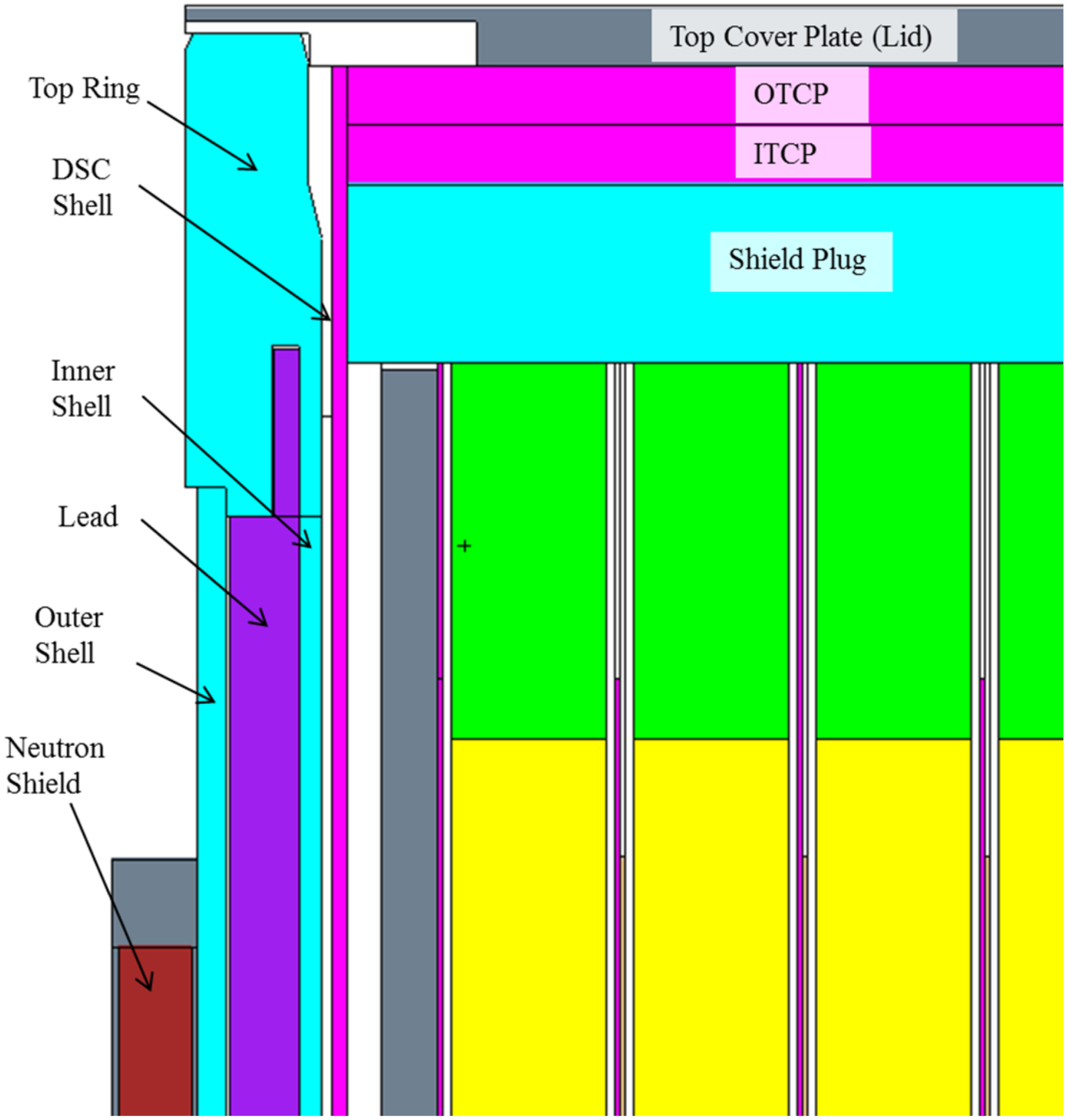
Figure 6-2
EOS-89BTH Heat Load Zone Configurations

P r o p r i e t a r y I n f o r m a t i o n
W i t h h e l d P u r s u a n t t o 1 0



EOS-89BTH DSC in EOS-TC108 depicted; other configurations are similar.

Figure 6-4
General EOS-TC MCNP Model, x-z View



EOS-89BTH DSC in EOS-TC108 depicted; other configurations are similar.

Figure 6-5
Detailed Upper View of EOS-TC MCNP Model

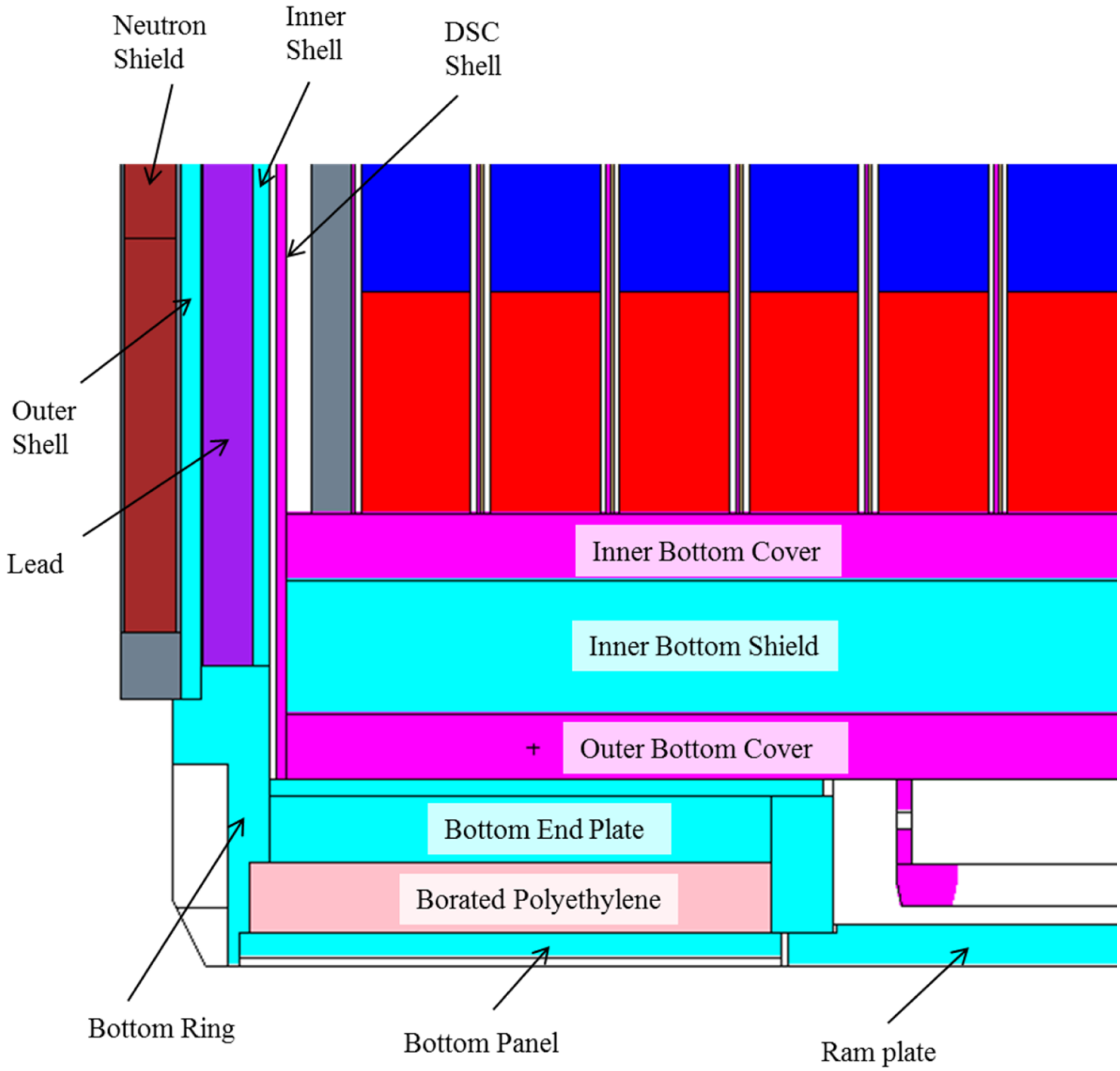
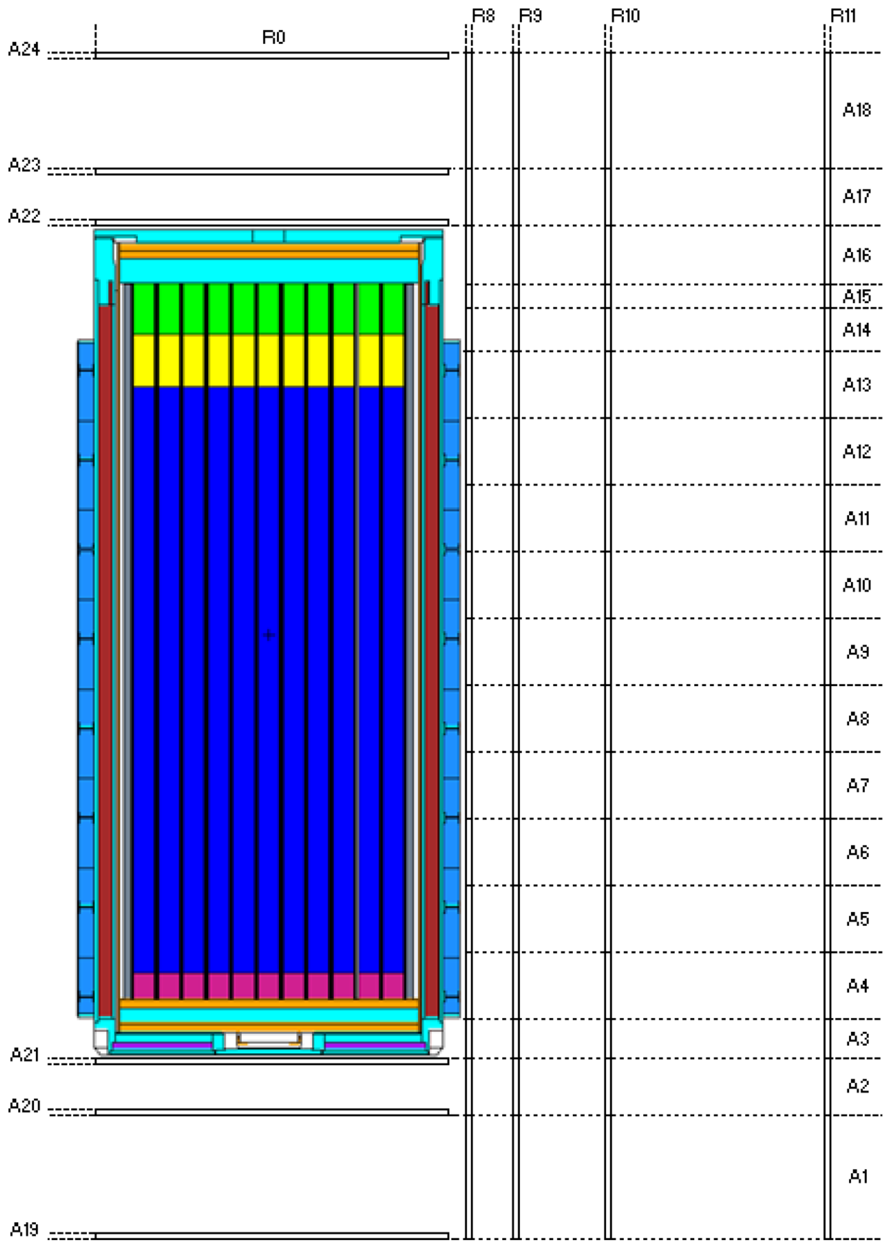
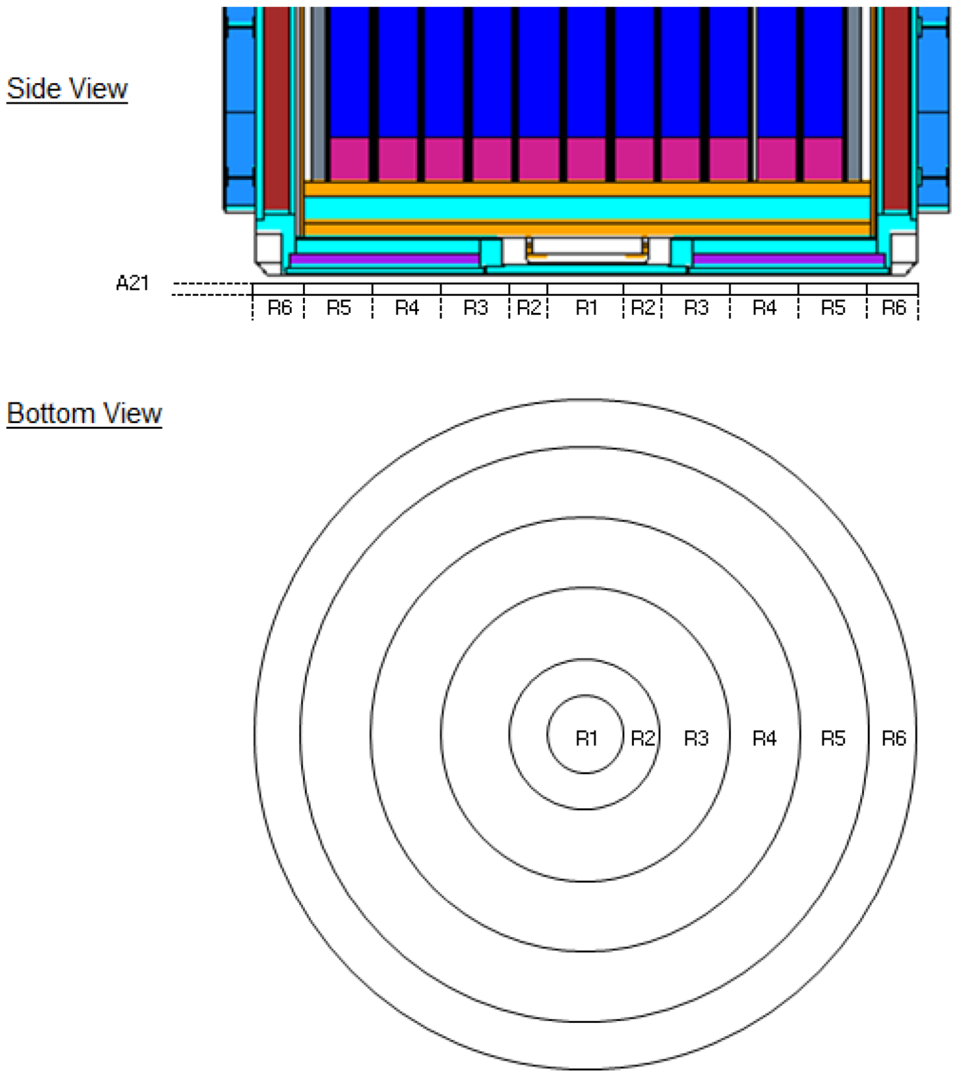


Figure 6-6
Detailed Lower View of EOS-TC MCNP Model



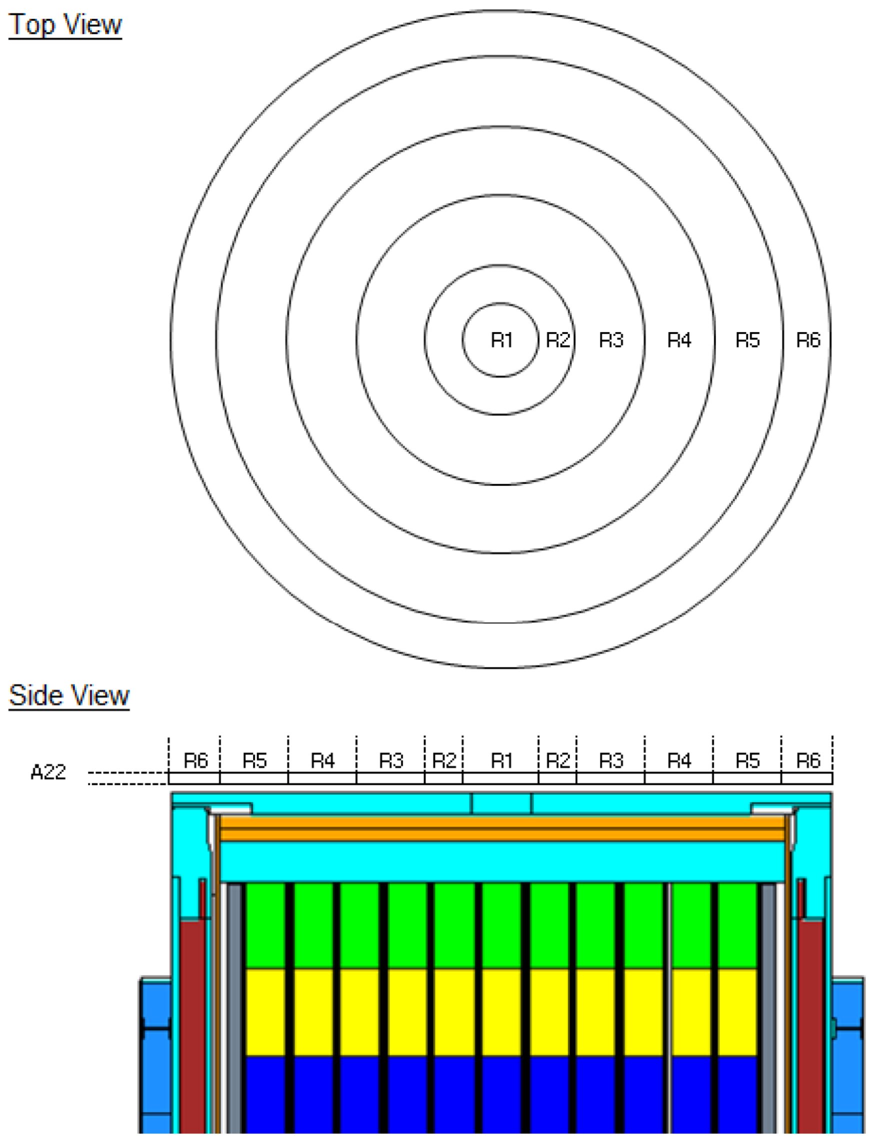
EOS-89BTH DSC in EOS-TC125/135 depicted; other configurations are similar.

Figure 6-7
EOS-TC General Dose Rate Tally Locations



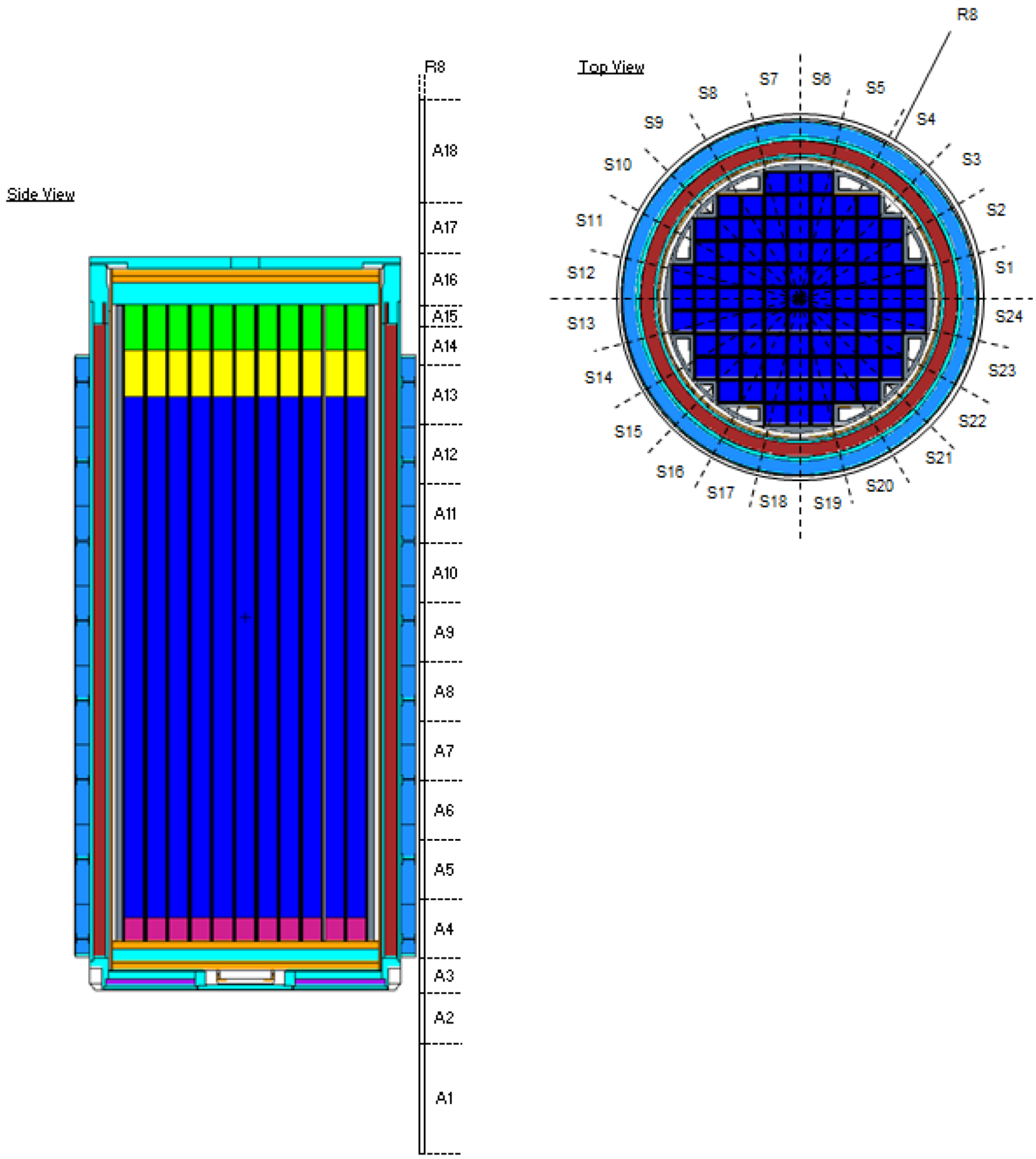
EOS-89BTH DSC in EOS-TC125/135 depicted; other configurations are similar.

Figure 6-8
EOS-TC Bottom Dose Rate Tally Locations, Transfer Configuration



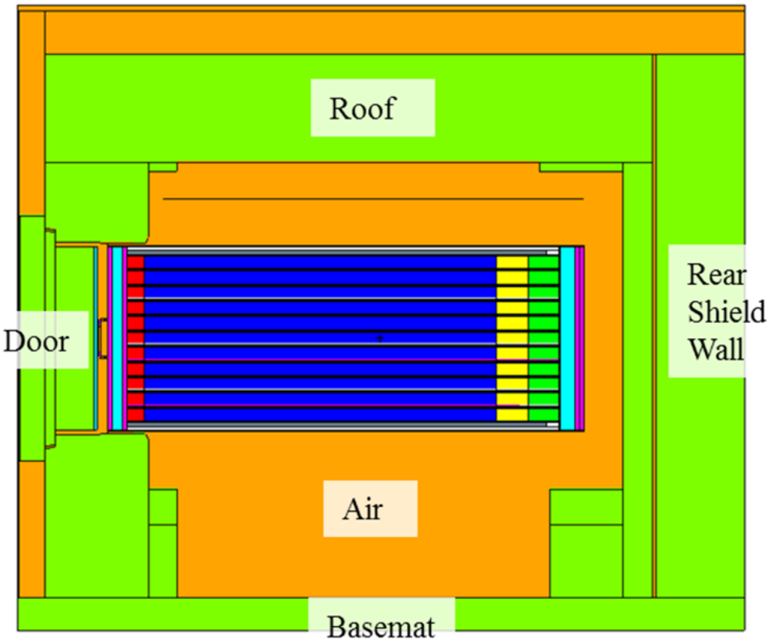
EOS-89BTH DSC in EOS-TC125/135 depicted; other configurations are similar.

Figure 6-9
EOS-TC Top Dose Rate Tally Locations, Transfer Configuration

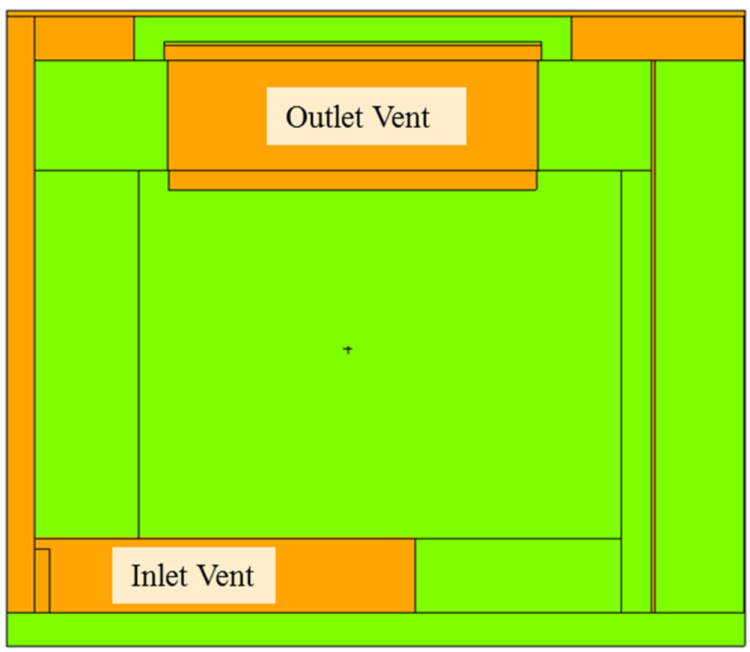


EOS-89BTH DSC in EOS-TC125/135 depicted; other configurations are similar.

Figure 6-10
EOS-TC Side Dose Rate Tally Locations, Transfer Configuration



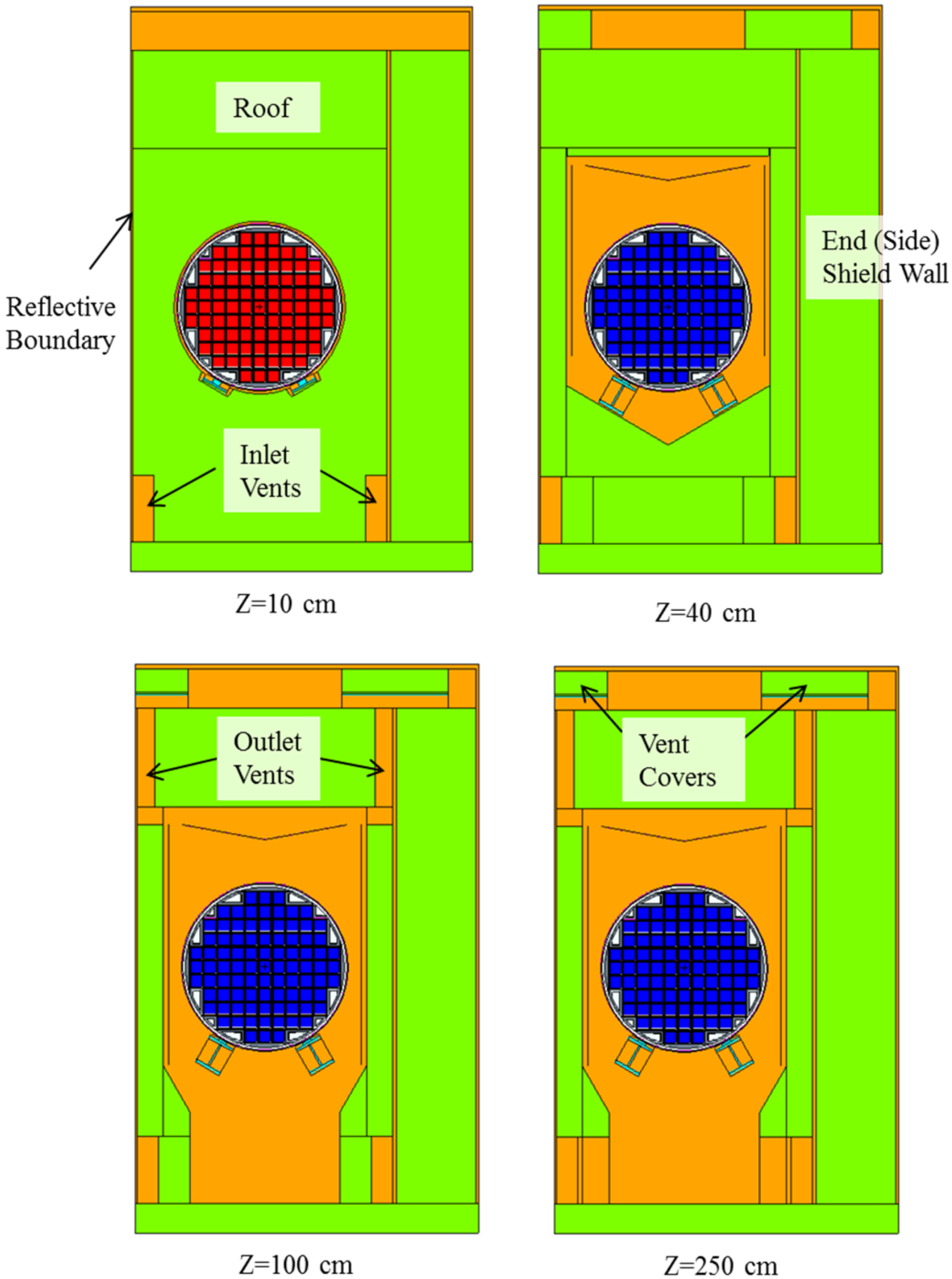
X = 0 cm



X = -143 cm

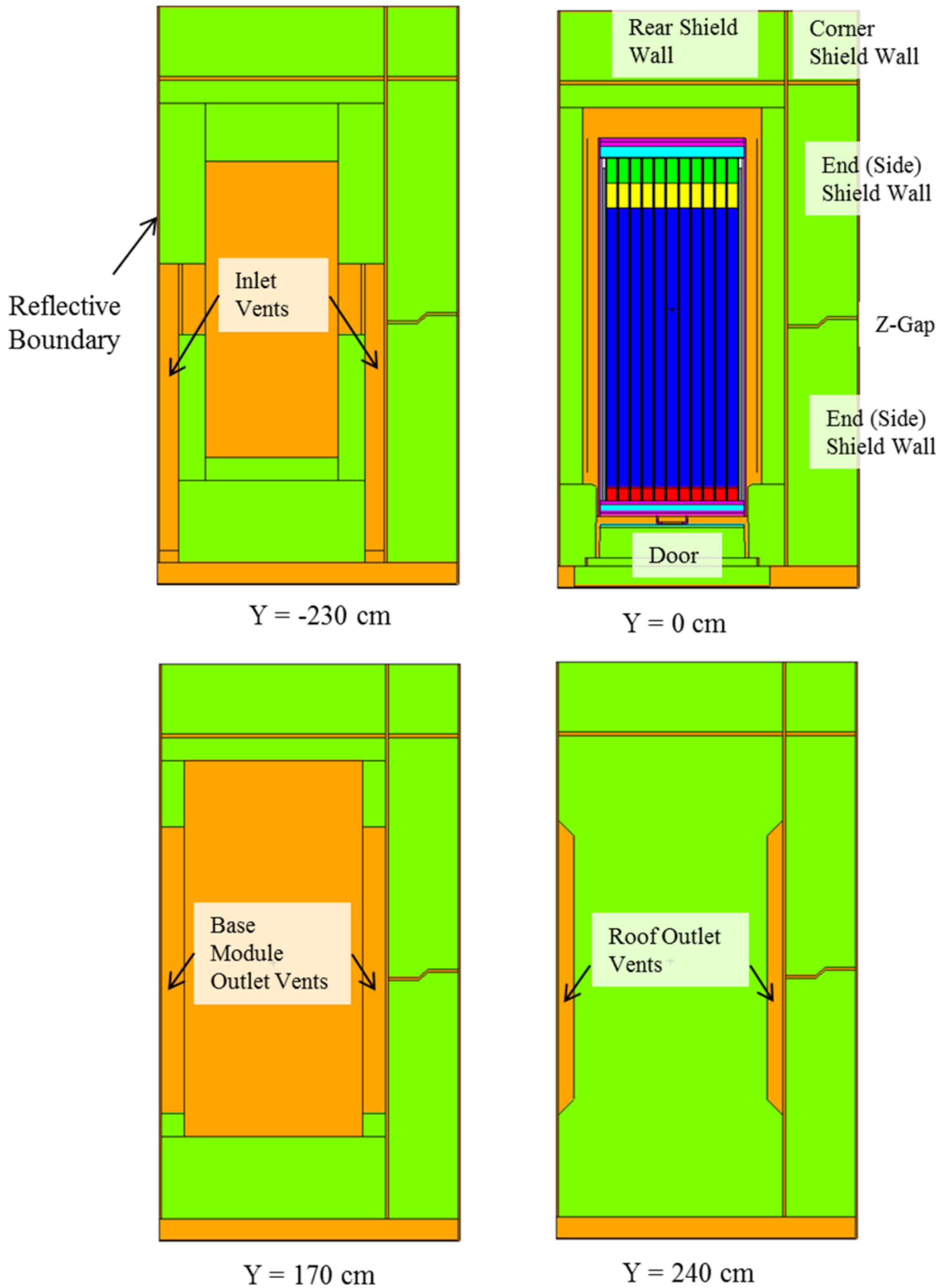
EOS-89BTH DSC in EOS-HSM-Medium depicted; other configurations are similar.

Figure 6-11
EOS-HSM MCNP Single-Reflection Model, z-y View



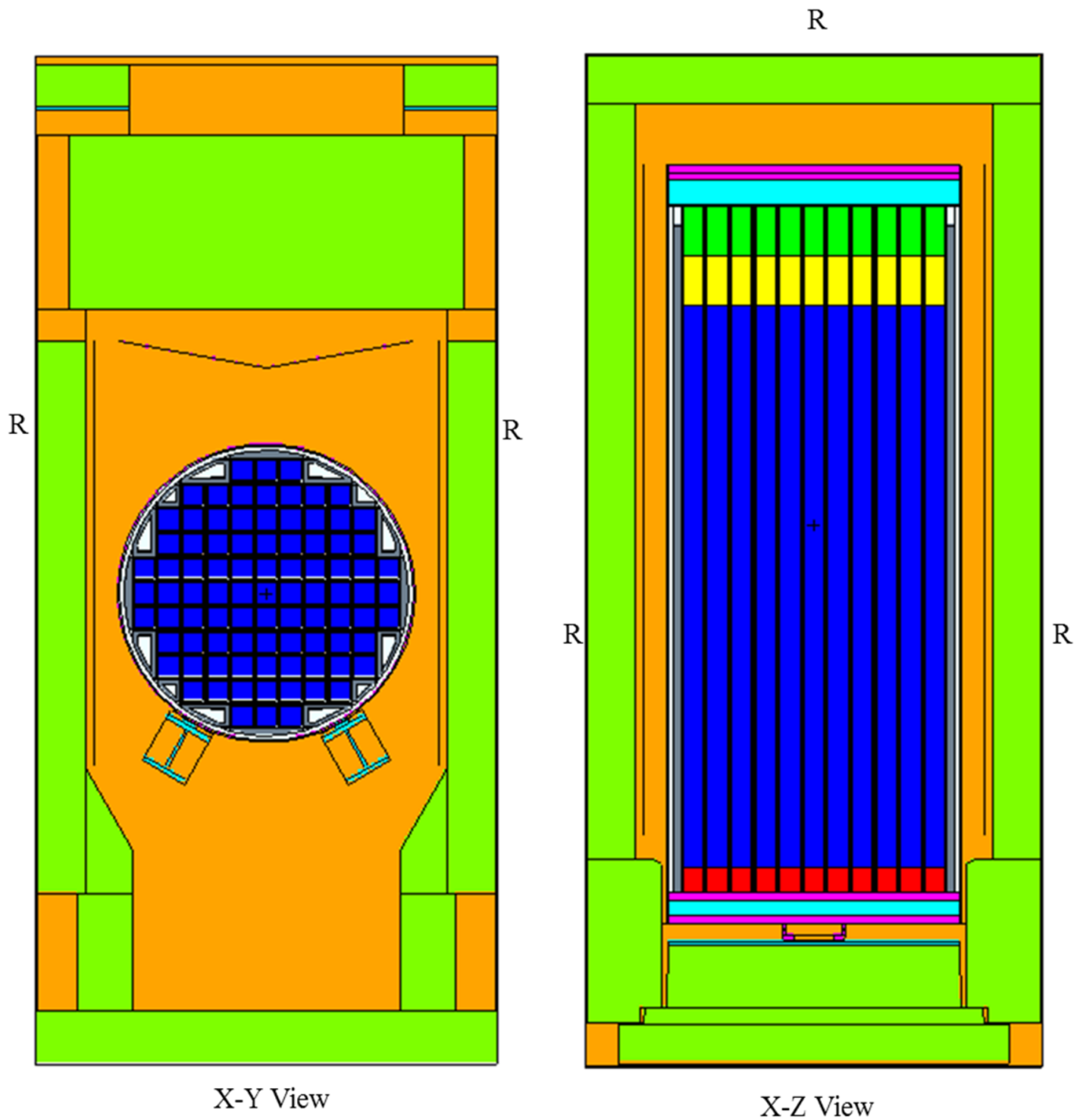
EOS-89BTH DSC in EOS-HSM-Medium depicted; other configurations are similar.

Figure 6-12
EOS-HSM MCNP Single-Reflection Model, x-y View



EOS-89BTH DSC in EOS-HSM-Medium depicted; other configurations are similar.

Figure 6-13
EOS-HSM MCNP Single-Reflection Model, x-z View



Reflective surfaces are denoted with an "R".

Figure 6-14
EOS-HSM MCNP Triple-Reflection Model

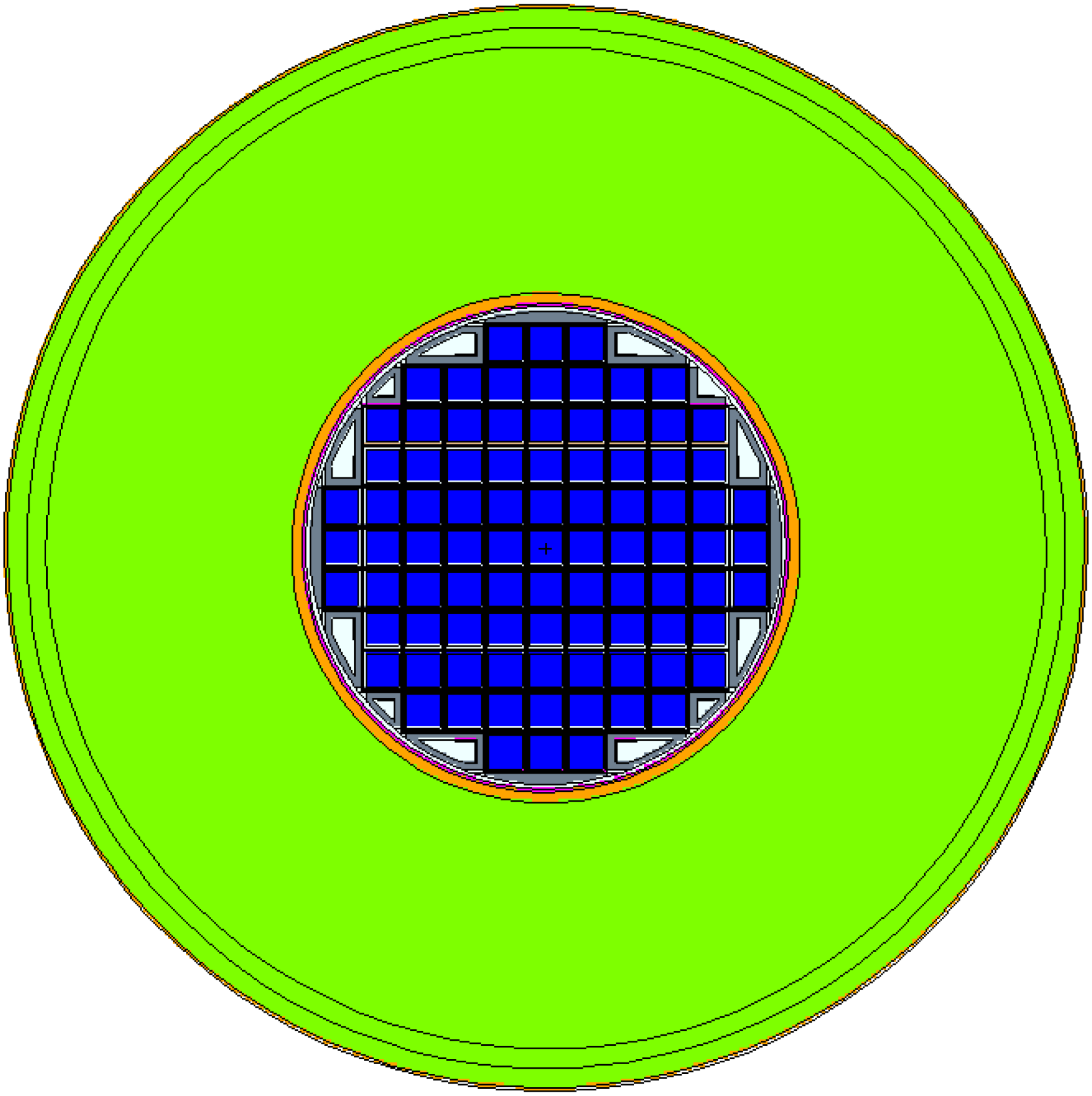


Figure 6-15
EOS-HSM MCNP Simplified Model for Grout Study

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CHAPTER 7
CRITICALITY EVALUATION

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7. CRITICALITY EVALUATION

NOTE: The referenced basket types throughout this chapter are based on the boron content in the poison plates. The term "basket types" in this chapter differs from the definition of the basket types in Chapter 1. The correlations between the basket types used in this chapter and the basket types identified in Chapter 1 are clarified below:

	Basket Type Identification in Chapter 1 and Technical Specifications	**Basket Type Identification in Chapter 7**
EOS-37PTH	A1/A2/A3	A
	B1/B2/B3	B
EOS-89BTH	A1/A2/A3	M1-A
	B1/B2/B3	M1-B
	C1/C2/C3	M2-A

The design criteria for the NUHOMS® EOS System dry shielded canisters (DSCs) require that the fuel loaded in the EOS-37PTH and EOS-89BTH DSCs remain subcritical under normal, off-normal and accident conditions as defined in 10 CFR Part 72. The criticality analyses performed to demonstrate that these DSCs satisfy the stated requirements are presented in this chapter.

The DSCs consist of a shell assembly, an internal basket assembly, and extruded aluminum open section transition rails that provide the transition to a cylindrical exterior surface to match the inside surface of the shell. The EOS-37PTH DSC is designed to store and transport up to 37 pressurized water reactor (PWR) fuel assemblies (FAs) with or without control components (CCs) while the EOS-89BTH is designed to store and transport up to 89 boiling water reactor (BWR) FAs with and without channels. The DSCs are of variable length to match the length of the fuel and CCs, as applicable, to be stored. The basket is composed of interlocking slotted plates to form an egg-crate type structure. The egg-crate structure forms a grid of 37 or 89 fuel compartments that house the spent fuel assemblies (SFAs). The egg-crate structure is composed of steel alloy, aluminum alloy, and poison plates.

The EOS-37PTH DSC criticality safety is ensured by fixed neutron absorbers, favorable geometry and credit for soluble boron in the pool during loading and unloading operations. The EOS-37PTH DSC uses a metal matrix composite (MMC) poison plate material, which is suitable for long-term use in radiation and thermal environments of a dry cask storage system. The EOS-37PTH DSC has two basket types, A and B that correspond to minimum B-10 loadings in mg B-10/cm^2, as shown in Table 7-1. In addition to the two different fixed poison loadings, soluble boron concentrations in the range 2000 to 2500 parts per million (ppm) are specified for the EOS-37PTH DSC as a function of fuel type and initial enrichment.

There are three basket types specified for the EOS-89BTH DSC. The EOS-89BTH DSC criticality safety is ensured by fixed neutron absorbers and favorable geometry. The baskets manufactured with MMC are designated M1-A and M1-B, while the baskets manufactured with BORAL® plates are designated M1-A, M1-B and M2-A, as shown in Table 7-2. In criticality evaluations, credit is taken only for 90% of B-10 areal density in the MMC and 75% in the BORAL® poison plates.

7.1 Discussion and Results

This section presents a general description of system application and criticality model for each DSC. The maximum allowable fuel enrichments as a function of basket type are specified for BWR and PWR fuel. Additionally, the minimum soluble boron concentrations (ppm) for the PWR fuel are also presented for each basket type.

The results of the evaluations demonstrate that the maximum calculated k_{eff}, including statistical uncertainty, are less than the upper subcritical limit (USL) determined from a statistical analysis of benchmark criticality experiments. The statistical analysis procedure includes a confidence band with an administrative safety margin of 0.05.

EOS-37PTH

The EOS-37PTH DSC consists of a shell assembly, which provides confinement and shielding, and an internal basket assembly, which locates and supports the PWR FAs. A detailed description of the DSC and Basket is found in Chapter 1, Section 1.2.1.1.

The FAs evaluated for storage in the EOS-37PTH DSC are considered intact. The authorized FAs are presented in Chapter 2. The payload of the EOS-37PTH DSC may include CCs that are contained within the FA. The authorized CCs are listed in Chapter 2. Reconstituted fuel assemblies with replacement rods that displace an equal amount of water as the original rods are also authorized for storage and are bounded by the intact fuel, for criticality purposes.

The criticality analysis performed uses the most reactive configuration to determine the maximum allowable enrichment for each FA class, as a function of soluble boron and basket type, with and without CCs. The results are presented in Table 7-3.

EOS-89BTH

The EOS-89BTH DSC consists of a shell assembly, which provides confinement and shielding, and an internal basket assembly, which locates and supports the BWR FAs. A detailed description of the DSC and Basket is found in Chapter 1, Section 1.2.1.2.

The FAs evaluated for storage in the EOS-89BTH DSC are considered intact. The authorized FAs are presented in Chapter 2. Reconstituted fuel assemblies with replacement rods that displace an equal or greater amount of water as the original rods are also authorized for storage and are bounded by the intact fuel, for criticality purposes.

The criticality analysis performed uses the most reactive configuration to determine the maximum allowable enrichment for the representative BWR FA design as a function of basket type. The results are presented in Table 7-4.

Computer models for the EOS-37PTH DSC and EOS-89BTH DSC are discussed in Section 7.3. The SCALE 6.0 [7-1] computer code package is employed to perform the criticality calculations. The criticality evaluation is presented in Section 7.4. The criticality benchmarks are described in Section 7.5.

7.2 Package Fuel Loading

EOS-37PTH

The EOS-37PTH DSC is designed to accommodate the FAs listed in Chapter 2. Since the assemblies listed represent a wide range of fuel types and configurations, it is demonstrated first that the most reactive case is obtained for each fuel class. In this context, the fuel class is defined as a set of fuel assembly designs with the same array size and rod pitch or rod outer diameter. The most reactive case for each class is used to determine the maximum allowable enrichments.

The payload of the EOS-37PTH DSC may include CCs that are contained within the FA. The authorized CCs are listed in Chapter 2. The only change to the package fuel loading required to evaluate the addition of the CCs is to replace the borated water in the water holes with CC materials. Since the CCs displace the borated moderator in the assembly guide and/or the instrument tubes, an evaluation is performed to determine the potential impact of the storage of CCs that extend into the active fuel region on the system reactivity. CCs that extend into the active fuel regions, such as burnable poison rod assemblies (BPRAs), control rod assemblies (CRAs), axial power shaping rod assemblies (APSRAs), control element assemblies (CEAs), and neutron source assemblies (NSAs) are conservatively assumed to exhibit the neutronic properties of $^{11}B_4C$ (no credit taken for B-10 content).

Since the criticality analysis models simulate only the active fuel height, any CC that is inserted into the FA in such a way that it does not extend into the active fuel region is considered as authorized for storage without adjustment to the soluble boron content or initial enrichment, as required for CCs that extend into the active fuel region.

EOS-89BTH

The EOS-89BTH DSC is designed to accommodate the FAs listed in Chapter 2. Since the assemblies listed represent a wide range of fuel types and configurations, a representative FA is determined by comparing k_{eff} values for the FAs listed, with and without various thicknesses of channels. The representative FA is used to obtain the maximum allowable enrichment as a function of basket type.

The maximum allowable enrichment is obtained using the GNF2 10x10 FA, except for the KKL-BWR 11/16 and SVEA-96Opt2 FAs, classified with a BWR fuel identification of ABB-10-C, in Table 7-39. The maximum allowable enrichment requirements for these two assembly designs are also specified in Table 7-4.

7.3 Model Specification

The following sections describe the physical models and materials of the EOS-37PTH and EOS-89BTH DSCs as loaded and transferred in the EOS transfer casks (TCs) (i.e., EOS-TC108, -TC125 or -TC135) used for input to KENO V.a. module of the CSAS5 sequence of SCALE 6.0 [7-1] to perform the criticality evaluation. The reactivity of the canister under storage conditions is bounded by the EOS-TC analysis with a zero internal moderator density case, which bounds the storage conditions in the EOS horizontal storage module (EOS-HSM) because: (1) the canister internals are always dry (purged and backfilled with helium) while in the EOS-HSM, and (2) the EOS-TC contains materials such as steel and lead, which provide close reflection of fast neutrons back into the basket, while the EOS-HSM materials (concrete) are much further from the sides of the DSC and thereby tend to reflect thermalized neutrons back to the canister, which are absorbed in the canister materials, reducing the system reactivity.

7.3.1 Description of Criticality Analysis Model

EOS-37PTH

The EOS-TC and DSC are explicitly modeled using the appropriate geometry options in the KENO V.a module, of the CSAS5 sequence of SCALE 6.0. Several models are developed to evaluate the fabrication tolerances of the DSC, FA type, assembly location, initial enrichment, fixed poison loading, soluble boron concentration, and storage of CCs with the fuel.

The criticality evaluation is performed using a basket section equivalent to the active fuel height with periodic axial boundary conditions, which effectively makes the model infinitely tall. The key basket dimensions utilized in the calculation are shown in Table 7-5. The basic KENO model, with a length equivalent to the active fuel height, is modeled with periodic boundary conditions axially, and reflective boundary conditions radially. The axial section essentially models an infinite active fuel height DSC. The model does not explicitly include the water neutron shield; however, the infinite array of casks without the neutron shield is conservatively modeled with unborated water between the casks and in the TC/DSC annulus. For the purpose of storage, the configuration is not expected to encounter any regions containing fresh water once the FAs are loaded. Therefore, this hypothetical configuration that models an infinite array of casks in close reflection is conservative.

The FAs within the basket are modeled as arrays of fuel pins and guide/instrument tubes. Spacer grids and subcomponents such as oversleeves are not modeled since their effect on reactivity is insignificant. The compartment plates effectively surround each FA. The basket cross section shown in Figure 7-1 illustrates that the basket is composed of eight sets of plates traversing vertically and horizontally across the DSC where this pattern occurs repeatedly in the axial region. [,

] The gaps for the egg-crate included in the starting KENO model are also illustrated in Figure 7-3. In addition, up to 1 inch of active fuel length is conservatively excluded from poison coverage as shown in Figure 7-4.

The basket structure is connected to the DSC shell by perimeter transition rail assemblies. The R45 and R90 transition rails that enable the cylindrical exterior surface of the basket to match the inside surface of the DSC are attached to the basket structure using steel fasteners that are installed at set intervals axially. The R90 rails are solid aluminum, while the R45 are open aluminum sections, held against the basket with steel plates of equal length and rail spacers of equal length that bridge the gap between an adjacent wall and a rail wall. The DSC is a 0.50-inch thick stainless steel shell with an inner diameter of 74.5 inches. The DSC dimension, the TC inner and outer shell dimensions and lead shield shell thickness are provided in Table 7-6. Note that although the dimensions provided for the 108-ton TC are not utilized in the model, an evaluation is performed to establish the bounding dimensions. The TC and DSC are explicitly modeled using the appropriate geometry options in the KENO V.a. module of the CSAS5 sequence in SCALE 6.0. The DSC and TC are a simple combination of cylindrical volumes, which are modeled in a straight forward manner. The R90 rails are formed by the virtue of the definition of the inner cylindrical volume of the DSC using the KENO geometry.

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<u>EOS-89BTH</u>

The EOS-TC and DSC are explicitly modeled using the appropriate geometry options in KENO V.a module of the CSAS5 sequence of SCALE 6.0. Several models are developed to evaluate the fabrication tolerances of the DSC, FA locations, FA type, initial enrichments, and fixed poison loading.

The nominal dimensions of the DSC and the EOS-TC models are summarized in Table 7-5 and Table 7-6. The materials modeled are listed in Table 7-8 and Table 7-9.

The model utilized is an egg-crate segment model with 12-inch height and full radial cross section of the cask and canister with periodic boundary conditions at the axial boundaries (top and bottom) and reflective boundary conditions at the radial boundaries (sides) to represent infinite arrays of package. This axial section essentially models one building block of the egg-crate basket structure. Periodic boundary conditions ensure that the resulting KENO model is essentially infinite in the axial direction. The model does not include the water neutron shield; however, the infinite array of casks without the neutron shield may contain unborated water between the casks and in the canister/TC gap.

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The criticality calculations are performed to determine:

- The most reactive fuel (MRF),
- The most reactive configuration (MRC) and
- The maximum allowable enrichments for the various B-10 loadings in the poison plates.

The MRF and MRC evaluations make use of the design specifications of the basket that differ from the actual design as described above.

The extruded aluminum open-section transition rails, which are reinforced with internal steel as necessary, provide the transition to a rounded surface to match the inside surface of the DSC shell. The transition rail geometry is modeled as a mixture of aluminum and water in the model and calculations are performed to determine the optimal ratio of the water in the rails. Pure aluminum is assumed in the starting KENO or base model for the MRF analyses.

The basket egg-crate structure is modeled with the various radial and axial gaps that are filled with internal moderator. These gaps are illustrated in Figure 7-19. Further, the design details summarized in Table 7-11 are employed to calculate the egg-crate slot width while the gap dimensions in Table 7-12 are employed in a sensitivity evaluation as part of the MRC analysis.

Figure 7-14 and Figure 7-15 show the cross section views of the base model. The GE12 10X10 FA is used with an enrichment of 4.20 wt. % U-235. The poison plate is modeled with a B-10 areal density of 29 mg B-10 per cm^2. The dimensions for TC108 are used and a 0.12-inch thick fuel channel is included. The FA is placed inwardly in the fuel compartment to have the FAs as close as possible. Figure 7-16 shows the radial cross section of the center fuel compartment in the base model. Figure 7-17 shows the radial cross section of the fuel pin. The gap between the cladding and the fuel pin is modeled as water. Figure 7-18 shows the plate thicknesses for the horizontal plates and the vertical plate thickness are the same as the horizontal plates. Note that the plate thicknesses presented in Table 7-5 are used in the MRF and MRC evaluations.

The EOS-89BTH DSC, as transferred in the EOS-TC, is capable of housing standard BWR FAs with or without fuel channels and as intact FAs. The FAs considered as authorized contents are listed in the Chapter 2. A unique identifier is associated with a modeled FA design as described in Table 7-39. The FA layouts are shown in Figure 7-20.

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The inward placement of FAs is chosen for the MRF analysis compared to the FAs centered in their fuel compartments since it leads to a higher system reactivity, which is also confirmed in the MRC analysis. The radial cross section view of one sample model for inwardly placed FAs is shown in Figure 7-21 and Figure 7-22.

It is assumed that the gap between the DSC and TC is filled with water in the MRF analysis. The optimum water density resulting in the highest system reactivity is investigated in the MRC analysis.

The TCs are cylindrical, shielded vessels constructed from carbon steel and lead. It is assumed that the neutron shielding on the TC has vanished and is not included in the model.

The outside of the TC is assumed to be void in the MRF analysis. The optimum water density for the highest system reactivity is investigated in the MRC analysis.

7.3.2 Package Regional Densities

The Oak Ridge National Laboratory (ORNL) SCALE 6.0 [7-1] code package contains a standard material data library for common elements, compounds, and mixtures. All materials used for the TC and canister analyses are available in this data library.

A list of the relevant materials used for the criticality evaluation is provided in Table 7-8 and Table 7-9. The poison plate is MMC for the EOS-37PTH DSC. The poison plate is either MMC or BORAL® for the EOS-89BTH DSC. The poison plate material specifications are modeled considering a 90% B-10 credit for the B-10 loading in the MMC and 75% credit for the B-10 loading in BORAL®.

7.4 Criticality Calculation

This section describes the various criticality evaluations carried out for the NUHOMS® EOS System DSCs. The analyses are performed with the CSAS5 module of the SCALE 6.0 code system. The most reactive configuration for each DSC is determined with consideration of fabrication tolerances, FA location, and basket dimensions important to criticality. A series of calculations is performed to determine the relative reactivity of the various FA designs to determine the most reactive assembly type. Finally, the maximum allowable enrichment for the minimum required criticality control mechanism, which is soluble boron and basket type for PWR FAs, and basket type for BWR FAs, is determined.

EOS-37PTH

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EOS-89BTH

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7.4.1 Calculational Method

7.4.1.1 Computer Codes

The CSAS5 control module of SCALE 6.0 [7-1], is used to calculate the effective multiplication factor (k_{eff}) of the fuel in the TC (bounds fuel in HSM). The CSAS5 control module allows simplified data input to the functional modules BONAMI, NITAWL, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL applies Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. Enough neutron histories are run so that the standard deviation is below 0.0010 for all calculations.

7.4.1.2 Physical and Nuclear Data

The physical and nuclear data required for the criticality analysis include the FA data and cross section data as described below.

The pertinent data for criticality analysis for each FA evaluated in the EOS-37PTH DSC and EOS-89BTH DSC are listed in Chapter 2. The criticality analysis uses the 44-group cross section library built into the SCALE 6.0 system. Oak Ridge National Laboratory used ENDF/B-V data to develop this broad-group library, specifically for the criticality analysis of a wide variety of thermal systems.

7.4.1.3 Bases and Assumptions

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The following assumptions are employed in the criticality calculations for intact fuel.

EOS 37PTH and EOS-89BTH

- Fresh fuel is assumed. No credit is taken for fissile depletion, fission product poison, or burnable absorbers.
- Fuel rods are filled with full density fresh water in the pellet-clad gap.

- The neutron shield and steel neutron shield jacket (outer skin) of the cask are conservatively removed and infinite arrays of casks are pushed close together with external moderator (unborated water) in the interstitial spaces.
- The MMC poison plates are modeled with minimum specified B-10 content required for safety.
- Temperature is 20 °C (293K).
- All steel and aluminum alloys of the basket structure are modeled as SS304 and aluminum, respectively. While these compositions, which are provided in the SCALE 6.0 standard composition library, have small differences with compositions of the various steels and aluminum, they have negligible effect on the results of the calculation.
- All zirconium-based materials in the fuel are modeled as Zircaloy-4 for PWR and Zircaloy-2 for BWR fuel evaluations. The small differences in the composition of the various clad/guide compartment materials have negligible effect on the results of the calculations.
- Omission of grid plates, spacers, and hardware in the FA.
- No integral burnable absorbers, such as gadolina, erbia or any other absorbers, are included.
- The fuel rods are modeled assuming a stack density of 97.5% theoretical density with no allowance for dishing or chamfer in the fuel rod model, which conservatively bounds the total fuel content in the FA authorized for storage.

<u>EOS 37PTH Only</u>

- Non-fuel assembly hardware that extends into the active fuel regions, such as BPRAs, CRAs, APSRAs, CEAs, and NSAs, are conservatively assumed to exhibit the neutronic properties of $^{11}B_4C$ (no credit taken for B-10 content). There is negligible neutron absorption from any of this hardware and it is collectively referred to as CCs.
- Water in the EOS-37PTH DSC cavity contains soluble boron at optimum density. The soluble boron is mixed with the moderator. By varying the moderator density from 50% to 100% of full density, the density of water at which the reactivity is maximized is determined.
- For intact fuel, the maximum planar average initial fuel enrichment is modeled as uniform everywhere throughout the assembly. Natural uranium blankets and axial or radial enrichment zones are modeled as enriched uranium at the planar average initial enrichment.
- The portion of the DSC corresponding to the axial length of the active fuel is modeled for the criticality analysis. The axial ends of the DSC are not modeled.

EOS 89BTH Only

- Only one section of height (12 in.) equal to one egg-crate section of the basket is modeled with periodic boundary conditions at the axial boundaries (top and bottom) and reflective boundary conditions at the radial boundaries (sides) to represent infinite long FAs an infinite arrays of package. From a criticality standpoint, modeling a repetitive egg-crate section with periodic axial boundary conditions or a full active fuel length with periodic axial boundary conditions (EOS-37PTH) that results in an infinite axial length are not different.
- For intact fuel, the pins are modeled assuming the maximum lattice average enrichment uniformily everywhere in the lattice. Natural uranium blankets, gadolinia, integral fuel burnable absorber (IFBA), erbia, or any other burnable absorber rods and axial or radial enrichment zones are modeled as uranium with the maximum lattice average enrichment.
- Water density is at optimum internal and external moderator density.
- It is assumed that the fuel rod outer diameter varies by ± 0.005 inch for this evaluation.

7.4.1.4 Determination of k_{eff}

The Monte Carlo calculations performed with CSAS5 (KENO V.a) use a flat neutron starting distribution. The total number of histories traced for each calculation is at least 800,000. This minimum number of histories is sufficient to achieve source convergence and produce standard deviations of less than 0.0010. The maximum k_{eff} for the calculation is determined with the following formula:

$$k_{eff} = k_{keno} + 2\sigma_{keno}$$

7.4.2 EOS 37PTH Fuel Loading Optimization

The first set of evaluations is performed to determine the most reactive configuration. This is achieved by modifying the base case model as described in Section 7.3.1. The most reactive configuration is used to compute the k_{eff} values of the various FAs provided in Chapter 2 as modeled in the DSC to determine the most reactive FA (in the second set of evaluations) for each FA class. Finally, the maximum allowable enrichment and criticality control requirements for each FA class is determined.

Proprietary Information on Pages 7-16 through 7-19
Withheld Pursuant to 10 CFR 2.390

B. Determination of the Most Reactive Fuel Type

The most reactive configuration obtained through the evaluations described in Section 7.4.2.A is used to determine the most reactive FAs for each class listed in Chapter 2.

The fuel designs listed in Chapter 2 are evaluated to determine the most reactive FA type with initial enrichment of 5.0 wt. % U-235, water with 2000 ppm soluble boron with 100% internal moderator density and poison plate B-10 content corresponding to the Type B basket specified in Table 7-1, or 31.5 mg/cm^2 B-10 in the model. The results are presented in Table 7-16 and a representative FA design from each class is selected for further evaluation. These fuel designs shown in **BOLD TEXT** in Table 7-16.

C. Determination of the Maximum Initial Enrichment for Assembly Class

The analysis performed in this section uses the most reactive fuel type for each assembly class to determine the maximum allowable planar average initial enrichment as a function of basket assembly type (poison plate loading) and soluble boron concentration. No credit is taken for burnup. Only the fixed poison plate loading is changed for each model. In addition, for each case the internal moderator density is varied to determine the peak reactivity for the specific configuration.

The EOS-37PTH DSC/EOS-TC model for this evaluation differs from the actual design in the following ways:

- The B-10 content is at the minimum required for reactivity control for each basket assembly type,
- The neutron shield and the neutron shield jacket (outer skin) of the EOS-TC are conservatively removed and EOS-TC pushed together with fresh water between the casks,
- The dimensions with limiting fabrication tolerances, as determined by the evaluations described in this section, are modeled,

WE 17x17 Class Fuel Assemblies

The most reactive WE 17x17 class FA is the WE 17x17 LOPAR assembly. The results for the WE 17x17 class assembly calculations without and with CCs in the Type A basket are presented in Table 7-17 and Table 7-18, respectively. The results in the Type B basket are presented in Table 7-19 and Table 7-20.

BW 15x15 Class Assemblies

The most reactive B&W 15x15 class assembly is the Mark B10 assembly. The results for the B&W 15x15 class assembly calculations without and with CCs in the Type A basket are presented in Table 7-21 and Table 7-22, respectively. The results in the Type B basket are presented in Table 7-23 and Table 7-24.

WE 15x15 Class Assemblies

The most reactive WE 15x15 class assembly is the Tihange 1 WE 15x15 assembly. The results for the WE 15x15 class assembly calculations without and with CCs in the Type A basket are presented in Table 7-25 and Table 7-26, respectively. The results in the Type B basket are presented in Table 7-27 and Table 7-28.

CE 15x15 Class Assemblies

The most reactive CE 15x15 class assembly is the Palisades assembly. The results for the CE 15x15 class assembly calculations without and with CCs in the Type A basket are presented in Table 7-29 and Table 7-30, respectively. The results in the Type B basket are presented in Table 7-31 and Table 7-32.

CE 14x14 Class Assemblies

The most reactive CE 14x14 class assembly is the Framatome CE assembly. The results for the CE 14x14 class assembly calculations without and with CCs in the Type A basket are presented in Table 7-33 and Table 7-34, respectively. These results indicate that the maximum allowable enrichment of 5.0 wt. % U-235 is obtained using type A basket and a soluble boron concentration of 2000 ppm.

14x14 Class Assemblies

The most reactive 14x14 class assembly is the WE 14x14 Std/LOPAR/ZCA/ZCB assembly. The results for the 14x14 class assembly calculations with and without CCs in the Type A basket are presented in Table 7-35 and Table 7-36, respectively. These results indicate that the maximum allowable enrichment of 5.0 wt. % U-235 is obtained using type A basket and a soluble boron concentration of 2000 ppm.

CE 16x16 Class Assemblies

The most reactive CE 16x16 class assembly is the AREVA design assembly. The results for the CE 16x16 class assembly calculations without and with CCs in the Type A basket are presented in Table 7-37 and Table 7-38, respectively. These results indicate that the maximum allowable enrichment of 5.0 wt. % U-235 is obtained using type A basket and a soluble boron concentration of 2000 ppm.

7.4.3 EOS 89BTH Fuel Loading Optimization

As described in Section 7.3.1, the base model is employed to perform criticality calculations to determine the MRF design. A representative FA design is chosen based on the MRF analysis to determine the MRC of the system. By using the representative fuel and the MRC, maximum allowable and enrichment as a function of basket type (B-10 loading) for BWR FAs in the EOS-89BTH DSC are determined to ensure the system k_{eff} is below the USL.

Proprietary Information on Pages 7-23 through 7-25
Withheld Pursuant to 10 CFR 2.390

C. Determination of the Maximum Initial Enrichment for BWR Fuel Assemblies

The design basis KENO model with the GNF-2 fuel assembly design is employed to determine the maximum allowable initial enrichment for the three allowable fixed poison loadings. The KENO model employed herein incorporates the bounding modeling features evaluated in the previous evaluations and also is consistent with the actual design dimensions as discussed in the Section 7.3.1. The results of the criticality analyses are shown in Table 7-43. These results demonstrate that the maximum k_{eff} of the system remains below that USL with a maximum enrichment of 4.80 wt. % U-235.

As described in the MRF analyses, separate enrichment limits are determined for the ABB-10-C type BWR fuel assemblies and also shown in Table 7-43. These results indicate that the maximum allowable enrichment is reduced by 0.25 wt. % U-235 for the Type M1-A and Type M1-B poison loading and by 0.20 wt. % U-235 for the Type M2-A poison loading compared to that for the GNF-2 fuel assembly.

7.4.4 Criticality Results

In Table 7-44, a summary of the bounding scenarios that exist for both the EOS-37PTH and EOS-89BTH are presented. These are: dry storage condition, applicable to the DSC and placed in the EOS-HSM, normal loading or unloading operation where the DSC is in the fuel pool with 100% internal moderator density, and condition where the internal moderator density is at the optimum calculated for maximum reactivity.

For the EOS-37PTH the most reactive case for the normal loading or unloading condition is calculated for the CE 15x15 class FA with 4.75 wt. % U-235, Type B basket, without CCs and 2000 ppm of soluble boron 100% internal moderator density, which is also the most optimum density. For the dry storage condition, this CE 15x15 case is modified by changing the internal and external moderator density to air, because this results in the a bounding dry condition scenario.

For the EOS-89BTH the most reactive case for the normal loading or unloading condition is calculated for the GNF2 FA with 4.80 wt. % U-235, Type M2-A basket and 100% internal moderator density, which is also the optimum density. For the dry storage condition, this GNF2 case is modified by changing the internal and external moderator density to air as this results in a bounding dry condition scenario.

The criterion for subciritality is that:

$$k_{keno} + 2\sigma_{keno} < USL$$

where USL is the upper subcriticality limit established by an analysis of benchmark criticality experiments. From Section 7.5 the USL for the EOS-37PTH DSC is 0.9404 while the USL for the EOS-89BTH DSC is 0.9418.

From Table 7-44, the most reactive case determined for PWR fuel storage is:

$$k_{keno} + 2\sigma_{keno} = 0.9371+2*0.0007=0.9385 < 0.9404,$$

From Table 7-44, the most reactive case determined for BWR fuel storage is:

$$k_{keno} + 2\sigma_{keno} = 0.9382+2*0.0008=0.9398 < 0.9418.$$

7.5 Critical Benchmark Experiments

The criticality safety analysis of the EOS-37PTH and EOS-89BTH system used the CSAS5 module of the SCALE 6.0 system of codes. The CSAS5 control module allows simplified data input to the functional modules BONAMI, NITAWL, and KENO V.a. These modules process the required cross section data and calcualte the k_{eff} of the system. BONAMI performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the effective neutron multiplaciton (k_{eff}) of a three-dimensional system.

The analysis presented herein uses the fresh fuel assumptions for criticality analysis. The analysis employed the 44-group ENDF/B-V cross section library because it has a small bias, as determined by the benchmark calculations. A total of 92 experiments are available. The 92 experiments are each selected for benchmarking system applications that utilize soluble boron (EOS-37PTH), while 51 of the experiments are selected to benchmark system applications that utilize water without soluble boron (EOS-89BTH). The upper subcritical limit (USL) was determined using the results of the benchmark experiments.

The experimental problems used to perform the benchmarking are selected after comparison of relevant features to the system applications as presented in Table 7-45. The comparison of fuel and structural materials in the table demonstrates the experiments utilize materials that are expected to produce a neturonic behavior found in the system applications. The experiments along with pertinent parameters are listed in Table 7-46.

The USL is dependent on the set of evaluated critical experiments where the models in the experiments must have features similar to the system evaluated. The expectation is that the final calculated k_{eff} of the the selected critical experiments. The experiments in general have similar features or parameters in common such that a trend of how these affect the final k_{eff} due to the limitations associated with modeling, calculation methodology and nuclear cross-section data, can be evaluated. The features or parameters considered are U-235 enrichment, fuel pitch (cm), energy of average lethargy of fission (EALF), average energy group causing fission (AEG), soluble boron (ppm), assembly separation (cm), and moderator-to-fuel volume ratio. Using the relevant parameters, the correlation (r-value) of the parameters to the k_{eff} of the experiments must be evaluated to assess the level of influence of the parameter on the system reactivity. The USLSTATS code [7-5] provides the means to obtain the USL functions that can be used to obtain the final USL value if it can be shown that the parameters are closely correlated with k_{eff}, that it, $|r|$, is nearly 1.0. As demonstrated in Section 7.5.2, there is no close correlation between the parameters and k_{eff}. In cases where no closely correlated parameters exist, the single-sided tolerance limit methodology described in NUREG/CR-6698 [7-6] is used, it can be shown that the k_{eff} values are normally distributed, which is demonstrated in Section 7.5.2.

Proprietary Information on Pages 7-29 through 7-31
Withheld Pursuant to 10 CFR 2.390

7.5.3 Results of the Benchmark Calculations

The k_{eff} values of the 92 experiments are examined to determine correlation against the independent parameters listed in Section 7.5.2. The results in Table 7-47 indicate that there is no close correlation. The k_{eff} values are normally distributed and therefore, a single-sided lower tolerance limit USL is computed according to the methodology described in NUREG/CR-6698. The USL for the EOS-37PTH DSC is 0.9404. The results are summarized in Table 7-48.

The highest k_{eff} obtained for fuels loaded in the EOS-37PTH DSC is 0.9371+2*0.0007=0.9385, which is less than the USL of 0.9404.

The k_{eff} values of the 51 experiments are examined to determine correlation against the independent parameters listed in Section 7.5.2. The results in Table 7-47 indicate that there is no close correlation. The k_{eff} values are normally distributed and, therefore, a single-sided lower tolerance limit USL is computed according to the methodology described in NUREG/CR 7109. The USL for the EOS-89BTH DSC is 0.9418. The results are summarized in Table 7-48.

The highest k_{eff} obtained for fuels loaded in the EOS-89BTH DSC is 0.9382+2*0.0008=0.9398, which is less than the USL of 0.9418.

7.6 References

7-1 SCALE 6: Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers, Oak Ridge National Laboratory, Radiation Shielding Information Center Code Package CCC-750, February 2009.

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7-4 International Criticality Safety Benchmark Evaluation Project (ICSBEP), “International Handbook of Evaluated Criticality Safety Benchmark Experiments,” NEA/NSC/DOC(95)03, NEA Nuclear Science Committee, September 2009, http://icsbep.inel.gov/.

7-5 USLSTATS: A Utility to Calculate Upper Subcritical Limits for Criticality Safety Applications, Version 6, Oak Ridge National Laboratory, January 26, 2009.

7-6 Dean, J.C., Tayloe Jr., R.W., “Guide for Validation of Nuclear Criticality Safety Calculational Methodology”, NUREG/CR-6698, January 2001.

Table 7-1
EOS-37PTH Minimum B-10 Content in the Neutron Poison Plates

Basket Type	Minimum B-10 Content for MMC (mg/cm^2)	B-10 Content Used in Criticality Evaluation (mg/cm^2)
A	28.0	25.2
B	35.0	31.5

Table 7-2
EOS-89BTH Minimum B-10 Content in the Neutron Poison Plates

Basket Type	B-10 Content Used in Criticality Evaluation (mg/cm^2)	Minimum B-10 Areal Density (mg/cm^2)	
		MMC	BORAL®
M1-A	29.4	32.7	39.2
M1-B	37.2	41.3	49.6
M2-A	45.0	-	60.0

Table 7-3
EOS-37PTH Maximum Planar Average Initial Enrichment
(2 Pages)

Fuel Assembly Class	Maximum Assembly Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)				
	Minimum Soluble Boron (ppm)	Basket Type			
		A		B	
		w/o CCs	w/ CCs	w/o CCs	w/ CCs
17x17 Assembly Class(1)	2000	4.35	4.35	4.50	4.45
	2100	4.50	4.45	4.65	4.60
	2200	4.60	4.55	4.75	4.70
	2300	4.70	4.65	4.85	4.85
	2400	4.85	4.80	5.00	4.95
	2500	4.95	4.90	-	5.00
CE 16x16 Assembly Class	2000	5.00	5.00	5.00	5.00
	2100	-	-	-	-
	2200	-	-	-	-
	2300	-	-	-	-
	2400	-	-	-	-
	2500	-	-	-	-
BW 15x15 Assembly Class	2000	4.25	4.20	4.40	4.35
	2100	4.40	4.30	4.55	4.45
	2200	4.50	4.45	4.65	4.60
	2300	4.60	4.55	4.80	4.70
	2400	4.75	4.65	4.90	4.85
	2500	4.85	4.75	5.00	4.90
15x15 Assembly Class(1) (excludes BW 15x15 and CE 15x15)	2000	4.45	4.40	4.55	4.55
	2100	4.60	4.55	4.65	4.65
	2200	4.70	4.65	4.80	4.80
	2300	4.85	4.75	5.00	4.95
	2400	4.95	4.90	-	5.00
	2500	5.00	5.00	-	-

Table 7-3
EOS-37PTH Maximum Planar Average Initial Enrichment

(2 Pages)

Fuel Assembly Class	Maximum Assembly Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)				
	Minimum Soluble Boron (ppm)	Basket Type			
		A		B	
		w/o CCs	w/ CCs	w/o CCs	w/ CCs
CE 15x15 Assembly Class	2000	4.60	4.55	4.75	4.70
	2100	4.70	4.65	4.85	4.85
	2200	4.85	4.80	5.00	4.95
	2300	5.00	4.90	-	5.00
	2400	-	5.00	-	-
	2500	-	-	-	-
CE 14x14 Assembly Class	2000	5.00	5.00	5.00	5.00
	2100	-	-	-	-
	2200	-	-	-	-
	2300	-	-	-	-
	2400	-	-	-	-
	2500	-	-	-	-
14x14 Assembly Class(1) (excludes CE 14x14)	2000	5.00	5.00	5.00	5.00
	2100	-	-	-	-
	2200	-	-	-	-
	2300	-	-	-	-
	2400	-	-	-	-
	2500	-	-	-	-

Note:
1. The loading requirements apply to FAs listed in Table 2-2 and Table 2-4 of Chapter 2.

Table 7-4
EOS-89BTH Maximum Lattice Average Initial Enrichment

Basket Type	Maximum Lattice Average Initial Enrichment (wt. % U-235) (1)
M1-A	4.10
M1-B	4.45
M2-A	4.80

Note:

1. For ABB-10-C FAs, the enrichment shall be reduced by 0.25 wt. % U-235 for Type M1-A and M2-A and 0.20 wt. % U-235 for Type M1-B.

Proprietary Information on Pages 7-38 through 7-40
Withheld Pursuant to 10 CFR 2.390

Table 7-8
Material Property Data

Material	ID	Density g/cm^3	Element	Wt. %	Atom Density (atoms/b-cm)
UO_2 (Enrichment – 1.0 to 5.0 wt. %) (1)	1	10.686	U-235	4.41	1.20668E-03
			U-238	83.74	2.26374E-02
			O	11.85	4.76881E-02
Zircaloy-4	2	6.56	Zr	98.23	4.2541E-02
			Sn	1.45	4.8254E-04
			Fe	0.21	1.4856E-04
			Cr	0.10	7.5978E-05
			Hf	0.01	2.2133E-06
Water (Pellet Clad Gap)	3	0.998	H	11.1	6.6769E-02
			O	88.9	3.3385E-02
Stainless Steel (SS304)	4	7.94	C	0.080	3.1877E-04
			Si	1.000	1.7025E-03
			P	0.045	6.9468E-05
			Cr	19.000	1.7473E-02
			Mn	2.000	1.7407E-03
			Fe	68.375	5.8545E-02
			Ni	9.500	7.7402E-03
Borated Water (2000 – 2500 ppm Boron) (2)	5	1.00	H	11.163	6.67515E-02
			O	88.587	3.33757E-02
			B-10	0.046	2.77126E-05
			B-11	0.204	1.11547E-04
$^{11}B_4C$ in CC	7	2.52	B-11	78.57	1.08305E-01
			C	21.43	2.70763E-02
Aluminum	8	2.702	Al	100.0	6.0307E-02
Water	10	0.998	H	11.1	6.6769E-02
			O	88.9	3.3385E-02
Lead	11	11.344	Pb	100.0	3.2969E-02

Note:

(1) The composition for maximum enrichment evaluated at 5.0 wt. % U-235 is provided.

(2) Applies to EOS-37PTH only. EOS-89BTH evaluated with 100% internal moderator density. The composition for the maximum soluble boron concentration at 100 % internal moderator density is provided.

Proprietary Information on Pages 7-42 through 7-44
Withheld Pursuant to 10 CFR 2.390

Table 7-13
EOS-37PTH Plate Slot Width Sensitivity Evaluation

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
Slot Width: Combination 1			
IMD 50 %	0.8900	0.0007	0.8914
IMD 60 %	0.9184	0.0007	0.9198
IMD 70 %	0.9379	0.0007	0.9393
IMD 80 %	0.9503	0.0007	0.9517
IMD 90 %	0.9567	0.0006	0.9579
IMD 100 %	0.9603	0.0008	0.9619
Slot Width: Combination 2			
IMD 50 %	0.8895	0.0007	0.8909
IMD 60 %	0.9184	0.0006	0.9196
IMD 70 %	0.9369	0.0008	0.9385
IMD 80 %	0.9510	0.0007	0.9524
IMD 90 %	0.9572	0.0006	0.9584
IMD 100 %	0.9590	0.0007	0.9604
Slot Width: Combination 3			
IMD 50 %	0.8909	0.0007	0.8923
IMD 60 %	0.9181	0.0007	0.9195
IMD 70 %	0.9370	0.0007	0.9384
IMD 80 %	0.9490	0.0007	0.9504
IMD 90 %	0.9556	0.0007	0.9570
IMD 100 %	0.9594	0.0007	0.9608
Slot Width: Combination 4			
IMD 50 %	0.8890	0.0007	0.8904
IMD 60 %	0.9182	0.0006	0.9194
IMD 70 %	0.9365	0.0007	0.9379
IMD 80 %	0.9492	0.0007	0.9506
IMD 90 %	0.9566	0.0006	0.9578
IMD 100 %	0.9591	0.0007	0.9605

Table 7-13
EOS-37PTH Plate Slot Width Sensitivity Evaluation

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
Slot Width: Combination 5			
IMD 50 %	0.8900	0.0007	0.8914
IMD 60 %	0.9182	0.0007	0.9196
IMD 70 %	0.9392	0.0007	0.9406
IMD 80 %	0.9504	0.0007	0.9518
IMD 90 %	0.9590	0.0006	0.9602
IMD 100 %	0.9603	0.0007	0.9617
Slot Width: Combination 6			
IMD 50 %	0.8887	0.0006	0.8899
IMD 60 %	0.9172	0.0006	0.9184
IMD 70 %	0.9385	0.0007	0.9399
IMD 80 %	0.9506	0.0006	0.9518
IMD 90 %	0.9562	0.0006	0.9574
IMD 100 %	0.9601	0.0006	0.9613

Table 7-14
EOS-37PTH Compartment Width Variation

Case Description	k_{keno}	σ_{keno}	k_{eff}
[		]	
IMD 50 %	0.8928	0.0007	0.8942
IMD 60 %	0.9228	0.0007	0.9242
IMD 70 %	0.9415	0.0008	0.9431
IMD 80 %	0.9541	0.0007	0.9555
IMD 90 %	0.9625	0.0007	0.9639
IMD 100 %	0.9675	0.0008	0.9691
[		]	
IMD 50 %	0.8870	0.0007	0.8884
IMD 60 %	0.9151	0.0009	0.9169
IMD 70 %	0.9339	0.0008	0.9355
IMD 80 %	0.9464	0.0007	0.9478
IMD 90 %	0.9507	0.0009	0.9525
IMD 100 %	0.9546	0.0008	0.9562

Table 7-15
EOS-37PTH Most Reactive Configuration Evaluation

2 Pages

Case Description	k_{keno}	σ_{keno}	k_{eff}
Poison Plate Thickness			
[ ]	0.9671	0.0008	0.9687
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9632	0.0007	0.9646
Basket Plate Steel Thickness			
[ ]	0.9657	0.0007	0.9671
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9630	0.0007	0.9644
[ ]	0.9660	0.0007	0.9674
Basket Aluminum Plate Thickness			
[ ]	0.9646	0.0007	0.9660
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9652	0.0007	0.9666
Inter Egg-Crate Gap Material Variation			
Borated Water	**0.9675**	**0.0008**	**0.9691**
Steel	0.9661	0.0006	0.9673
Axial Location of 1 inch Gap			
Bottom	**0.9675**	**0.0008**	**0.9691**
Middle	0.9659	0.0008	0.9675
Top	0.9661	0.0008	0.9677
Rail Approximation Radius			
[ ]	0.9660	0.0007	0.9674
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9664	0.0007	0.9678
EOS-TC Type			
TC108	0.9648	0.0007	0.9662
TC125/TC135	**0.9675**	**0.0008**	**0.9691**

Table 7-15
EOS-37PTH Most Reactive Configuration Evaluation
2 Pages

Case Description	$\mathbf{k_{keno}}$	$\boldsymbol{\sigma_{keno}}$	$\mathbf{k_{eff}}$
DSC Shell Thickness with TC125/TC135			
[ ]	0.9646	0.0007	0.9660
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9647	0.0008	0.9663
TC Dimension Variation: TC Inner Shell			
[ ]	0.9662	0.0007	0.9676
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9643	0.0007	0.9657
TC Dimension Variation: Lead Thickness			
[ ]	0.9664	0.0007	0.9678
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9665	0.0007	0.9679
TC Dimension Variation: TC Outer Shell			
[ ]	0.9655	0.0007	0.9669
[ ]	**0.9675**	**0.0008**	**0.9691**
[ ]	0.9656	0.0008	0.9672
Fuel Assembly Location			
Center in compartment	0.9619	0.0007	0.9633
Pushed to center of canister	**0.9675**	**0.0008**	**0.9691**
Pushed to outside of canister	0.9539	0.0008	0.9555
Material Between EOS-TCs			
Void	0.9656	0.0007	0.9670
Fresh Water	**0.9675**	**0.0008**	**0.9691**

Table 7-16
Most Reactive Fuel Evaluation

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
WE, BW Mark C, and Framatome 17x17			
	0.9678	0.0009	0.9696
	0.9627	0.0008	0.9643
	0.9675	**0.0008**	**0.9691**
	0.9467	0.0008	0.9483
	0.9491	0.0007	0.9505
	0.9655	0.0008	0.9671
	0.9651	0.0007	0.9665
	0.9637	0.0008	0.9653
	0.9657	0.0008	0.9673
	0.9672	0.0008	0.9688
	0.9653	0.0007	0.9667
	0.9630	0.0007	0.9644
	0.9675	**0.0008**	**0.9691**
	BW 15x15		
	0.9662	0.0007	0.9676
	0.9684	0.0007	0.9698
	0.9697	**0.0007**	**0.9711**
	0.9577	0.0007	0.9591
	WE 15x15		
	0.9457	0.0007	0.9471
	0.9561	**0.0008**	**0.9577**
	0.9506	0.0008	0.9522
	0.9566	0.0009	0.9584
	0.9579	**0.0007**	**0.9593**
	CE 15x15		
	0.9486	**0.0007**	**0.9500**
	0.9441	0.0008	0.9457
	WE 14x14		
	0.8808	0.0007	0.8822
	0.8804	0.0009	0.8822
	0.8755	0.0007	0.8769
	0.9020	**0.0008**	**0.9036**

Table 7-16
Most Reactive Fuel Evaluation

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
	0.9021	**0.0007**	**0.9035**
	0.8990	0.0010	0.9010
	CE 14x14		
	0.9076	0.0008	0.9092
	0.9076	0.0008	0.9092
	0.9111	**0.0007**	**0.9125**
	CE 16x16		
	0.9190	**0.0008**	**0.9206**
	0.9179	0.0007	0.9193
	0.9189	0.0009	0.9207
	0.9181	0.0007	0.9195

Table 7-17
WE 17x17 Class Fuel Assembly without CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.35 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8731	0.0008	0.8747
IMD 60 %	0.8998	0.0008	0.9014
IMD 70 %	0.9163	0.0007	0.9177
IMD 80 %	0.9267	0.0007	0.9281
IMD 90 %	0.9331	0.0007	0.9345
IMD 100 %	0.9331	0.0007	0.9345
4.50 wt. % U-235, 2100 ppm, W/O CCs, Type A			
IMD 50 %	0.8775	0.0007	0.8789
IMD 60 %	0.9028	0.0007	0.9042
IMD 70 %	0.9208	0.0008	0.9224
IMD 80 %	0.9308	0.0007	0.9322
IMD 90 %	0.9354	0.0007	0.9368
IMD 100 %	0.9352	0.0007	0.9366
4.60 wt. % U-235, 2200 ppm, W/O CCs, Type A			
IMD 50 %	0.8801	0.0008	0.8817
IMD 60 %	0.9041	0.0007	0.9055
IMD 70 %	0.9202	0.0007	0.9216
IMD 80 %	0.9303	0.0008	0.9319
IMD 90 %	0.9337	0.0010	0.9357
IMD 100 %	0.9333	0.0008	0.9349
4.70 wt. % U-235, 2300 ppm, W/O CCs, Type A			
IMD 50 %	0.8823	0.0006	0.8835
IMD 60 %	0.9055	0.0007	0.9069
IMD 70 %	0.9210	0.0007	0.9224
IMD 80 %	0.9312	0.0006	0.9324
IMD 90 %	0.9344	0.0007	0.9358
IMD 100 %	0.9334	0.0007	0.9348

Table 7-17
WE 17x17 Class Fuel Assembly without CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.85 wt. % U-235, 2400 ppm, W/O CCs, Type A			
IMD 50 %	0.8853	0.0008	0.8869
IMD 60 %	0.9081	0.0007	0.9095
IMD 70 %	0.9241	0.0007	0.9255
IMD 80 %	0.9323	0.0007	0.9337
IMD 90 %	0.9351	0.0007	0.9365
IMD 100 %	0.9339	0.0008	0.9355
4.95 wt. % U-235, 2500 ppm, W/O CCs, Type A			
IMD 50 %	0.8879	0.0008	0.8895
IMD 60 %	0.9116	0.0009	0.9134
IMD 70 %	0.9248	0.0008	0.9264
IMD 80 %	0.9318	0.0007	0.9332
IMD 90 %	0.9343	0.0008	0.9359
IMD 100 %	0.9332	0.0007	0.9346

Table 7-18
WE 17x17 Class Fuel Assembly with CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.35 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8581	0.0007	0.8595
IMD 60 %	0.8876	0.0008	0.8892
IMD 70 %	0.9077	0.0007	0.9091
IMD 80 %	0.9212	0.0007	0.9226
IMD 90 %	0.9305	0.0008	0.9321
IMD 100 %	0.9356	0.0007	0.9370
4.45 wt. % U-235, 2100 ppm, W/ CCs, Type A			
IMD 50 %	0.8613	0.0008	0.8629
IMD 60 %	0.8898	0.0007	0.8912
IMD 70 %	0.9079	0.0006	0.9091
IMD 80 %	0.9218	0.0007	0.9232
IMD 90 %	0.9307	0.0008	0.9323
IMD 100 %	0.9341	0.0007	0.9355
4.55 wt. % U-235, 2200 ppm, W/ CCs, Type A			
IMD 50 %	0.8637	0.0007	0.8651
IMD 60 %	0.8902	0.0007	0.8916
IMD 70 %	0.9092	0.0007	0.9106
IMD 80 %	0.9233	0.0007	0.9247
IMD 90 %	0.9304	0.0008	0.9320
IMD 100 %	0.9339	0.0007	0.9353
4.65 wt. % U-235, 2300 ppm, W/ CCs, Type A			
IMD 50 %	0.8660	0.0008	0.8676
IMD 60 %	0.8936	0.0009	0.8954
IMD 70 %	0.9117	0.0007	0.9131
IMD 80 %	0.9234	0.0007	0.9248
IMD 90 %	0.9299	0.0006	0.9311
IMD 100 %	0.9337	0.0007	0.9351

Table 7-18
WE 17x17 Class Fuel Assembly with CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.8 wt. % U-235, 2400 ppm, W/ CCs, Type A			
IMD 50 %	0.8707	0.0007	0.8721
IMD 60 %	0.8971	0.0008	0.8987
IMD 70 %	0.9162	0.0007	0.9176
IMD 80 %	0.9265	0.0009	0.9283
IMD 90 %	0.9327	0.0008	0.9343
IMD 100 %	0.9359	0.0006	0.9371
4.9 wt. % U-235, 2500 ppm, W/ CCs, Type A			
IMD 50 %	0.8728	0.0006	0.8740
IMD 60 %	0.8968	0.0008	0.8984
IMD 70 %	0.9150	0.0007	0.9164
IMD 80 %	0.9245	0.0007	0.9259
IMD 90 %	0.9328	0.0007	0.9342
IMD 100 %	0.9356	0.0007	0.9370

Table 7-19
WE 17x17 Class Fuel Assembly without CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.5 wt. % U-235, 2000 ppm, W/O CCs, Type B			
IMD 50 %	0.8667	0.0007	0.8681
IMD 60 %	0.8952	0.0007	0.8966
IMD 70 %	0.9147	0.0008	0.9163
IMD 80 %	0.9266	0.0008	0.9282
IMD 90 %	0.9330	0.0007	0.9344
IMD 100 %	0.9347	0.0008	0.9363
4.65 wt. % U-235, 2100 ppm, W/O CCs, Type B			
IMD 50 %	0.8708	0.0008	0.8724
IMD 60 %	0.8976	0.0008	0.8992
IMD 70 %	0.9164	0.0007	0.9178
IMD 80 %	0.9295	0.0007	0.9309
IMD 90 %	0.9349	0.0007	0.9363
IMD 100 %	0.9366	0.0007	0.9380
4.75 wt. % U-235, 2200 ppm, W/O CCs, Type B			
IMD 50 %	0.8731	0.0007	0.8745
IMD 60 %	0.8987	0.0007	0.9001
IMD 70 %	0.9174	0.0006	0.9186
IMD 80 %	0.9268	0.0007	0.9282
IMD 90 %	0.9333	0.0007	0.9347
IMD 100 %	0.9349	0.0007	0.9363
4.85 wt. % U-235, 2300 ppm, W/O CCs, Type B			
IMD 50 %	0.8764	0.0007	0.8778
IMD 60 %	0.8997	0.0008	0.9013
IMD 70 %	0.9176	0.0007	0.9190
IMD 80 %	0.9288	0.0007	0.9302
IMD 90 %	0.9322	0.0007	0.9336
IMD 100 %	0.9342	0.0006	0.9354

Table 7-19
WE 17x17 Class Fuel Assembly without CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.0 wt. % U-235, 2400 ppm, W/O CCs, Type B			
IMD 50 %	0.8791	0.0008	0.8807
IMD 60 %	0.9039	0.0007	0.9053
IMD 70 %	0.9207	0.0007	0.9221
IMD 80 %	0.9303	0.0007	0.9317
IMD 90 %	0.9352	0.0007	0.9366
IMD 100 %	0.9341	0.0007	0.9355

Table 7-20
WE 17x17 Class Fuel Assembly with CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.45 wt. % U-235, 2000 ppm, W/ CCs, Type B			
IMD 50 %	0.8477	0.0008	0.8493
IMD 60 %	0.8785	0.0007	0.8799
IMD 70 %	0.9024	0.0008	0.9040
IMD 80 %	0.9162	0.0007	0.9176
IMD 90 %	0.9270	0.0008	0.9286
IMD 100 %	0.9323	0.0007	0.9337
4.60 wt. % U-235, 2100 ppm, W/ CCs, Type B			
IMD 50 %	0.8531	0.0007	0.8545
IMD 60 %	0.8832	0.0006	0.8844
IMD 70 %	0.9049	0.0008	0.9065
IMD 80 %	0.9204	0.0008	0.9220
IMD 90 %	0.9298	0.0007	0.9312
IMD 100 %	0.9349	0.0008	0.9365
4.70 wt. % U-235, 2200 ppm, W/ CCs, Type B			
IMD 50 %	0.8560	0.0009	0.8578
IMD 60 %	0.8857	0.0007	0.8871
IMD 70 %	0.9045	0.0008	0.9061
IMD 80 %	0.9199	0.0008	0.9215
IMD 90 %	0.9301	0.0010	0.9321
IMD 100 %	0.9325	0.0007	0.9339
4.85 wt. % U-235, 2300 ppm, W/ CCs, Type B			
IMD 50 %	0.8608	0.0008	0.8624
IMD 60 %	0.8883	0.0008	0.8899
IMD 70 %	0.9099	0.0007	0.9113
IMD 80 %	0.9238	0.0008	0.9254
IMD 90 %	0.9310	0.0007	0.9324
IMD 100 %	0.9361	0.0007	0.9375

Table 7-20
WE 17x17 Class Fuel Assembly with CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.95 wt. % U-235, 2400 ppm, W/ CCs, Type B			
IMD 50 %	0.8639	0.0007	0.8653
IMD 60 %	0.8896	0.0008	0.8912
IMD 70 %	0.9097	0.0007	0.9111
IMD 80 %	0.9232	0.0007	0.9246
IMD 90 %	0.9318	0.0007	0.9332
IMD 100 %	0.9359	0.0008	0.9375
5.0 wt. % U-235, 2500 ppm, W/ CCs, Type B			
IMD 50 %	0.8624	0.0008	0.8640
IMD 60 %	0.8906	0.0007	0.8920
IMD 70 %	0.9087	0.0007	0.9101
IMD 80 %	0.9207	0.0007	0.9221
IMD 90 %	0.9279	0.0007	0.9293
IMD 100 %	0.9311	0.0008	0.9327

Table 7-21
B&W 15x15 Class Assembly without CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.25 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8742	0.0007	0.8756
IMD 60 %	0.8994	0.0007	0.9008
IMD 70 %	0.9151	0.0008	0.9167
IMD 80 %	0.9274	0.0006	0.9286
IMD 90 %	0.9335	0.0007	0.9349
IMD 100 %	0.9326	0.0007	0.9340
4.40 wt. % U-235, 2100 ppm, W/O CCs, Type A			
IMD 50 %	0.8786	0.0008	0.8802
IMD 60 %	0.9023	0.0007	0.9037
IMD 70 %	0.9218	0.0007	0.9232
IMD 80 %	0.9313	0.0007	0.9327
IMD 90 %	0.9353	0.0008	0.9369
IMD 100 %	0.9345	0.0007	0.9359
4.50 wt. % U-235, 2200 ppm, W/O CCs, Type A			
IMD 50 %	0.8797	0.0007	0.8811
IMD 60 %	0.9047	0.0007	0.9061
IMD 70 %	0.9212	0.0007	0.9226
IMD 80 %	0.9305	0.0007	0.9319
IMD 90 %	0.9342	0.0008	0.9358
IMD 100 %	0.9335	0.0007	0.9349
4.60 wt. % U-235, 2300 ppm, W/O CCs, Type A			
IMD 50 %	0.8829	0.0009	0.8847
IMD 60 %	0.9057	0.0008	0.9073
IMD 70 %	0.9216	0.0009	0.9234
IMD 80 %	0.9310	0.0007	0.9324
IMD 90 %	0.9340	0.0007	0.9354
IMD 100 %	0.9319	0.0007	0.9333

Table 7-21
B&W 15x15 Class Assembly without CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.75 wt. % U-235, 2400 ppm, W/O CCs, Type A			
IMD 50 %	0.8861	0.0007	0.8875
IMD 60 %	0.9114	0.0008	0.9130
IMD 70 %	0.9252	0.0007	0.9266
IMD 80 %	0.9322	0.0007	0.9336
IMD 90 %	0.9358	0.0007	0.9372
IMD 100 %	0.9337	0.0007	0.9351
4.85 wt. % U-235, 2500 ppm, W/O CCs, Type A			
IMD 50 %	0.8897	0.0007	0.8911
IMD 60 %	0.9116	0.0007	0.9130
IMD 70 %	0.9256	0.0007	0.9270
IMD 80 %	0.9353	0.0007	0.9367
IMD 90 %	0.9357	0.0007	0.9371
IMD 100 %	0.9325	0.0008	0.9341

Table 7-22
B&W 15x15 Class Assembly with CCs Final Results, Type A Basket
(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.20 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8570	0.0009	0.8588
IMD 60 %	0.8863	0.0008	0.8879
IMD 70 %	0.9063	0.0007	0.9077
IMD 80 %	0.9208	0.0009	0.9226
IMD 90 %	0.9301	0.0008	0.9317
IMD 100 %	0.9331	0.0007	0.9345
4.30 wt. % U-235, 2100 ppm, W/ CCs, Type A			
IMD 50 %	0.8597	0.0007	0.8611
IMD 60 %	0.8876	0.0007	0.8890
IMD 70 %	0.9078	0.0008	0.9094
IMD 80 %	0.9224	0.0007	0.9238
IMD 90 %	0.9287	0.0007	0.9301
IMD 100 %	0.9331	0.0007	0.9345
4.45 wt. % U-235, 2200 ppm, W/ CCs, Type A			
IMD 50 %	0.8638	0.0009	0.8656
IMD 60 %	0.8928	0.0008	0.8944
IMD 70 %	0.9127	0.0007	0.9141
IMD 80 %	0.9252	0.0009	0.9270
IMD 90 %	0.9318	0.0007	0.9332
IMD 100 %	0.9345	0.0008	0.9361
4.55 wt. % U-235, 2300 ppm, W/ CCs, Type A			
IMD 50 %	0.8684	0.0007	0.8698
IMD 60 %	0.8946	0.0008	0.8962
IMD 70 %	0.9129	0.0008	0.9145
IMD 80 %	0.9240	0.0007	0.9254
IMD 90 %	0.9310	0.0007	0.9324
IMD 100 %	0.9339	0.0006	0.9351
4.65 wt. % U-235, 2400 ppm, W/ CCs, Type A			
IMD 50 %	0.8692	0.0008	0.8708
IMD 60 %	0.8972	0.0007	0.8986
IMD 70 %	0.9131	0.0008	0.9147
IMD 80 %	0.9249	0.0007	0.9263

Table 7-22
B&W 15x15 Class Assembly with CCs Final Results, Type A Basket
(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
IMD 90 %	0.9328	0.0007	0.9342
IMD 100 %	0.9345	0.0007	0.9359
4.75 wt. % U-235, 2500 ppm, W/ CCs, Type A			
IMD 50 %	0.8726	0.0007	0.8740
IMD 60 %	0.8967	0.0008	0.8983
IMD 70 %	0.9171	0.0007	0.9185
IMD 80 %	0.9259	0.0007	0.9273
IMD 90 %	0.9322	0.0007	0.9336
IMD 100 %	0.9348	0.0007	0.9362

Table 7-23
B&W 15x15 Class Assembly without CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.40 wt. % U-235, 2000 ppm, W/O CCs, Type B			
IMD 50 %	0.8689	0.0008	0.8705
IMD 60 %	0.8948	0.0007	0.8962
IMD 70 %	0.9136	0.0007	0.9150
IMD 80 %	0.9257	0.0008	0.9273
IMD 90 %	0.9339	0.0007	0.9353
IMD 100 %	0.9344	0.0007	0.9358
4.55 wt. % U-235, 2100 ppm, W/O CCs, Type B			
IMD 50 %	0.8716	0.0007	0.8730
IMD 60 %	0.8986	0.0007	0.9000
IMD 70 %	0.9184	0.0007	0.9198
IMD 80 %	0.9291	0.0009	0.9309
IMD 90 %	0.9336	0.0008	0.9352
IMD 100 %	0.9352	0.0007	0.9366
4.65 wt. % U-235, 2200 ppm, W/O CCs, Type B			
IMD 50 %	0.8740	0.0008	0.8756
IMD 60 %	0.9014	0.0008	0.9030
IMD 70 %	0.9176	0.0007	0.9190
IMD 80 %	0.9288	0.0007	0.9302
IMD 90 %	0.9328	0.0005	0.9338
IMD 100 %	0.9353	0.0007	0.9367
4.80 wt. % U-235, 2300 ppm, W/O CCs, Type B			
IMD 50 %	0.8777	0.0008	0.8793
IMD 60 %	0.9048	0.0009	0.9066
IMD 70 %	0.9202	0.0008	0.9218
IMD 80 %	0.9315	0.0008	0.9331
IMD 90 %	0.9352	0.0008	0.9368
IMD 100 %	0.9358	0.0007	0.9372

Table 7-23
B&W 15x15 Class Assembly without CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.90 wt. % U-235, 2400 ppm, W/O CCs, Type B			
IMD 50 %	0.8786	0.0007	0.8800
IMD 60 %	0.9058	0.0007	0.9072
IMD 70 %	0.9225	0.0007	0.9239
IMD 80 %	0.9302	0.0008	0.9318
IMD 90 %	0.9352	0.0008	0.9368
IMD 100 %	0.9347	0.0007	0.9361
5.00 wt. % U-235, 2500 ppm, W/O CCs, Type B			
IMD 50 %	0.8836	0.0007	0.8850
IMD 60 %	0.9063	0.0007	0.9077
IMD 70 %	0.9214	0.0007	0.9228
IMD 80 %	0.9300	0.0007	0.9314
IMD 90 %	0.9344	0.0007	0.9358
IMD 100 %	0.9328	0.0008	0.9344

Table 7-24
B&W 15x15 Class Assembly with CCs Final Results, Type B Basket
(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.35 wt. % U-235, 2000 ppm, W/ CCs, Type B			
IMD 50 %	0.8508	0.0008	0.8524
IMD 60 %	0.8802	0.0007	0.8816
IMD 70 %	0.9022	0.0008	0.9038
IMD 80 %	0.9182	0.0008	0.9198
IMD 90 %	0.9276	0.0008	0.9292
IMD 100 %	0.9364	0.0007	0.9378
4.45 wt. % U-235, 2100 ppm, W/ CCs, Type B			
IMD 50 %	0.8520	0.0008	0.8536
IMD 60 %	0.8828	0.0007	0.8842
IMD 70 %	0.9037	0.0008	0.9053
IMD 80 %	0.9189	0.0008	0.9205
IMD 90 %	0.9279	0.0007	0.9293
IMD 100 %	0.9337	0.0008	0.9353
4.60 wt. % U-235, 2200 ppm, W/ CCs, Type B			
IMD 50 %	0.8585	0.0006	0.8597
IMD 60 %	0.8867	0.0008	0.8883
IMD 70 %	0.9091	0.0007	0.9105
IMD 80 %	0.9231	0.0008	0.9247
IMD 90 %	0.9317	0.0007	0.9331
IMD 100 %	0.9365	0.0008	0.9381
4.70 wt. % U-235, 2300 ppm, W/ CCs, Type B			
IMD 50 %	0.8610	0.0008	0.8626
IMD 60 %	0.8883	0.0007	0.8897
IMD 70 %	0.9097	0.0007	0.9111
IMD 80 %	0.9235	0.0007	0.9249
IMD 90 %	0.9313	0.0006	0.9325
IMD 100 %	0.9356	0.0007	0.9370
4.85 wt. % U-235, 2400 ppm, W/ CCs, Type B			
IMD 50 %	0.8650	0.0007	0.8664
IMD 60 %	0.8933	0.0006	0.8945
IMD 70 %	0.9123	0.0008	0.9139
IMD 80 %	0.9257	0.0007	0.9271

Table 7-24
B&W 15x15 Class Assembly with CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
IMD 90 %	0.9345	0.0007	0.9359
IMD 100 %	0.9370	0.0006	0.9382
4.90 wt. % U-235, 2500 ppm, W/ CCs, Type B			
IMD 50 %	0.8654	0.0007	0.8668
IMD 60 %	0.8906	0.0007	0.8920
IMD 70 %	0.9110	0.0007	0.9124
IMD 80 %	0.9220	0.0008	0.9236
IMD 90 %	0.9307	0.0007	0.9321
IMD 100 %	0.9331	0.0008	0.9347

Table 7-25
WE 15x15 Class Fuel Assembly without CCs Final Results, Type A Basket

(2 pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.45 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8754	0.0007	0.8768
IMD 60 %	0.9008	0.0007	0.9022
IMD 70 %	0.9191	0.0008	0.9207
IMD 80 %	0.9305	0.0007	0.9319
IMD 90 %	0.9331	0.0007	0.9345
IMD 100 %	0.9341	0.0006	0.9353
4.60 wt. % U-235, 2100 ppm, W/O CCs, Type A			
IMD 50 %	0.8810	0.0007	0.8824
IMD 60 %	0.9061	0.0008	0.9077
IMD 70 %	0.9228	0.0008	0.9244
IMD 80 %	0.9297	0.0007	0.9311
IMD 90 %	0.9344	0.0007	0.9358
IMD 100 %	0.9346	0.0007	0.9360
4.70 wt. % U-235, 2200 ppm, W/O CCs, Type A			
IMD 50 %	0.8820	0.0007	0.8834
IMD 60 %	0.9062	0.0006	0.9074
IMD 70 %	0.9223	0.0007	0.9237
IMD 80 %	0.9299	0.0008	0.9315
IMD 90 %	0.9341	0.0007	0.9355
IMD 100 %	0.9327	0.0006	0.9339
4.85 wt. % U-235, 2300 ppm, W/O CCs, Type A			
IMD 50 %	0.8860	0.0007	0.8874
IMD 60 %	0.9098	0.0007	0.9112
IMD 70 %	0.9265	0.0007	0.9279
IMD 80 %	0.9349	0.0007	0.9363
IMD 90 %	0.9368	0.0007	0.9382
IMD 100 %	0.9354	0.0007	0.9368
4.95 wt. % U-235, 2400 ppm, W/O CCs, Type A			
IMD 50 %	0.8872	0.0008	0.8888
IMD 60 %	0.9112	0.0007	0.9126
IMD 70 %	0.9251	0.0007	0.9265

Table 7-25
WE 15x15 Class Fuel Assembly without CCs Final Results, Type A Basket

(2 pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
IMD 80 %	0.9333	0.0007	0.9347
IMD 90 %	0.9346	0.0007	0.9360
IMD 100 %	0.9319	0.0007	0.9333
5.00 wt. % U-235, 2500 ppm, W/O CCs, Type A			
IMD 50 %	0.8876	0.0007	0.8890
IMD 60 %	0.9095	0.0008	0.9111
IMD 70 %	0.9231	0.0007	0.9245
IMD 80 %	0.9292	0.0006	0.9304
IMD 90 %	0.9309	0.0007	0.9323
IMD 100 %	0.9268	0.0007	0.9282

Table 7-26
WE 15x15 Class Fuel Assembly with CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.40 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8546	0.0009	0.8564
IMD 60 %	0.8855	0.0007	0.8869
IMD 70 %	0.9061	0.0008	0.9077
IMD 80 %	0.9200	0.0008	0.9216
IMD 90 %	0.9299	0.0008	0.9315
IMD 100 %	0.9346	0.0007	0.9360
4.55 wt. % U-235, 2100 ppm, W/ CCs, Type A			
IMD 50 %	0.8608	0.0007	0.8622
IMD 60 %	0.8892	0.0008	0.8908
IMD 70 %	0.9104	0.0007	0.9118
IMD 80 %	0.9228	0.0007	0.9242
IMD 90 %	0.9324	0.0007	0.9338
IMD 100 %	0.9368	0.0007	0.9382
4.65 wt. % U-235, 2200 ppm, W/ CCs, Type A			
IMD 50 %	0.8632	0.0007	0.8646
IMD 60 %	0.8906	0.0008	0.8922
IMD 70 %	0.9106	0.0007	0.9120
IMD 80 %	0.9239	0.0008	0.9255
IMD 90 %	0.9337	0.0007	0.9351
IMD 100 %	0.9364	0.0008	0.9380
4.75 wt. % U-235, 2300 ppm, W/ CCs, Type A			
IMD 50 %	0.8649	0.0007	0.8663
IMD 60 %	0.8916	0.0007	0.8930
IMD 70 %	0.9122	0.0007	0.9136
IMD 80 %	0.9248	0.0008	0.9264
IMD 90 %	0.9309	0.0007	0.9323
IMD 100 %	0.9348	0.0008	0.9364
4.90 wt. % U-235, 2400 ppm, W/ CCs, Type A			
IMD 50 %	0.8702	0.0006	0.8714
IMD 60 %	0.8978	0.0008	0.8994
IMD 70 %	0.9145	0.0008	0.9161

Table 7-26
WE 15x15 Class Fuel Assembly with CCs Final Results, Type A Basket

(2 Pages)

IMD 80 %	0.9263	0.0008	0.9279
IMD 90 %	0.9338	0.0008	0.9354
IMD 100 %	0.9361	0.0007	0.9375
5.00 wt. % U-235, 2500 ppm, W/ CCs, Type A			
IMD 50 %	0.8733	0.0007	0.8747
IMD 60 %	0.8986	0.0007	0.9000
IMD 70 %	0.9169	0.0007	0.9183
IMD 80 %	0.9282	0.0009	0.9300
IMD 90 %	0.9347	0.0008	0.9363
IMD 100 %	0.9362	0.0007	0.9376

Table 7-27
WE 15x15 Class Fuel Assembly without CCs Final Results, Type B Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.55 wt. % U-235, 2000 ppm, W/O CCs, Type B			
IMD 50 %	0.8658	0.0008	0.8674
IMD 60 %	0.8932	0.0007	0.8946
IMD 70 %	0.9132	0.0008	0.9148
IMD 80 %	0.9249	0.0007	0.9263
IMD 90 %	0.9302	0.0007	0.9316
IMD 100 %	0.9307	0.0007	0.9321
4.65 wt. % U-235, 2100 ppm, W/O CCs, Type B			
IMD 50 %	0.8677	0.0007	0.8691
IMD 60 %	0.8954	0.0007	0.8968
IMD 70 %	0.9144	0.0009	0.9162
IMD 80 %	0.9240	0.0008	0.9256
IMD 90 %	0.9308	0.0008	0.9324
IMD 100 %	0.9283	0.0008	0.9299
4.80 wt. % U-235, 2200 ppm, W/O CCs, Type B			
IMD 50 %	0.8740	0.0007	0.8754
IMD 60 %	0.9002	0.0009	0.9020
IMD 70 %	0.9167	0.0008	0.9183
IMD 80 %	0.9270	0.0007	0.9284
IMD 90 %	0.9304	0.0007	0.9318
IMD 100 %	0.9307	0.0007	0.9321
5.00 wt. % U-235, 2300 ppm, W/O CCs, Type B			
IMD 50 %	0.8791	0.0008	0.8807
IMD 60 %	0.9062	0.0008	0.9078
IMD 70 %	0.9215	0.0007	0.9229
IMD 80 %	0.9321	0.0008	0.9337
IMD 90 %	0.9350	0.0007	0.9364
IMD 100 %	0.9355	0.0008	0.9371

Table 7-28
WE 15x15 Class Fuel Assembly with CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.55 wt. % U-235, 2000 ppm, W/ CCs, Type B			
IMD 50 %	0.8478	0.0007	0.8492
IMD 60 %	0.8794	0.0007	0.8808
IMD 70 %	0.9020	0.0007	0.9034
IMD 80 %	0.9174	0.0008	0.9190
IMD 90 %	0.9299	0.0007	0.9313
IMD 100 %	0.9353	0.0007	0.9367
4.65 wt. % U-235, 2100 ppm, W/ CCs, Type B			
IMD 50 %	0.8499	0.0008	0.8515
IMD 60 %	0.8804	0.0007	0.8818
IMD 70 %	0.9045	0.0007	0.9059
IMD 80 %	0.9196	0.0008	0.9212
IMD 90 %	0.9284	0.0007	0.9298
IMD 100 %	0.9321	0.0007	0.9335
4.80 wt. % U-235, 2200 ppm, W/ CCs, Type B			
IMD 50 %	0.8546	0.0007	0.8560
IMD 60 %	0.8845	0.0008	0.8861
IMD 70 %	0.9055	0.0007	0.9069
IMD 80 %	0.9221	0.0007	0.9235
IMD 90 %	0.9298	0.0008	0.9314
IMD 100 %	0.9360	0.0008	0.9376
4.95 wt. % U-235, 2300 ppm, W/ CCs, Type B			
IMD 50 %	0.8598	0.0006	0.8610
IMD 60 %	0.8891	0.0007	0.8905
IMD 70 %	0.9102	0.0008	0.9118
IMD 80 %	0.9228	0.0007	0.9242
IMD 90 %	0.9326	0.0007	0.9340
IMD 100 %	0.9368	0.0006	0.9380

Table 7-28
WE 15x15 Class Fuel Assembly with CCs Final Results, Type B Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2400 ppm, W/ CCs, Type B			
IMD 50 %	0.8592	0.0008	0.8608
IMD 60 %	0.8878	0.0008	0.8894
IMD 70 %	0.9087	0.0009	0.9105
IMD 80 %	0.9216	0.0007	0.9230
IMD 90 %	0.9293	0.0007	0.9307
IMD 100 %	0.9331	0.0007	0.9345

Table 7-29
CE 15x15 Class Assembly without CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.60 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8634	0.0007	0.8648
IMD 60 %	0.8907	0.0008	0.8923
IMD 70 %	0.9099	0.0007	0.9113
IMD 80 %	0.9239	0.0007	0.9253
IMD 90 %	0.9304	0.0008	0.9320
IMD 100 %	0.9356	0.0007	0.9370
4.70 wt. % U-235, 2100 ppm, W/O CCs, Type A			
IMD 50 %	0.8661	0.0008	0.8677
IMD 60 %	0.8929	0.0007	0.8943
IMD 70 %	0.9120	0.0008	0.9136
IMD 80 %	0.9226	0.0007	0.9240
IMD 90 %	0.9298	0.0008	0.9314
IMD 100 %	0.9343	0.0008	0.9359
4.85 wt. % U-235, 2200 ppm, W/O CCs, Type A			
IMD 50 %	0.8691	0.0008	0.8707
IMD 60 %	0.8966	0.0007	0.8980
IMD 70 %	0.9144	0.0007	0.9158
IMD 80 %	0.9250	0.0008	0.9266
IMD 90 %	0.9332	0.0008	0.9348
IMD 100 %	0.9358	0.0007	0.9372
5.00 wt. % U-235, 2300 ppm, W/O CCs, Type A			
IMD 50 %	0.8751	0.0007	0.8765
IMD 60 %	0.8998	0.0005	0.9008
IMD 70 %	0.9166	0.0010	0.9186
IMD 80 %	0.9282	0.0007	0.9296
IMD 90 %	0.9338	0.0008	0.9354
IMD 100 %	0.9365	0.0008	0.9381

Table 7-30
CE 15x15 Class Assembly with CCs Final Results, Type A Basket
(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.55 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8624	0.0007	0.8638
IMD 60 %	0.8886	0.0008	0.8902
IMD 70 %	0.9090	0.0008	0.9106
IMD 80 %	0.9224	0.0007	0.9238
IMD 90 %	0.9310	0.0008	0.9326
IMD 100 %	0.9367	0.0007	0.9381
4.65 wt. % U-235, 2100 ppm, W/ CCs, Type A			
IMD 50 %	0.8644	0.0007	0.8658
IMD 60 %	0.8903	0.0008	0.8919
IMD 70 %	0.9101	0.0006	0.9113
IMD 80 %	0.9221	0.0008	0.9237
IMD 90 %	0.9309	0.0008	0.9325
IMD 100 %	0.9334	0.0008	0.9350
4.80 wt. % U-235, 2200 ppm, W/ CCs, Type A			
IMD 50 %	0.8683	0.0007	0.8697
IMD 60 %	0.8945	0.0007	0.8959
IMD 70 %	0.9129	0.0007	0.9143
IMD 80 %	0.9264	0.0008	0.9280
IMD 90 %	0.9337	0.0007	0.9351
IMD 100 %	0.9363	0.0007	0.9377
4.90 wt. % U-235, 2300 ppm, W/ CCs, Type A			
IMD 50 %	0.8705	0.0008	0.8721
IMD 60 %	0.8967	0.0007	0.8981
IMD 70 %	0.9132	0.0007	0.9146
IMD 80 %	0.9256	0.0007	0.9270
IMD 90 %	0.9310	0.0007	0.9324
IMD 100 %	0.9346	0.0007	0.9360

Table 7-30
CE 15x15 Class Assembly with CCs Final Results, Type A Basket

(2 Pages)

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2400 ppm, W/ CCs, Type A			
IMD 50 %	0.8707	0.0007	0.8721
IMD 60 %	0.8965	0.0007	0.8979
IMD 70 %	0.9145	0.0008	0.9161
IMD 80 %	0.9247	0.0007	0.9261
IMD 90 %	0.9313	0.0007	0.9327
IMD 100 %	0.9328	0.0007	0.9342

Table 7-31
CE 15x15 Class Assembly without CCs Final Results, Type B Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.75 wt. % U-235, 2000 ppm, W/O CCs, Type B			
IMD 50 %	0.8577	0.0008	0.8593
IMD 60 %	0.8852	0.0008	0.8868
IMD 70 %	0.9060	0.0007	0.9074
IMD 80 %	0.9213	0.0007	0.9227
IMD 90 %	0.9311	0.0007	0.9325
IMD 100 %	0.9371	0.0007	0.9385
4.85 wt. % U-235, 2100 ppm, W/O CCs, Type B			
IMD 50 %	0.8590	0.0008	0.8606
IMD 60 %	0.8860	0.0008	0.8876
IMD 70 %	0.9071	0.0007	0.9085
IMD 80 %	0.9206	0.0007	0.9220
IMD 90 %	0.9300	0.0009	0.9318
IMD 100 %	0.9345	0.0008	0.9361
5.00 wt. % U-235, 2200 ppm, W/O CCs, Type B			
IMD 50 %	0.8629	0.0008	0.8645
IMD 60 %	0.8899	0.0008	0.8915
IMD 70 %	0.9098	0.0008	0.9114
IMD 80 %	0.9236	0.0008	0.9252
IMD 90 %	0.9292	0.0007	0.9306
IMD 100 %	0.9350	0.0008	0.9366

Table 7-32
CE 15x15 Class Assembly with CCs Final Results, Type B Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
4.70 wt. % U-235, 2000 ppm, W/ CCs, Type B			
IMD 50 %	0.8547	0.0007	0.8561
IMD 60 %	0.8834	0.0008	0.8850
IMD 70 %	0.9056	0.0008	0.9072
IMD 80 %	0.9207	0.0007	0.9221
IMD 90 %	0.9296	0.0007	0.9310
IMD 100 %	0.9365	0.0007	0.9379
4.85 wt. % U-235, 2100 ppm, W/ CCs, Type B			
IMD 50 %	0.8599	0.0007	0.8613
IMD 60 %	0.8868	0.0007	0.8882
IMD 70 %	0.9089	0.0007	0.9103
IMD 80 %	0.9218	0.0007	0.9232
IMD 90 %	0.9325	0.0008	0.9341
IMD 100 %	0.9359	0.0007	0.9373
4.95 wt. % U-235, 2200 ppm, W/ CCs, Type B			
IMD 50 %	0.8596	0.0007	0.8610
IMD 60 %	0.8894	0.0008	0.8910
IMD 70 %	0.9096	0.0008	0.9112
IMD 80 %	0.9216	0.0008	0.9232
IMD 90 %	0.9317	0.0007	0.9331
IMD 100 %	0.9350	0.0007	0.9364
5.00 wt. % U-235, 2300 ppm, W/ CCs, Type B			
IMD 50 %	0.8596	0.0007	0.8610
IMD 60 %	0.8876	0.0009	0.8894
IMD 70 %	0.9064	0.0007	0.9078
IMD 80 %	0.9193	0.0008	0.9209
IMD 90 %	0.9272	0.0008	0.9288
IMD 100 %	0.9324	0.0009	0.9342

Table 7-33
CE 14x14 Class Assembly without CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8763	0.0008	0.8779
IMD 60 %	0.8996	0.0008	0.9012
IMD 70 %	0.9144	0.0008	0.9160
IMD 80 %	0.9191	0.0007	0.9205
IMD 90 %	0.9194	0.0007	0.9208
IMD 100 %	0.9199	0.0007	0.9213

Table 7-34
CE 14x14 Class Assembly with CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8637	0.0007	0.8651
IMD 60 %	0.8913	0.0009	0.8931
IMD 70 %	0.9110	0.0008	0.9126
IMD 80 %	0.9226	0.0007	0.9240
IMD 90 %	0.9329	0.0008	0.9345
IMD 100 %	0.9354	0.0007	0.9368

Table 7-35
WE 14x14 Class Fuel Assembly without CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8662	0.0009	0.8680
IMD 60 %	0.8876	0.0008	0.8892
IMD 70 %	0.8994	0.0010	0.9014
IMD 80 %	0.9090	0.0008	0.9106
IMD 90 %	0.9097	0.0008	0.9113
IMD 100 %	0.9086	0.0008	0.9102

Table 7-36
WE 14x14 Class Fuel Assembly with CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8505	0.0008	0.8521
IMD 60 %	0.8735	0.0008	0.8751
IMD 70 %	0.8899	0.0008	0.8915
IMD 80 %	0.9010	0.0007	0.9024
IMD 90 %	0.9057	0.0007	0.9071
IMD 100 %	0.9098	0.0008	0.9114

Table 7-37
CE 16x16 Class Assembly without CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2000 ppm, W/O CCs, Type A			
IMD 50 %	0.8694	0.0008	0.8710
IMD 60 %	0.8937	0.0007	0.8951
IMD 70 %	0.9116	0.0008	0.9132
IMD 80 %	0.9224	0.0008	0.9240
IMD 90 %	0.9275	0.0007	0.9289
IMD 100 %	0.9299	0.0008	0.9315

Table 7-38
CE 16x16 Class Assembly with CCs Final Results, Type A Basket

Case Description	k_{keno}	σ_{keno}	k_{eff}
5.00 wt. % U-235, 2000 ppm, W/ CCs, Type A			
IMD 50 %	0.8577	0.0008	0.8593
IMD 60 %	0.8862	0.0008	0.8878
IMD 70 %	0.9059	0.0007	0.9073
IMD 80 %	0.9190	0.0007	0.9204
IMD 90 %	0.9293	0.0009	0.9311
IMD 100 %	0.9322	0.0008	0.9338

Proprietary Information on This Page
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Table 7-40
Most Reactive Fuel Lattice
(7 Pages)

GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
ABB-10-1	0.9141	0.0008	0.9157	
ABB-10-1	0.9147	0.0009	0.9166	
ABB-10-1	0.9170	0.0009	0.9188	
ABB-10-1	0.9106	0.0011	0.9128	
ABB-10-2	0.9159	0.0009	0.9177	
ABB-10-2	0.9152	0.0009	0.9170	
ABB-10-2	0.9187	0.0008	0.9204	
ABB-10-2	0.9127	0.0009	0.9145	
ABB-10-3	0.9226	0.0008	0.9242	
ABB-10-3	0.9229	0.0008	0.9246	
ABB-10-3	0.9262	0.0008	0.9279	
ABB-10-3	0.9215	0.0009	0.9233	
ABB-10-4	0.9130	0.0008	0.9147	
ABB-10-4	0.9135	0.0010	0.9155	
ABB-10-4	0.9162	0.0008	0.9178	
ABB-10-4	0.9105	0.0009	0.9123	
ABB-10-5	0.8774	0.0009	0.8792	
ABB-10-5	0.8781	0.0008	0.8796	
ABB-10-5	0.8788	0.0008	0.8804	
ABB-10-5	0.8783	0.0008	0.8798	
ABB-10-6	0.9129	0.0008	0.9146	
ABB-10-6	0.9147	0.0009	0.9164	
ABB-10-6	0.9154	0.0008	0.9170	
ABB-10-6	0.9105	0.0008	0.9121	
ABB-8-1	0.9116	0.0009	0.9134	
ABB-8-1	0.9118	0.0009	0.9135	
ABB-8-1	0.9135	0.0008	0.9152	
ABB-8-1	0.9058	0.0008	0.9073	
ABB-8-2	0.9158	0.0008	0.9175	
ABB-8-2	0.9164	0.0008	0.9180	
ABB-8-2	0.9175	0.0009	0.9192	
ABB-8-2	0.9117	0.0008	0.9132	

Table 7-40
Most Reactive Fuel Lattice
(7 Pages)

GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
ATRIUM-10	0.9266	0.0009	0.9283	
ATRIUM-10	0.9271	0.0009	0.9289	
ATRIUM-10	0.9301	0.0008	0.9317	
ATRIUM-10	0.9215	0.0009	0.9233	
ATRIUM-10	0.9323	0.0009	0.9341	
ATRIUM-10	0.9317	0.0009	0.9335	
ATRIUM-10	0.9353	0.0008	0.9369	
ATRIUM-10	0.9257	0.0009	0.9274	
ENC- IIIA	0.9234	0.0009	0.9251	
ENC- IIIA	0.9226	0.0008	0.9242	
ENC- IIIA	0.9260	0.0009	0.9277	
ENC- IIIA	0.9182	0.0010	0.9201	
ENC- IIIA	0.9205	0.0008	0.9222	
ENC- IIIA	0.9238	0.0008	0.9254	
ENC- IIIA	0.9240	0.0008	0.9256	
ENC- IIIA	0.9166	0.0008	0.9182	
ENC-III	0.9198	0.0009	0.9215	
ENC-III	0.9199	0.0008	0.9215	
ENC-III	0.9240	0.0009	0.9258	
ENC-III	0.9136	0.0009	0.9154	
ENC-III	0.9196	0.0009	0.9213	
ENC-III	0.9206	0.0010	0.9225	
ENC-III	0.9221	0.0009	0.9238	
ENC-III	0.9174	0.0010	0.9194	
ENC Va and Vb	0.9067	0.0008	0.9083	
ENC Va and Vb	0.9099	0.0009	0.9118	
ENC Va and Vb	0.9118	0.0009	0.9135	
ENC Va and Vb	0.9023	0.0008	0.9039	
FANP 9X9	0.9275	0.0008	0.9292	
FANP 9X9	0.9251	0.0010	0.9270	
FANP 9X9	0.9286	0.0009	0.9303	
FANP 9X9	0.9218	0.0008	0.9234	

Table 7-40
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GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
FANP 8X8-2	0.9215	0.0010	0.9235	
FANP 8X8-2	0.9191	0.0009	0.9208	
FANP 8X8-2	0.9232	0.0010	0.9251	
FANP 8X8-2	0.9147	0.0008	0.9164	
FANP 8X8-2	0.9238	0.0008	0.9254	
FANP 8X8-2	0.9236	0.0009	0.9255	
FANP 8X8-2	0.9265	0.0008	0.9282	
FANP 8X8-2	0.9190	0.0008	0.9207	
FANP 8X8-2	0.9246	0.0009	0.9264	
FANP 8X8-2	0.9242	0.0008	0.9258	
FANP 8X8-2	0.9264	0.0008	0.9280	
FANP 8X8-2	0.9170	0.0009	0.9188	
FANP 9X9-2	0.9307	0.0009	0.9324	
FANP 9X9-2	0.9307	0.0008	0.9324	
FANP 9X9-2	**0.9346**	**0.0009**	**0.9364**	
FANP 9X9-2	0.9260	0.0010	0.9280	
FANP 9X9-2	0.9280	0.0008	0.9297	
FANP 9X9-2	0.9277	0.0009	0.9296	
FANP 9X9-2	0.9307	0.0009	0.9325	
FANP 9X9-2	0.9223	0.0010	0.9242	
GE11, GE13	0.9267	0.0009	0.9285	
GE11, GE13	0.9287	0.0010	0.9307	
GE11, GE13	0.9307	0.0008	0.9324	
GE11, GE13	0.9222	0.0008	0.9238	
GE12, GE14	0.9293	0.0010	0.9312	
GE12, GE14	0.9323	0.0008	0.9339	
GE12, GE14	0.9328	0.0009	0.9346	
GE12, GE14	0.9232	0.0009	0.9250	
GE1, GE2, GE3	0.9276	0.0008	0.9293	
GE1, GE2, GE3	0.9270	0.0008	0.9286	
GE1, GE2, GE3	0.9302	0.0008	0.9318	
GE1, GE2, GE3	0.9224	0.0008	0.9240	

Table 7-40
Most Reactive Fuel Lattice
(7 Pages)

GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
GE1, GE2, GE3	0.9226	0.0008	0.9242	
GE1, GE2, GE3	0.9228	0.0008	0.9244	
GE1, GE2, GE3	0.9245	0.0009	0.9263	
GE1, GE2, GE3	0.9171	0.0009	0.9188	
GE4	0.9194	0.0009	0.9211	
GE4	0.9188	0.0008	0.9204	
GE4	0.9206	0.0008	0.9222	
GE4	0.9107	0.0009	0.9124	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9285	0.0009	0.9302	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9281	0.0009	0.9299	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9299	0.0009	0.9316	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9209	0.0012	0.9233	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9278	0.0008	0.9294	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9282	0.0008	0.9298	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9319	0.0009	0.9336	
GE-5, GE-Pres, GE-Barrier GE8 Type I	0.9209	0.0009	0.9226	
GE8 Type II	0.9267	0.0009	0.9285	
GE8 Type II	0.9267	0.0009	0.9284	
GE8 Type II	0.9289	0.0008	0.9305	
GE8 Type II	0.9206	0.0009	0.9223	
GE8 Type II	0.9285	0.0009	0.9302	
GE8 Type II	0.9280	0.0009	0.9297	
GE8 Type II	0.9309	0.0009	0.9327	
GE8 Type II	0.9202	0.0008	0.9218	
GE9, GE10	0.9272	0.0008	0.9288	
GE9, GE10	0.9302	0.0008	0.9318	

Table 7-40
Most Reactive Fuel Lattice
(7 Pages)

GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
GE9, GE10	0.9300	0.0008	0.9316	
GE9, GE10	0.9234	0.0008	0.9251	
GNF2	0.9340	0.0008	0.9356	
GNF2	0.9344	0.0008	0.9360	
GNF2	**0.9367**	**0.0010**	**0.9386**	
GNF2	0.9287	0.0010	0.9307	
Japan-BWR 8x8 Step II	0.9289	0.0009	0.9306	
Japan-BWR 8x8 Step II	0.9292	0.0010	0.9311	
Japan-BWR 8x8 Step II	0.9315	0.0008	0.9332	
Japan-BWR 8x8 Step II	0.9217	0.0009	0.9234	
KKL-BWR 10/15	0.9259	0.0009	0.9277	
KKL-BWR 10/15	0.9261	0.0009	0.9280	
KKL-BWR 10/15	0.9278	0.0008	0.9295	
KKL-BWR 10/15	0.9232	0.0008	0.9248	
KKL-BWR 10/15	0.9222	0.0009	0.9239	
KKL-BWR 10/15	0.9246	0.0008	0.9262	
KKL-BWR 10/15	0.9265	0.0008	0.9281	
KKL-BWR 10/15	0.9214	0.0008	0.9230	
KKL-BWR 11/16	0.9405	0.0010	0.9424	
KKL-BWR 11/16	0.9459	0.0008	0.9476	
KKL-BWR 11/16	0.9447	0.0009	0.9464	
KKL-BWR 11/16	0.9412	0.0008	0.9428	
KKL-BWR 1/4	0.9271	0.0010	0.9291	
KKL-BWR 1/4	0.9279	0.0009	0.9296	
KKL-BWR 1/4	0.9304	0.0008	0.9319	
KKL-BWR 1/4	0.9206	0.0008	0.9222	
KKL-BWR 2/5/8	0.9268	0.0009	0.9285	
KKL-BWR 2/5/8	0.9268	0.0010	0.9287	
KKL-BWR 2/5/8	0.9308	0.0008	0.9324	
KKL-BWR 2/5/8	0.9220	0.0010	0.9240	
KKL-BWR 3/9/12/13	0.9239	0.0009	0.9258	
KKL-BWR 3/9/12/13	0.9241	0.0009	0.9259	

Table 7-40
Most Reactive Fuel Lattice

(7 Pages)

GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
KKL-BWR 3/9/12/13	0.9266	0.0008	0.9281	
KKL-BWR 3/9/12/13	0.9194	0.0009	0.9212	
KKL-BWR 6	0.9116	0.0009	0.9134	
KKL-BWR 6	0.9118	0.0009	0.9135	
KKL-BWR 6	0.9135	0.0008	0.9152	
KKL-BWR 6	0.9058	0.0008	0.9073	
KKL-BWR 6	0.9053	0.0008	0.9069	
KKL-BWR 6	0.9052	0.0009	0.9069	
KKL-BWR 6	0.9102	0.0008	0.9118	
KKL-BWR 6	0.9010	0.0010	0.9030	
KKL-BWR 7/14	0.9155	0.0008	0.9172	
KKL-BWR 7/14	0.9150	0.0009	0.9167	
KKL-BWR 7/14	0.9200	0.0008	0.9216	
KKL-BWR 7/14	0.9130	0.0009	0.9148	
SVEA-96Opt2	0.9428	0.0008	0.9445	
SVEA-96Opt2	0.9434	0.0008	0.9450	
SVEA-96Opt2	0.9457	0.0008	0.9473	
SVEA-96Opt2	0.9427	0.0008	0.9443	
SVEA-96Opt	0.9183	0.0008	0.9200	
SVEA-96Opt2	0.9207	0.0008	0.9224	
SVEA-96Opt2	0.9232	0.0008	0.9248	
SVEA-96Opt2	0.9180	0.0010	0.9199	
SVEA-96Opt2	0.9005	0.0009	0.9023	
SVEA-96Opt2	0.9006	0.0009	0.9025	
SVEA-96Opt2	0.9033	0.0008	0.9049	
SVEA-96Opt2	0.8962	0.0009	0.8979	
SVEA-96Opt	0.9187	0.0010	0.9207	
SVEA-96Opt2	0.9178	0.0008	0.9194	
SVEA-96Opt	0.9219	0.0008	0.9235	
SVEA-96Opt	0.9163	0.0008	0.9179	
SVEA-96Opt	0.9163	0.0008	0.9179	
SVEA-96Opt	0.9173	0.0009	0.9190	

Table 7-40
Most Reactive Fuel Lattice
(7 Pages)

GE or Equivalent Reload Fuel Designation	k_{keno}	σ_{keno}	k_{eff}	Channel Size
SVEA-96Opt	0.9194	0.0009	0.9211	
SVEA-96Opt	0.9143	0.0009	0.9161	
Siemens QFA 9x9	0.9286	0.0009	0.9303	
Siemens QFA 9x9	0.9300	0.0007	0.9314	
Siemens QFA 9x9	0.9324	0.0009	0.9342	
Siemens QFA 9x9	0.9231	0.0009	0.9248	
XXX-RCN	0.9129	0.0009	0.9147	
XXX-RCN	0.9153	0.0009	0.9170	
XXX-RCN	0.9157	0.0009	0.9175	
XXX-RCN	0.9087	0.0010	0.9107	
STD GE-4 w/ higher exp.	0.9194	0.0009	0.9211	
STD GE-4 w/ higher exp.	0.9188	0.0008	0.9204	
STD GE-4 w/ higher exp.	0.9206	0.0008	0.9222	
STD GE-4 w/ higher exp.	0.9107	0.0009	0.9124	

Table 7-41
EOS-89BTH Dimensions of System Components with Fabrication Tolerance

Components	Dimension (inch)			
	Nominal	Minimum(1)	Maximum(1)	Tolerance (3)
Fuel Rod OD(4)	0.424	0.419	0.429	0.005
Steel Plate(5)	[ ]	[ ]	[ ]	[ ]
	[ ]	[ ]	[ ]	[ ]
Aluminum Plate	[ ]	[ ]	[ ]	[ ]
	[ ]	[ ]	[ ]	[ ]
Poison Plate	[ ]	[ ]	[ ]	[ ]
DSC shell thickness	0.5	0.49	0.55	0.05
Lead Gamma Shield Thickness(2)	2.5		3.625	

Notes:

(1) The maximum is the nominal plus the fabrication tolerance; the minimum is the nominal minus the fabrication tolerance.

(2) The maximum lead thickness represents the TC125 (Table 7-5).

(3) The tolerance information is assumed; the nominal value is for FANP 9X9-2 FAs.

(4) The steel plate dimension used in the determination of minimum B-10 requirements is 0.27” (HVA, HVB) and 0.25" (HVC-F).

Table 7-42
Most Reactive Configuration
(4 Pages)

Model Description	k_{keno}	σ_{keno}	k_{eff}
Fuel Assemblies Position in the Fuel Compartment			
Fuel Assemblies Centered	0.9300	0.0008	0.9316
Fuel Assemblies Positioned Inwardly	**0.9346**	**0.0009**	**0.9364**
2: Fuel Rod OD			
[ ]	0.9337	0.0009	0.9355
[ ]	**0.9343**	**0.0009**	**0.9361**
[ ]	0.9346	0.0009	<u>0.9364</u>
Gaps at Plate Joints			
[ ]	0.9353	0.0009	0.9371
[ ]	0.9354	0.0008	0.9370
[ ]	0.9344	0.0008	0.9360
[ ]	0.9356	0.0008	0.9372
[ ]	0.9343	0.0008	0.9359
[ ]	0.9351	0.0008	0.9367
[ ]	0.9352	0.0008	0.9368
[ ]	0.9343	0.0008	0.9359
[ ]	0.9351	0.0009	0.9369
[ ]	0.9343	0.0009	0.9361
[ ]	0.9356	0.0008	<u>0.9372</u>
[ ]	**0.9347**	**0.0008**	**0.9363**
Steel Plate Thickness			
[ ]	0.9334	0.0009	0.9352
[ ]	0.9347	0.0008	0.9363
[ ]	0.9344	0.0008	0.9360
[ ]	**0.9355**	**0.0008**	**0.9371**
[ ]	0.9339	0.0008	0.9355

Table 7-42
Most Reactive Configuration

(4 Pages)

Model Description	k_{keno}	σ_{keno}	k_{eff}
[ ]	0.9370	0.0008	0.9386
[ ]	0.9347	0.0008	0.9363
Al Plate Thickness			
[ ]	0.9349	0.0008	0.9365
[ ]	0.9355	0.0008	0.9371
[ ]	0.9356	0.0009	0.9374
[ ]	0.9364	0.0009	0.9382
[ ]	**0.9356**	**0.0009**	**0.9374**
[ ]	0.9347	0.0008	0.9363
Poison Plate Thickness (B-10 Content is constant - 29 mg/cm^2)			
[ ]	0.9364	0.0008	0.9380
[ ]	**0.9356**	**0.0009**	**0.9374**
[ ]	0.9342	0.0009	0.9360
DSC Shell Thickness			
[ ]	0.9344	0.0009	0.9362
[ ]	**0.9356**	**0.0009**	**0.9374**
[ ]	0.9363	0.0008	0.9379
Lead Shield Thickness			
[ ]	**0.9356**	**0.0009**	**0.9374**
[ ]	0.9339	0.0009	0.9357
Transition Rail Material			
[ ]	0.9296	0.0008	0.9312
[ ]	0.9316	0.0009	0.9334
[ ]	0.9300	0.0008	0.9316
[ ]	0.9314	0.0009	0.9332

Table 7-42
Most Reactive Configuration

(4 Pages)

Model Description	k_{keno}	σ_{keno}	k_{eff}
[ ]	0.9321	0.0008	0.9337
[ ]	0.9329	0.0008	0.9345
[ ]	**0.9356**	**0.0009**	**0.9374**
Internal Moderator Density (IMD)			
IMD = 1% TD	0.4518	0.0004	0.4526
IMD = 10% TD	0.4882	0.0004	0.4890
IMD = 20% TD	0.5475	0.0006	0.5487
IMD = 30% TD	0.6169	0.0006	0.6181
IMD = 40% TD	0.6824	0.0007	0.6838
IMD = 50% TD	0.7408	0.0007	0.7422
IMD = 60% TD	0.7931	0.0008	0.7947
IMD = 70% TD	0.8359	0.0008	0.8375
IMD = 80% TD	0.8759	0.0009	0.8777
IMD = 90% TD	0.9073	0.0008	0.9089
IMD = 100% TD	**0.9356**	**0.0009**	**0.9374**
Gap between DSC and TC			
Gap with void	**0.9401**	**0.0008**	**0.9417**
Gap with water (1% TD)	0.9403	0.0008	0.9419
Gap with water (10% TD)	0.9393	0.0008	0.9409
Gap with water (20% TD)	0.9373	0.0008	0.9389
Gap with water (30% TD)	0.9372	0.0008	0.9388
Gap with water (40% TD)	0.9368	0.0009	0.9386
Gap with water (50% TD)	0.9364	0.0008	0.9380
Gap with water (60% TD)	0.9341	0.0009	0.9359
Gap with water (70% TD)	0.9361	0.0008	0.9377
Gap with water (80% TD)	0.9357	0.0009	0.9375
Gap with water (90% TD)	0.9346	0.0009	0.9364
Gap with water (100% TD)	0.9356	0.0009	0.9374
External Moderator Density (EMD)			
EMD = zero (void)	**0.9401**	**0.0008**	**0.9417**
EMD = 1% TD	0.9379	0.0009	0.9397
EMD = 10% TD	0.9367	0.0008	0.9383

Table 7-42
Most Reactive Configuration

(4 Pages)

Model Description	k_{keno}	σ_{keno}	k_{eff}
EMD = 20% TD	0.9368	0.0009	0.9386
EMD = 30% TD	0.9352	0.0008	0.9368
EMD = 40% TD	0.9366	0.0008	0.9382
EMD = 50% TD	0.9363	0.0008	0.9379
EMD = 60% TD	0.9344	0.0009	0.9362
EMD = 70% TD	0.9353	0.0008	0.9369
EMD = 80% TD	0.9345	0.0008	0.9361
EMD = 90% TD	0.9350	0.0008	0.9366
EMD = 100% TD	0.9356	0.0008	0.9372
Constant Poison Plate Density Definition			
EMD = zero (void)	0.9408	0.0009	<u>0.9426</u>

Table 7-43
Determination of Minimum Poison Loading Requirement

Basket Type	**Enrichment (wt% of U-235)**	**B-10 Content (mg/cm^2)**	k_{keno}	σ_{keno}	k_{eff}
All Fuel Except ABB-10-C					
MMC	4.1	29.4	0.9343	0.0008	0.9359
	4.45	37.2	0.9369	0.0009	0.9387
BORAL®	**4.8**	**45.0**	**0.9382**	**0.0008**	**0.9398**
ABB-10-C Fuel					
MMC	3.85	29.4	0.9271	0.0008	0.9287
	4.25	37.2	0.9329	0.0009	0.9347
BORAL®	4.55	45.0	0.9347	0.0008	0.9363

Table 7-44
Criticality Results

EOS-37PTH: Regulatory Requirements for Storage			
Dry Storage (Bounded by infinite array of undamaged storage casks)	0.6203	0.0003	0.6209
Normal Loading and Unloading Conditions (Optimum Moderator Density)	0.9371	0.0007	0.9385
EOS-89BTH: Regulatory Requirements for Storage			
Dry Storage (Bounded by infinite array of undamaged storage casks)	0.5187	0.0003	0.5193
Normal Loading and Unloading Conditions (Optimum Moderator Density)	0.9382	0.0008	0.9398

Table 7-45
Comparison of Materials used in Design Calculation and Benchmark Models

	System Application	Benchmark KENO V.a Models
Tank/Canister	Carbon steel	none
Support structures	6061-aluminum plates Stainless steel Concrete Poison plates in aluminium matrix	6061-aluminum plates 1100-aluminum plates 5052-aluminum plates D16-aluminum alloy plates Acrylic support plates Lucite plates
Fuel	UO_2	UO_2
Clad	Stainless steel Zircaloy-4 Zircaloy-2	Stainless steel Zircaloy-4 Zircaloy-2 6061-aluminum
Moderator	Pure water Water with soluble boron	Pure water Water with soluble boron
Reflecting material	Water Lead Steel	Water Lead Steel Depleted uranium

Table 7-46
Benchmark Experimental KENO V.a Simulation Results
(5 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{eff}	σ
LCT-001-001	2.35	2.032	-	0	2.918	36.24	9.64E-02	0.9954	0.0009
LCT-001-002	2.35	2.032	11.92	0	2.918	36.26	9.56E-02	0.9951	0.0009
LCT-001-003	2.35	2.032	8.41	0	2.918	36.29	9.46E-02	0.9955	0.0009
LCT-001-004	2.35	2.032	10.05	0	2.918	36.27	9.53E-02	0.9946	0.0009
LCT-001-005	2.35	2.032	6.39	0	2.918	36.31	9.40E-00	0.9932	0.0009
LCT-001-006	2.35	2.032	8.01	0	2.918	36.27	9.53E-02	0.9955	0.0009
LCT-001-007	2.35	2.032	4.46	0	2.918	36.32	9.35E-02	0.9935	0.0008
LCT-001-008	2.35	2.032	7.57	0	2.918	36.30	9.42E-02	0.9926	0.0008
LCT-002-001	4.31	2.54	-	0	3.882	35.74	1.14E-01	0.9948	0.0010
LCT-002-002	4.31	2.54	-	0	3.882	35.74	1.14E-01	0.9977	0.0009
LCT-002-003	4.31	2.54	-	0	3.882	35.74	1.14E-01	0.9975	0.0009
LCT-002-004	4.31	2.54	10.62	0	3.882	35.77	1.13E-01	0.9969	0.0010
LCT-002-005	4.31	2.54	7.11	0	3.882	35.77	1.13E-01	0.9969	0.0010
LCT-008-001	2.459	1.636	-	1511	1.841	33.59	2.86E-01	0.9960	0.0006
LCT-008-002	2.459	1.636	-	1336	1.841	33.90	2.52E-01	0.9971	0.0008
LCT-008-003	2.459	1.636	-	1336	1.841	33.90	2.52E-01	0.9974	0.0007
LCT-008-004	2.459	1.636	-	1182	1.841	33.89	2.53E-01	0.9966	0.0006
LCT-008-005	2.459	1.636	-	1182	1.841	33.90	2.52E-01	0.9983	0.0007
LCT-008-006	2.459	1.636	-	1033	1.841	33.90	2.52E-01	0.9965	0.0006
LCT-008-007	2.459	1.636	-	1033	1.841	33.91	2.51E-01	0.9956	0.0006
LCT-008-008	2.459	1.636	-	794	1.841	33.93	2.49E-01	0.9950	0.0007

Table 7-46
Benchmark Experimental KENO V.a Simulation Results
(5 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{eff}	σ
LCT-008-009	2.459	1.636	-	779	1.841	33.93	2.50E-01	0.9950	0.0008
LCT-008-010	2.459	1.636	-	1245	1.841	33.88	2.54E-01	0.9972	0.0008
LCT-008-011	2.459	1.636	-	1384	1.841	33.81	2.61E-01	0.9979	0.0007
LCT-008-012	2.459	1.636	-	1348	1.841	33.88	2.53E-01	0.9970	0.0006
LCT-008-013	2.459	1.636	-	1348	1.841	33.87	2.55E-01	0.9980	0.0007
LCT-008-014	2.459	1.636	-	1363	1.841	33.84	2.57E-01	0.9968	0.0006
LCT-008-015	2.459	1.636	-	1363	1.841	33.85	2.56E-01	0.9972	0.0007
LCT-008-016	2.459	1.636	-	1158	1.841	34.10	2.32E-01	0.9963	0.0006
LCT-008-017	2.459	1.636	-	921	1.841	34.43	2.03E-01	0.9958	0.0007
LCT-010-005	4.31	2.54	14.26	0	3.882	33.42	3.90E-01	0.9988	0.0011
LCT-010-016	4.31	1.892	15.39	0	1.597	33.39	2.94E-01	1.0005	0.0009
LCT-010-017	4.31	1.892	15.36	0	1.597	33.46	2.87E-01	0.9999	0.0009
LCT-010-018	4.31	1.892	14.97	0	1.597	33.50	2.83E-01	0.9986	0.0009
LCT-010-019	4.31	1.892	13.34	0	1.597	33.58	2.76E-01	0.9983	0.0010
LCT-017-003	2.35	2.032	10.51	0	2.918	36.29	9.46E-02	0.9985	0.0008
LCT-017-004	2.35	2.032	11.09	0	2.918	34.74	2.13E-01	0.9947	0.0009
LCT-017-005	2.35	2.032	13.19	0	2.918	35.01	1.86E-01	0.9975	0.0009
LCT-017-006	2.35	2.032	13.37	0	2.918	35.15	1.74E-01	0.9989	0.0007
LCT-017-007	2.35	2.032	12.96	0	2.918	35.24	1.66E-01	0.9977	0.0008
LCT-017-008	2.35	2.032	9.95	0	2.918	35.62	1.36E-01	0.9938	0.0010
LCT-017-009	2.35	2.032	7.82	0	2.918	36.02	1.10E-01	0.9945	0.0008

Table 7-46
Benchmark Experimental KENO V.a Simulation Results
(5 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{eff}	σ
LCT-017-010	2.35	2.032	9.89	0	2.918	36.15	9.93E-02	0.9996	0.0009
LCT-017-011	2.35	2.032	10.44	0	2.918	36.20	9.75E-02	0.9994	0.0009
LCT-017-012	2.35	2.032	10.44	0	2.918	36.23	9.63E-02	0.9976	0.0008
LCT-017-013	2.35	2.032	9.6	0	2.918	36.28	9.48E-02	0.9964	0.0008
LCT-017-014	2.35	2.032	8.75	0	2.918	36.29	9.43E-02	0.9959	0.0008
LCT-017-015	2.35	1.684	8.57	0	1.600	34.74	1.79E-01	0.9962	0.0010
LCT-017-016	2.35	1.684	9.17	0	1.600	34.82	1.74E-01	0.9974	0.0009
LCT-017-017	2.35	1.684	9.1	0	1.600	34.89	1.69E-01	0.9983	0.0008
LCT-017-019	2.35	1.684	8.87	0	1.600	34.97	1.64E-01	0.9960	0.0009
LCT-017-020	2.35	1.684	8.65	0	1.600	35.00	1.63E-01	0.9933	0.0010
LCT-017-021	2.35	1.684	8.13	0	1.600	35.03	1.61E-01	0.9939	0.0008
LCT-017-022	2.35	1.684	7.26	0	1.600	35.05	1.60E-01	0.9932	0.0009
LCT-017-023	2.35	1.684	9.65	0	1.600	34.85	1.72E-01	0.9998	0.0009
LCT-017-024	2.35	1.684	9.7	0	1.600	34.93	1.67E-01	0.9977	0.0009
LCT-017-025	2.35	1.684	8.09	0	1.600	35.06	1.59E-01	0.9936	0.0009
LCT-017-028	2.35	1.684	7.65	0	1.600	33.88	3.02E-01	0.9953	0.0008
LCT-017-029	2.35	1.684	9.09	0	1.600	34.16	2.62E-01	0.9963	0.0010
LCT-042-001	2.35	1.684	8.28	0	1.600	34.86	1.72E-01	0.9965	0.0008
LCT-042-002	2.35	1.684	4.8	0	1.600	34.76	1.78E-01	0.9965	0.0009
LCT-042-003	2.35	1.684	2.69	0	1.600	34.68	1.85E-01	0.9962	0.0009
LCT-042-004	2.35	1.684	2.98	0	1.600	34.69	1.83E-01	0.9980	0.0009

Table 7-46
Benchmark Experimental KENO V.a Simulation Results
(5 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{eff}	σ
LCT-042-005	2.35	1.684	3.86	0	1.600	34.74	1.79E-01	0.9975	0.0008
LCT-042-006	2.35	1.684	7.79	0	1.600	34.84	1.72E-01	0.9975	0.0009
LCT-042-007	2.35	1.684	5.43	0	1.600	34.78	1.77E-01	0.9966	0.0008
LCT-050-001	4.738	1.3	-	0	2.032	34.24	2.04E-01	0.9964	0.0010
LCT-050-002	4.738	1.3	-	0	2.032	34.35	1.95E-01	0.9939	0.0010
LCT-050-003	4.738	1.3	-	822	2.032	34.14	2.12E-01	0.9948	0.0009
LCT-050-004	4.738	1.3	-	822	2.032	34.24	2.03E-01	0.9981	0.0009
LCT-050-005	4.738	1.3	-	5030	2.032	33.96	2.28E-01	0.9979	0.0009
LCT-050-006	4.738	1.3	-	5030	2.032	34.04	2.19E-01	0.9976	0.0009
LCT-050-007	4.738	1.3	-	5030	2.032	34.09	2.15E-01	0.9975	0.0010
LCT-051-001	2.459	1.636	4.91	143	1.841	35.23	1.48E-01	0.9942	0.0007
LCT-051-002	2.459	1.636	1.64	510	1.841	34.46	2.00E-01	0.9956	0.0009
LCT-051-003	2.459	1.636	1.64	514	1.841	34.46	2.00E-01	0.9951	0.0008
LCT-051-004	2.459	1.636	1.64	501	1.841	34.46	2.01E-01	0.9966	0.0009
LCT-051-005	2.459	1.636	1.64	493	1.841	34.45	2.02E-01	0.9946	0.0007
LCT-051-006	2.459	1.636	1.64	474	1.841	34.42	2.05E-01	0.9948	0.0009
LCT-051-007	2.459	1.636	1.64	462	1.841	34.42	2.05E-01	0.9939	0.0008
LCT-051-008	2.459	1.636	1.64	432	1.841	34.43	2.05E-01	0.9963	0.0009
LCT-051-009	2.459	1.636	3.27	217	1.841	34.89	1.69E-01	0.9933	0.0008
LCT-051-010	2.459	1.636	1.64	15	1.841	34.53	1.96E-01	0.9945	0.0008
LCT-051-011	2.459	1.636	1.64	28	1.841	34.51	1.98E-01	0.9921	0.0009

Table 7-46
Benchmark Experimental KENO V.a Simulation Results

(5 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{eff}	σ
LCT-051-012	2.459	1.636	1.64	92	1.841	34.48	1.99E-01	0.9901	0.0009
LCT-051-013	2.459	1.636	1.64	395	1.841	34.40	2.06E-01	0.9865	0.0009
LCT-051-014	2.459	1.636	3.27	121	1.841	34.85	1.72E-01	0.9865	0.0008
LCT-051-015	2.459	1.636	1.64	487	1.841	34.40	2.06E-01	0.9913	0.0007
LCT-051-016	2.459	1.636	3.27	197	1.841	34.84	1.73E-01	0.9893	0.0009
LCT-051-017	2.459	1.636	1.64	634	1.841	34.39	2.06E-01	0.9929	0.0008
LCT-051-018	2.459	1.636	3.27	320	1.841	34.83	1.73E-01	0.9901	0.0008
LCT-051-019	2.459	1.636	4.91	72	1.841	35.15	1.53E-01	0.9895	0.0009

Table 7-47
Correlation Coefficients |r| for Independent Parameters

Parameter	EOS-37PTH	EOS-89BTH
U-235 Enrichment	0.271	0.218
Pitch (cm)	0.170	0.053
Moderator to Fuel Volume Ratio	0.108	0.141
AEG	0.150	-
Soluble Boron (ppm)	0.506	-
EALF	-	0.340
Assembly Separation (cm)	0.612	0.487

Table 7-48
USL Evaluations

Equation Parameter	EOS-37PTH	EOS-89BTH
	Value	
$\bar{k}_{eff}$	0.9958	0.9964
S_P	2.70E-3	2.23E-3
U	2.0	2.065
Δ_{sm}	0.05	0.05
USL	**0.9404**	**0.9418**

Proprietary Information on Pages 7-103 through 7-104
Withheld Pursuant to 10 CFR 2.390

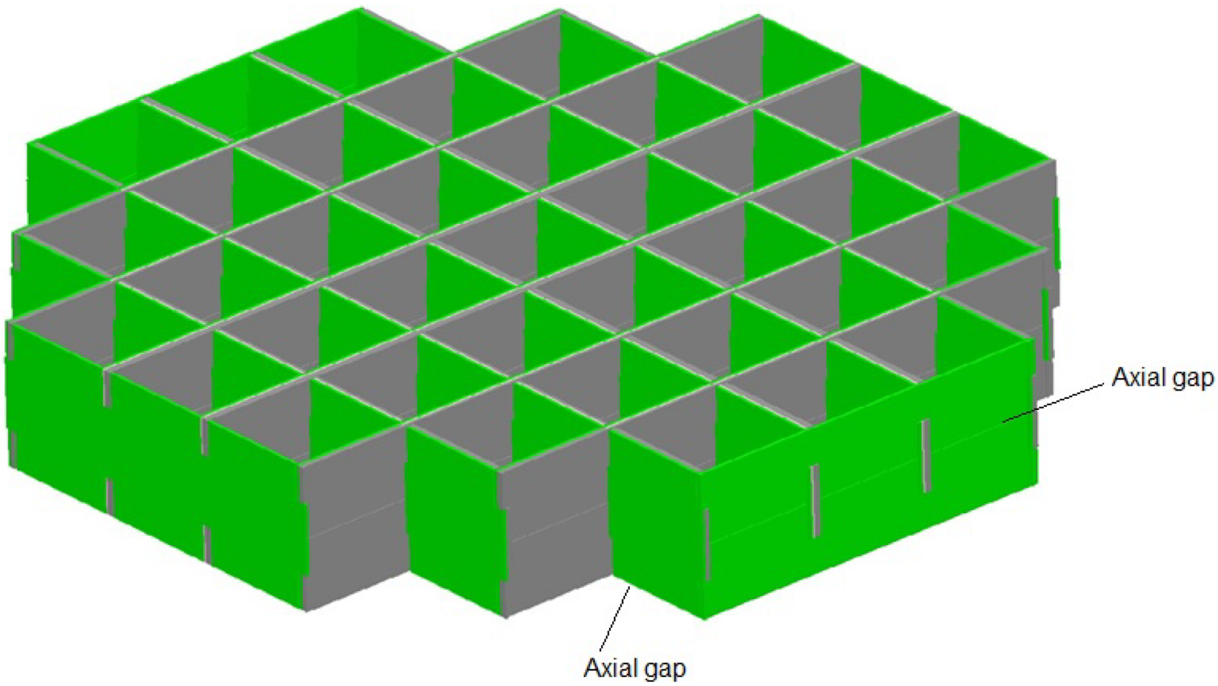

Figure 7-3
EOS-37PTH DSC Single Egg-Crate Basket Section

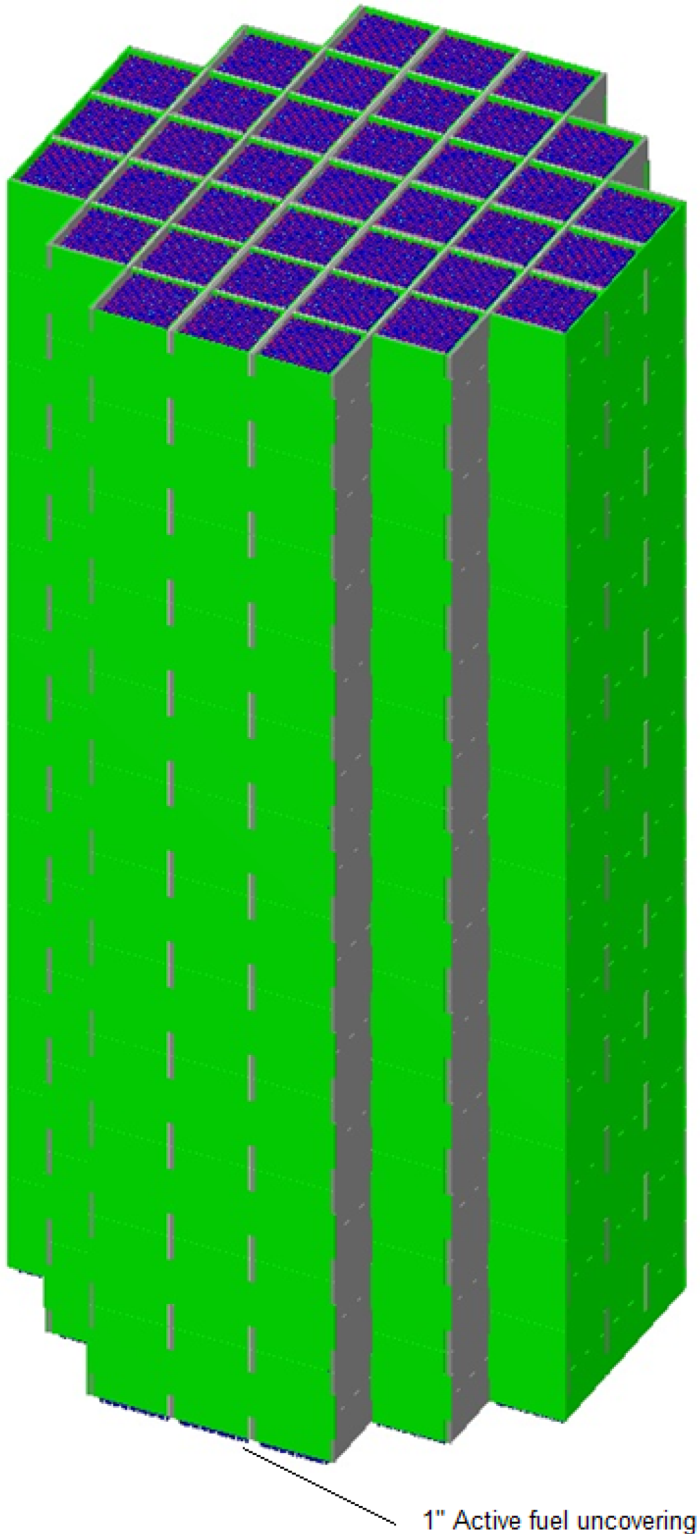

Figure 7-4
EOS-37PTH DSC Full-Length Axial Basket Structure with Fuel Assemblies

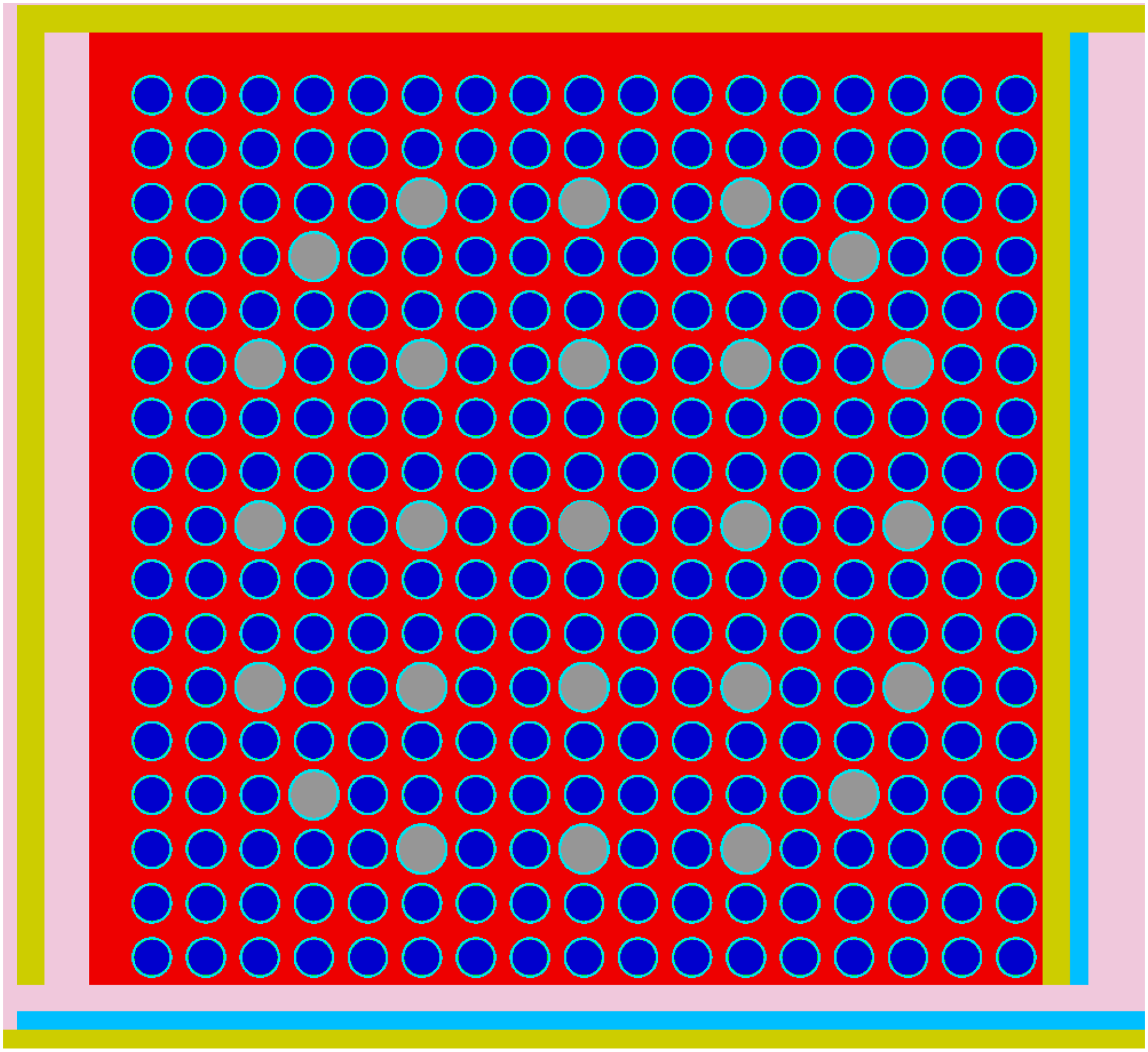

Figure 7-5
WE 17x17 Fuel Assembly with Non-Fuel Assembly Hardware

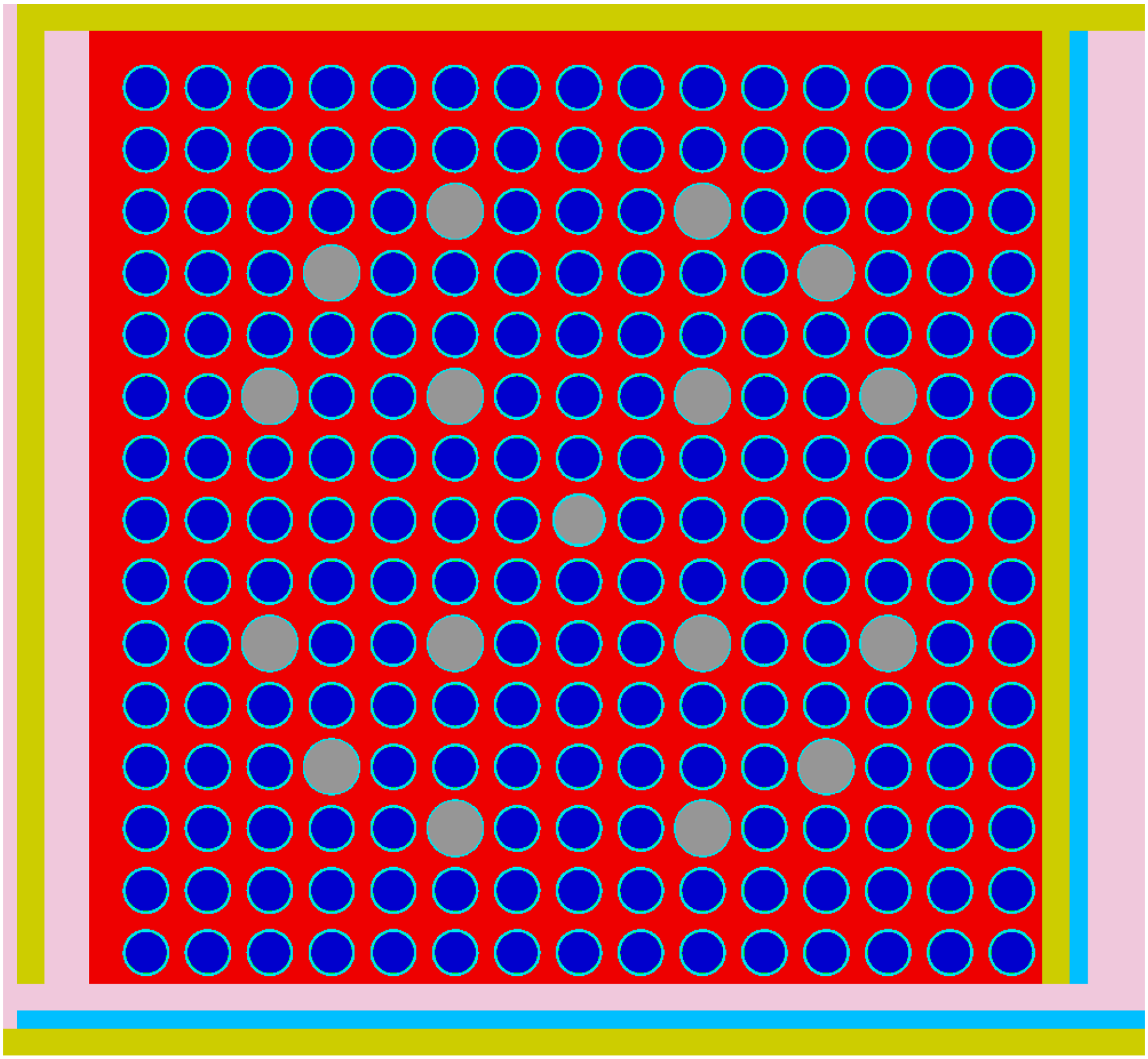

Figure 7-6
B&W 15x15 Fuel Assembly with Non-Fuel Assembly Hardware

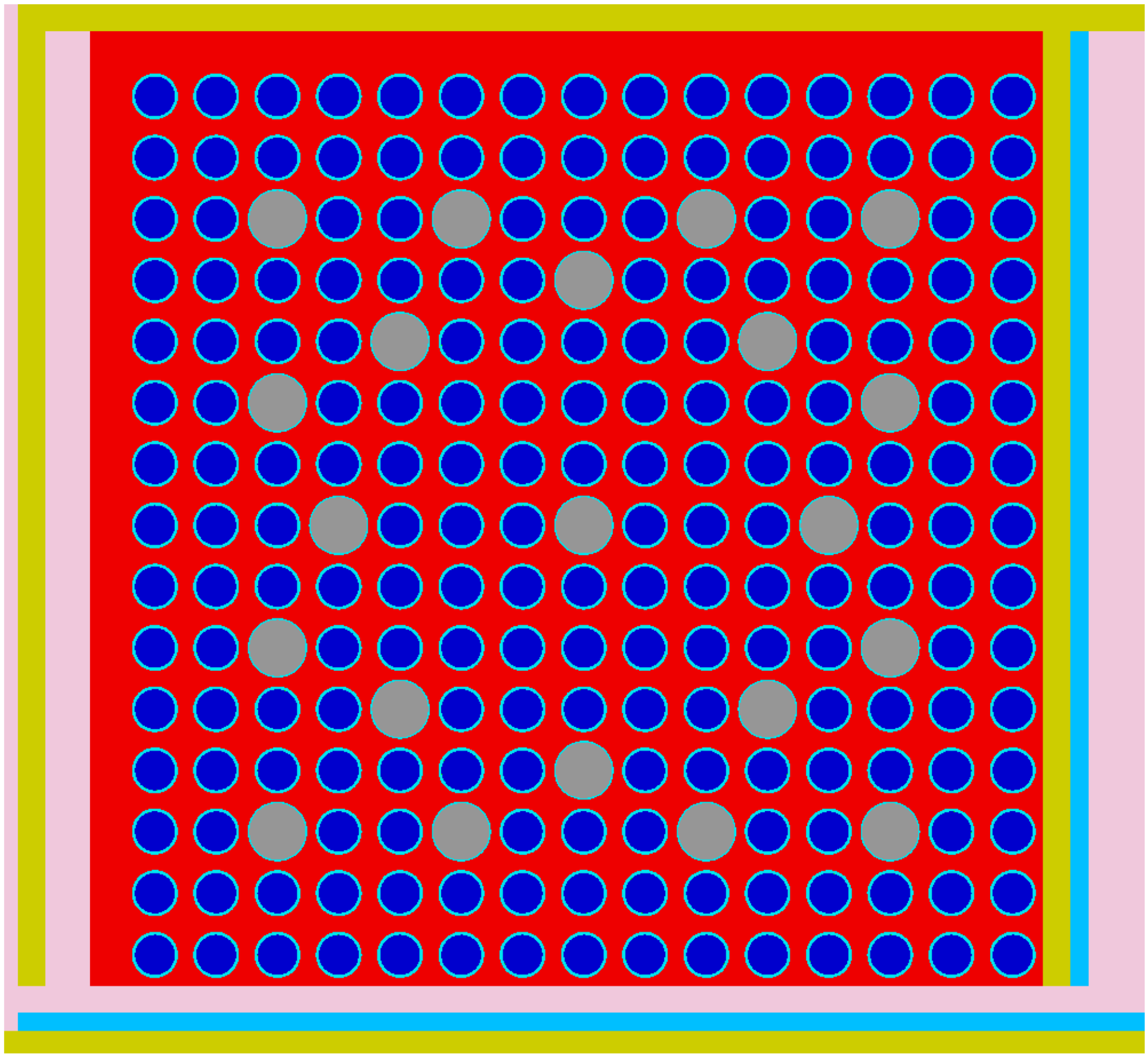

Figure 7-7
WE 15x15 Fuel Assembly with Non-Fuel Assembly Hardware

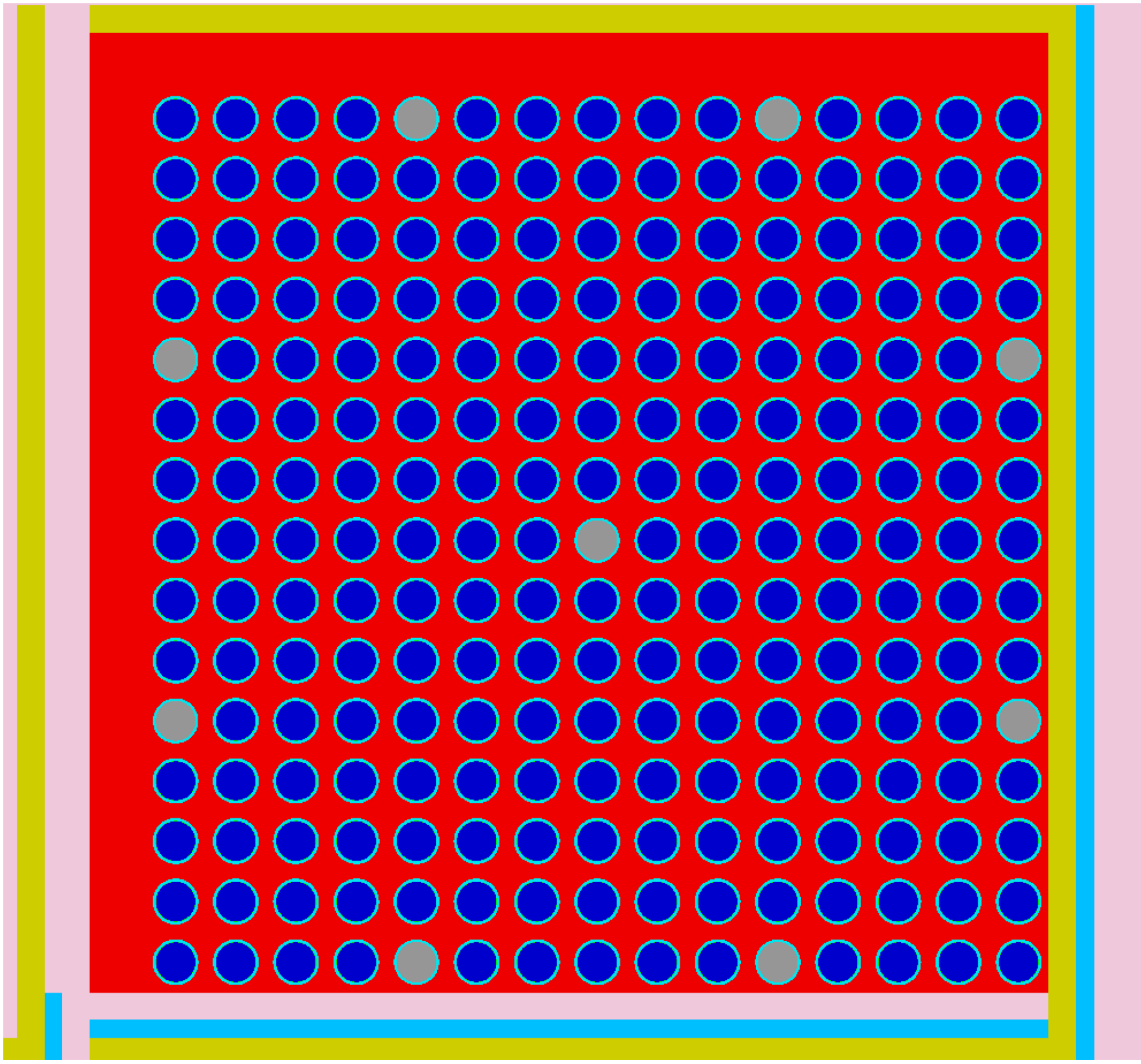

Figure 7-8
CE 15x15 Fuel Assembly with Non-Fuel Assembly Hardware

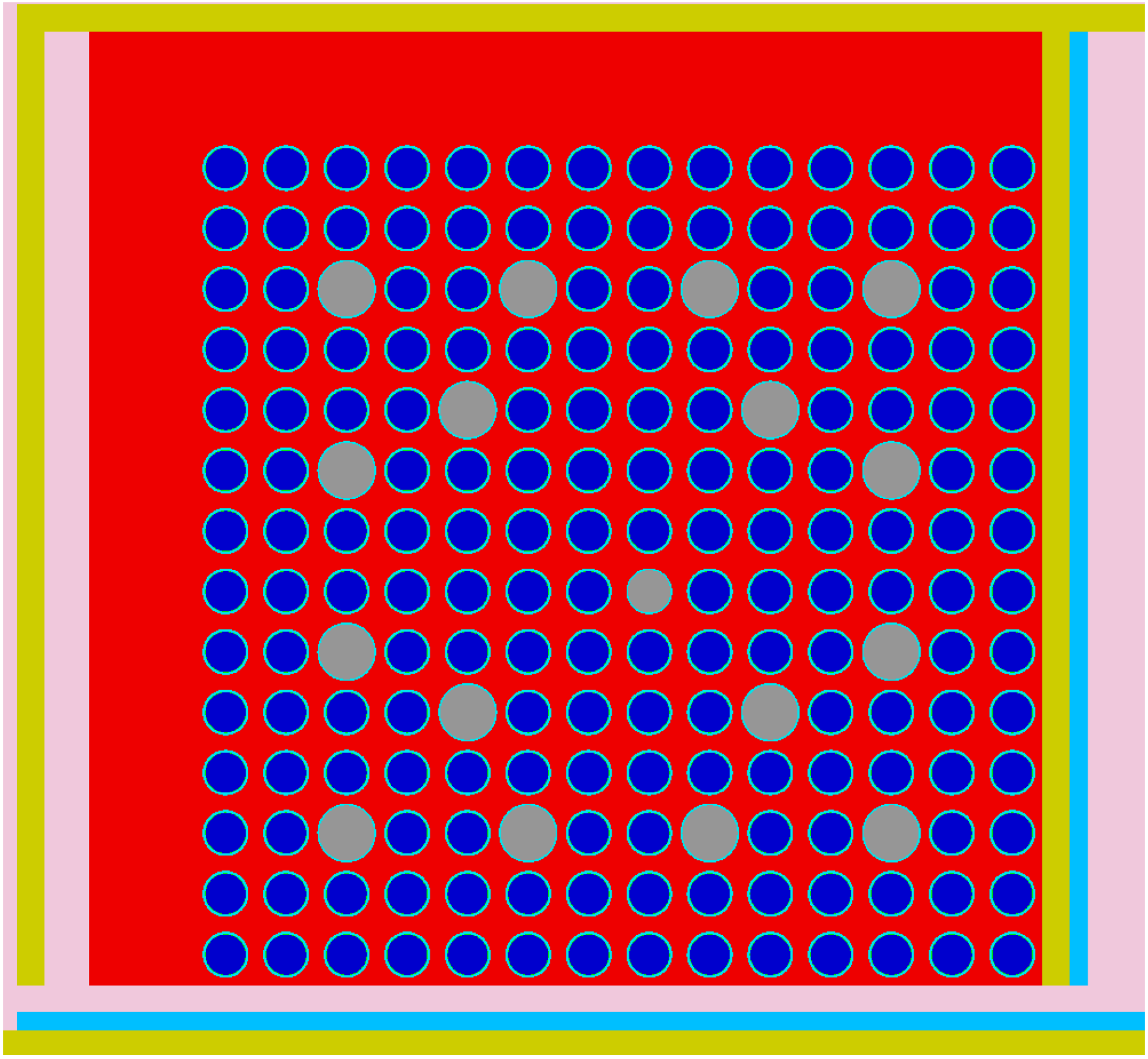

Figure 7-9
WE 14x14 Fuel Assembly with Non-Fuel Assembly Hardware

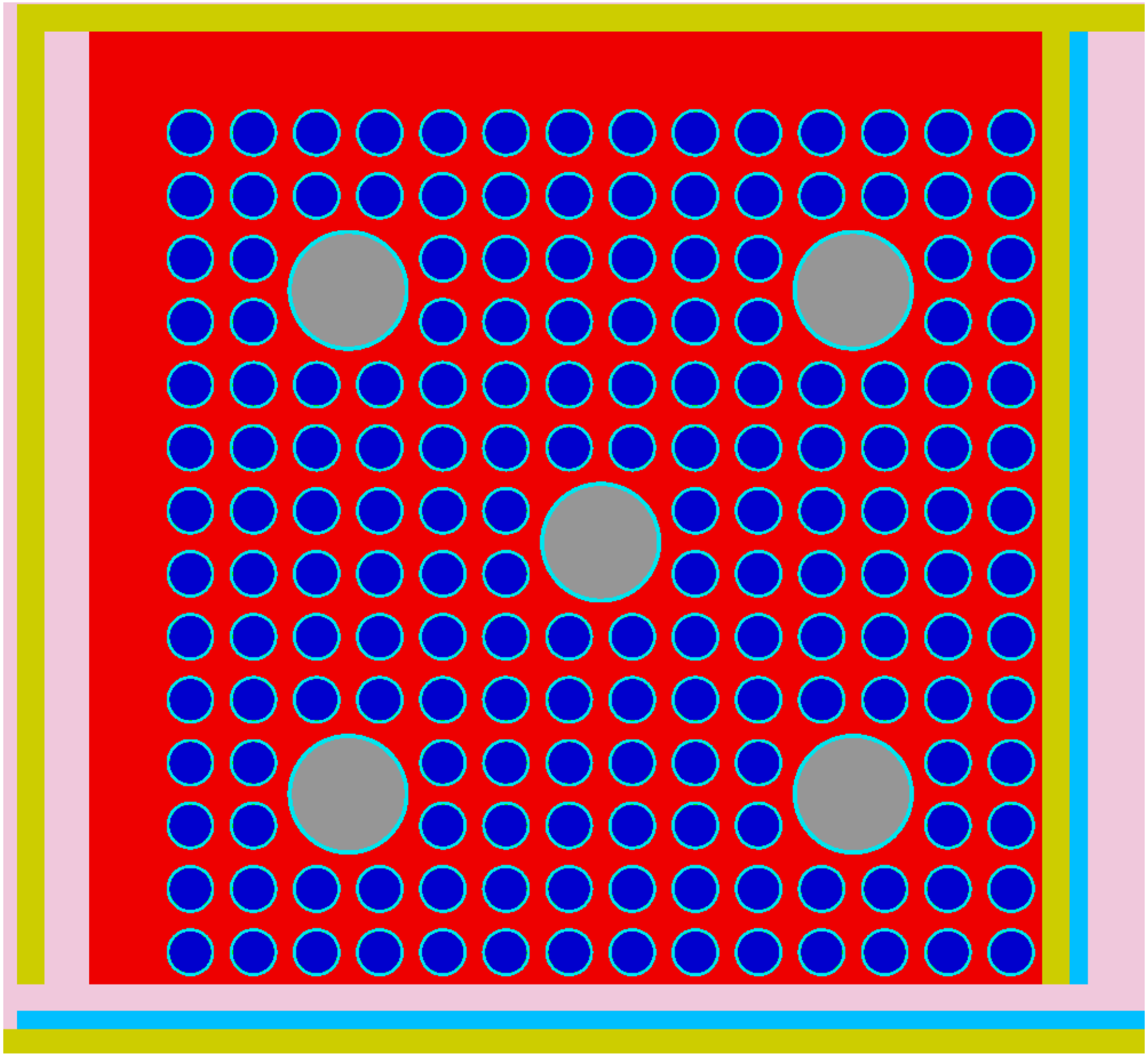

Figure 7-10
CE 14x14 Fuel Assembly with Non-Fuel Assembly Hardware

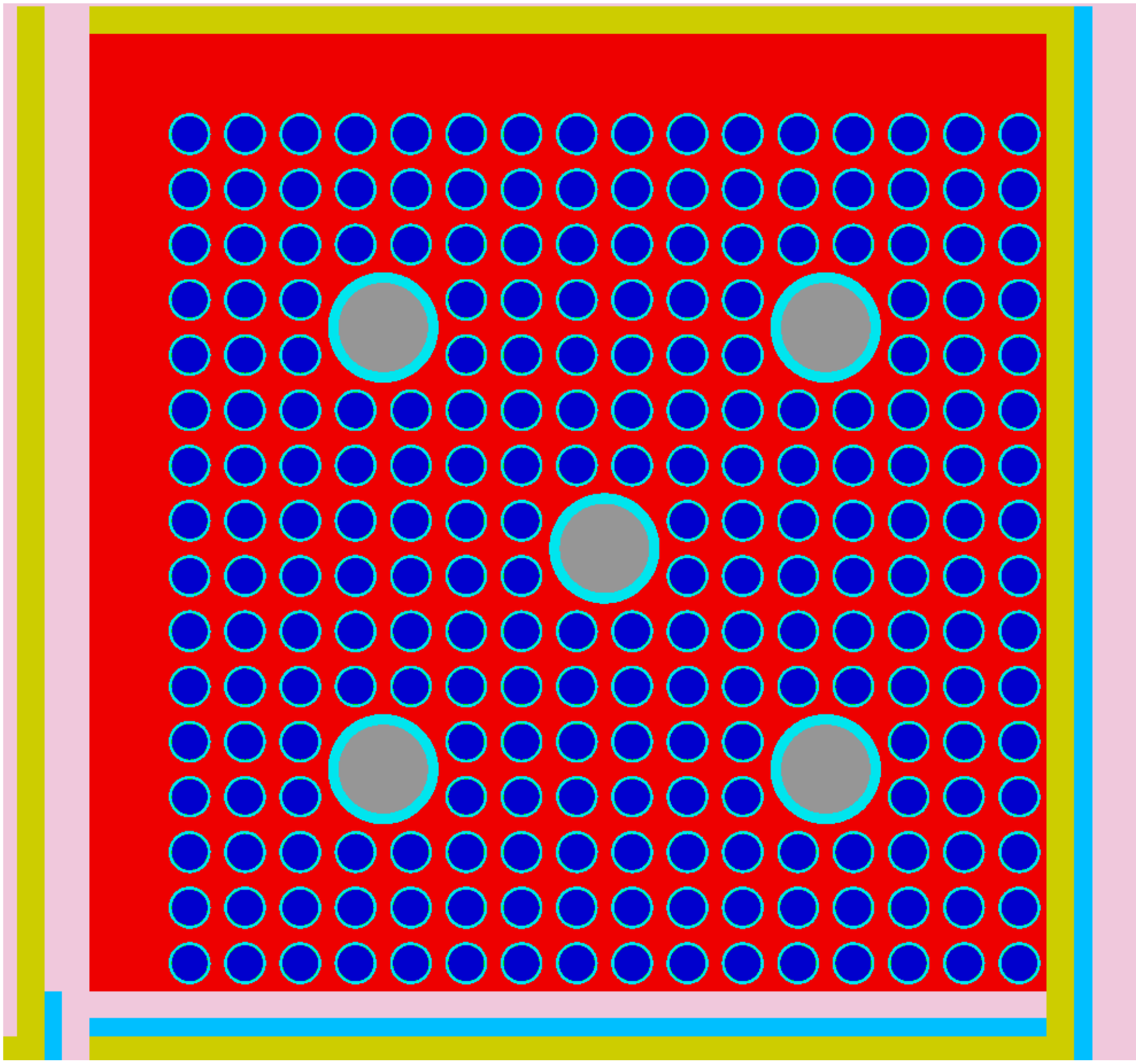

Figure 7-11
CE 16x16 Fuel Assembly with Non-Fuel Assembly Hardware

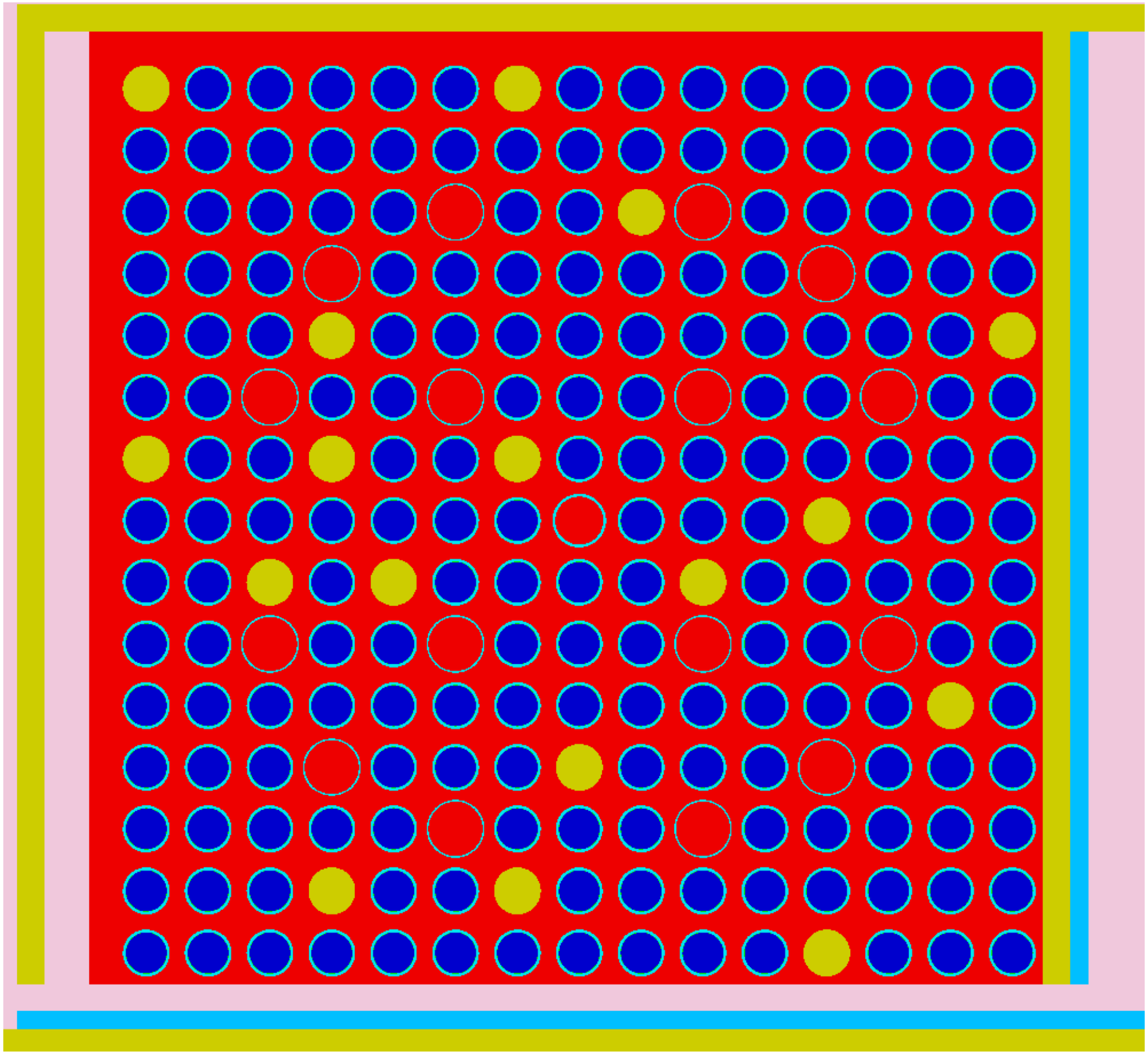

Figure 7-12
Reconstituted Fuel Assembly Study: 17 Stainless Steel Rods

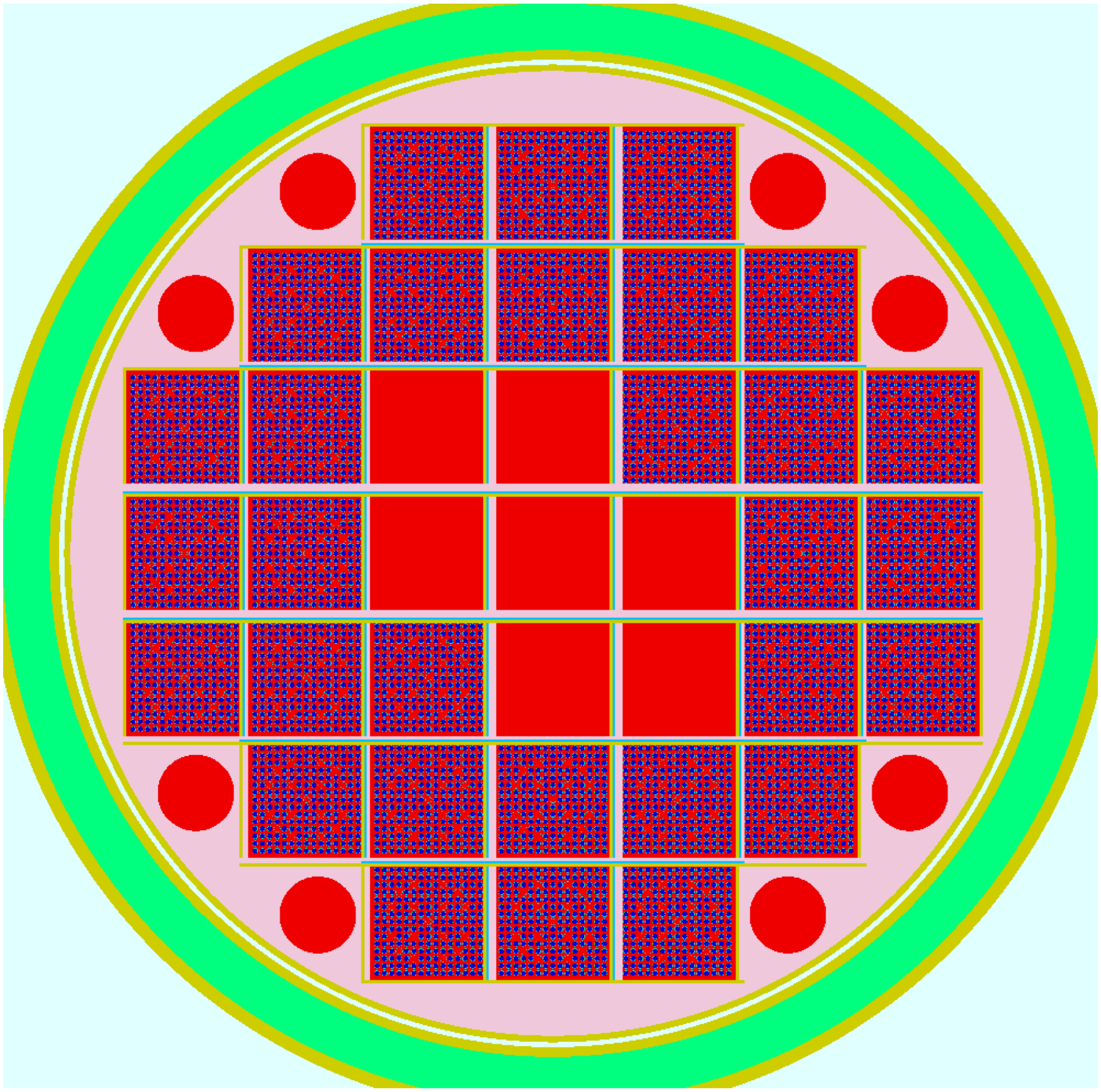

Figure 7-13
Empty Fuel Compartment Study: Seven Empty Compartments

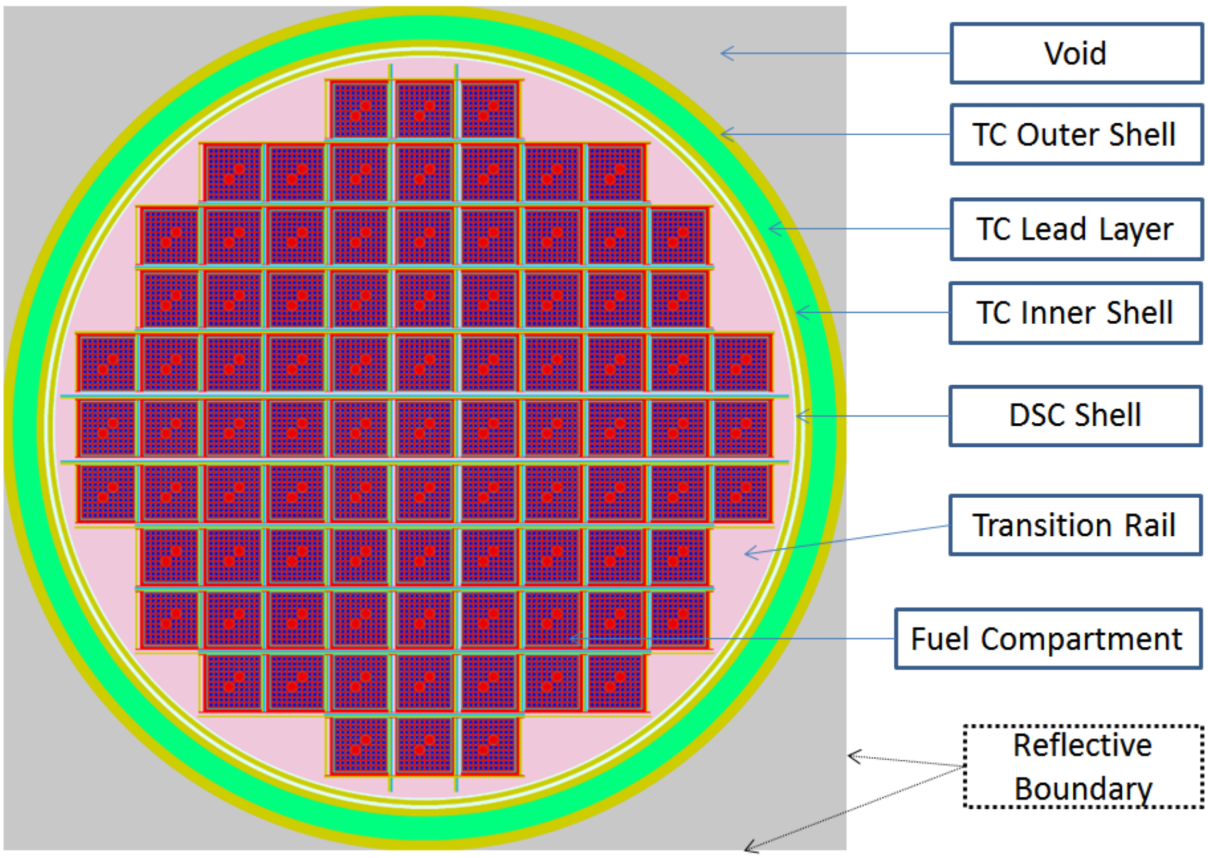

Figure 7-14
Radial Cross Section of the Model

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Withheld Pursuant to 10 CFR 2.390

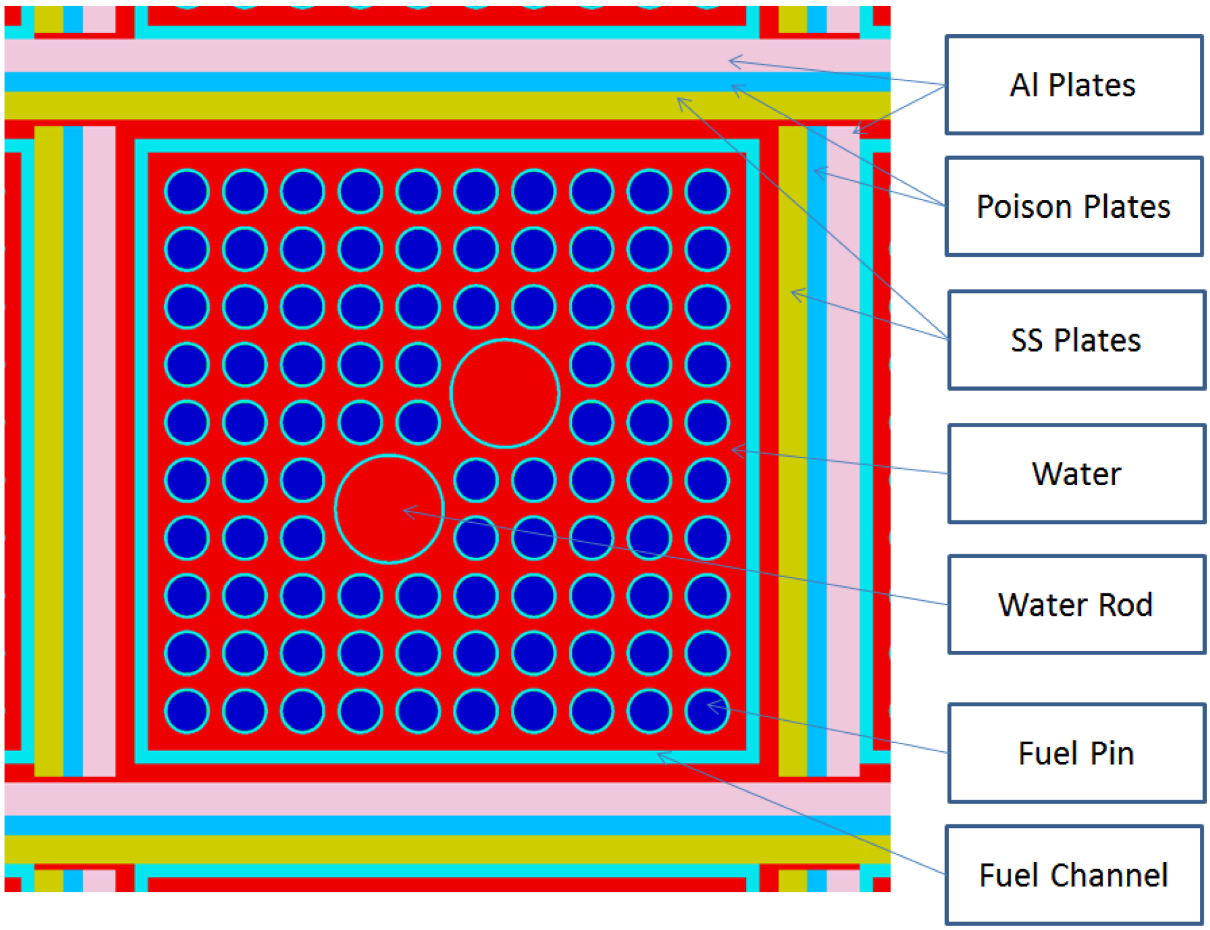

Figure 7-16
Radial Cross Section of the Fuel Compartment

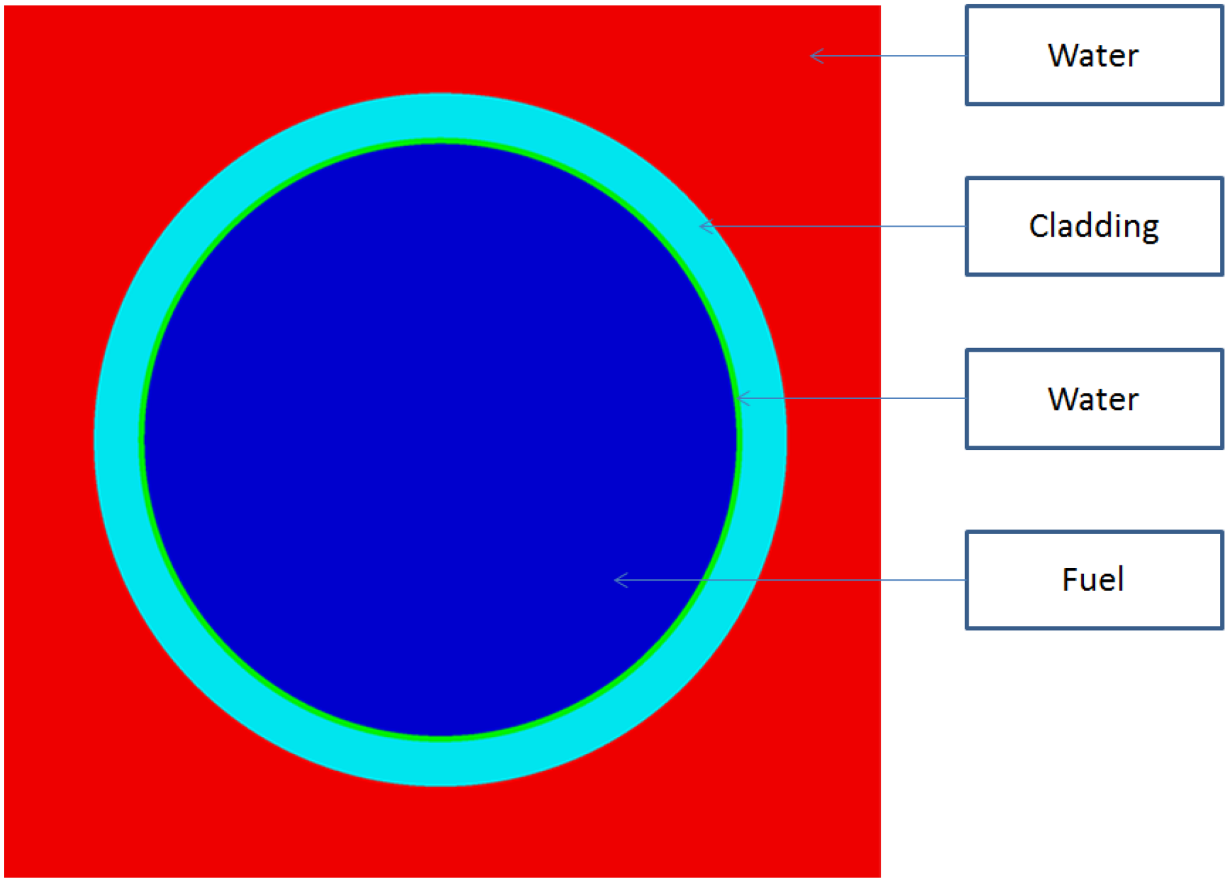

Figure 7-17
Radial Cross Section of the Fuel Pin

P r o p r i e t a r y I n f o r m a t i o n o n
W i t h h e l d P u r s u a n t t o 1 0

1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
2	2	2	2	2	4	2	2	2	2	2
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1

Model set:4x5x5
1=Fuel Rod
2,3,4 = Water Rod

5	1	1	1	1	3	1	1	1	1	5
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	5	3	5	1	1	1	1
2	2	2	2	2	4	2	2	2	2	2
1	1	1	1	5	3	5	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
5	1	1	1	1	3	1	1	1	1	5

Model set:4x5x5-2
1=Fuel Rod
2,3,4,5 = Water Rod

1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	5	3	5	1	1	1	1
2	2	2	2	2	4	2	2	2	2	2
1	1	1	1	5	3	5	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1
1	1	1	1	1	3	1	1	1	1	1

Model set:4x5x5-1, 4x5x5-1-sq
1=Fuel Rod
2,3,4,5 = Water Rod

1	1	1	1	3	1	1	1	1
1	1	1	1	3	1	1	1	1
1	1	1	1	3	1	1	1	1
1	1	1	1	3	1	1	1	1
2	2	2	2	4	2	2	2	2
1	1	1	1	3	1	1	1	1
1	1	1	1	3	1	1	1	1
1	1	1	1	3	1	1	1	1
1	1	1	1	3	1	1	1	1

Model set:4x4x4
1=Fuel Rod
2,3,4 = Water Rod

1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1

Model set:9x9-0w
1=Fuel Rod

1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1

Model set:8x8-0w
1=Fuel Rod

1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	2	1	1	1	1
1	1	1	2	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1

Model set:9x9-2w
1=Fuel Rod
2=Water Rod

1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	2	1	1	1	1	1
1	1	1	1	1	2	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1

Model set:10x10-2w
1=Fuel Rod
2=Water Rod

1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	2	2	2	1	1	1	1
1	1	1	2	2	2	1	1	1	1
1	1	1	2	2	2	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1

Model set:10x10-1w-sq
1=Fuel Rod
2=Water Rod

Figure 7-20
Fuel Assembly Layout
(Part 1 of 2)

1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	2	2	1	1	1	1
1	1	1	2	2	2	1	1	1
1	1	1	1	2	2	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1

Model set:9x9-2w-lg
1=Fuel Rod
2=Water Rod

1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	2	2	2	1	1	1
1	1	1	2	2	2	1	1	1
1	1	1	2	2	2	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1

Model set:9x9-1w-sq
1=Fuel Rod
2=Water Rod

1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	21	21	1	1	1
1	1	1	21	21	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1

Model set:8x8-4z
1=Fuel Rod
21=Zirc Rod

1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	2	2	1	1	1
1	1	1	2	2	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1

Model set:8x8-1w-lg
1=Fuel Rod
2=Water Rod

1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	22	23	1	1	1
1	1	1	23	22	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1

Model set:8x8-4w
1=Fuel Rod
22=Water Rod 1
23=Water Rod 2

1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	22	1	1	1	1
1	1	1	1	22	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1

Model set:8x8-2w
1=Fuel Rod
22=Water Rod

1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	22	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1

Model set:8x8-1w
1=Fuel Rod
22=Water Rod

1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	21	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1

Model set:7x7-1z
1=Fuel Rod
21=Zirc Rod

1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1
1	1	1	1	1	1	1

Model set:7x7-0w
1=Fuel Rod

Figure 7-20
Fuel Assembly Layout (Part 2 of 2)

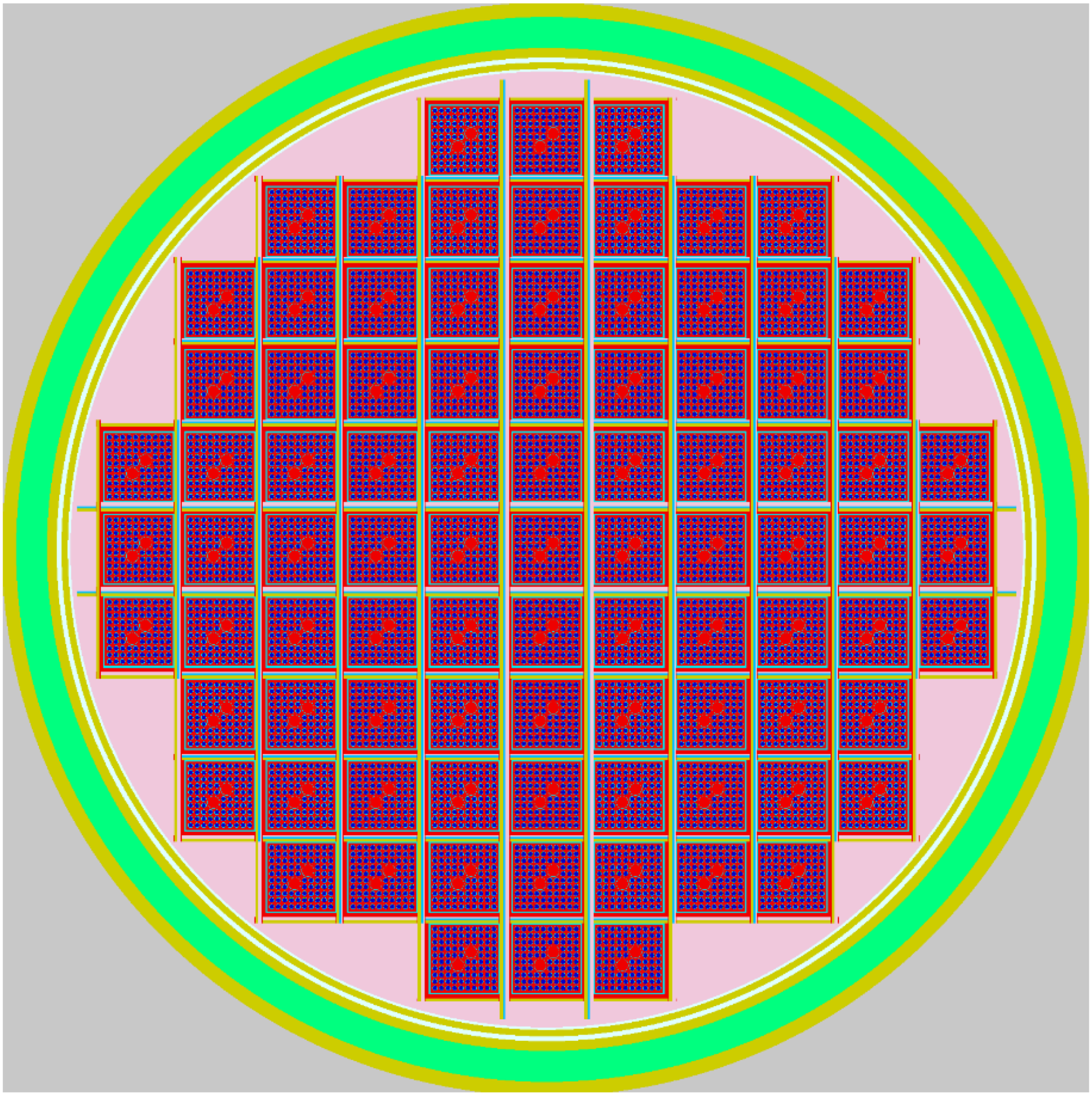

Figure 7-21
Radial Cross Section (Fuel Assemblies Placed Inwardly)

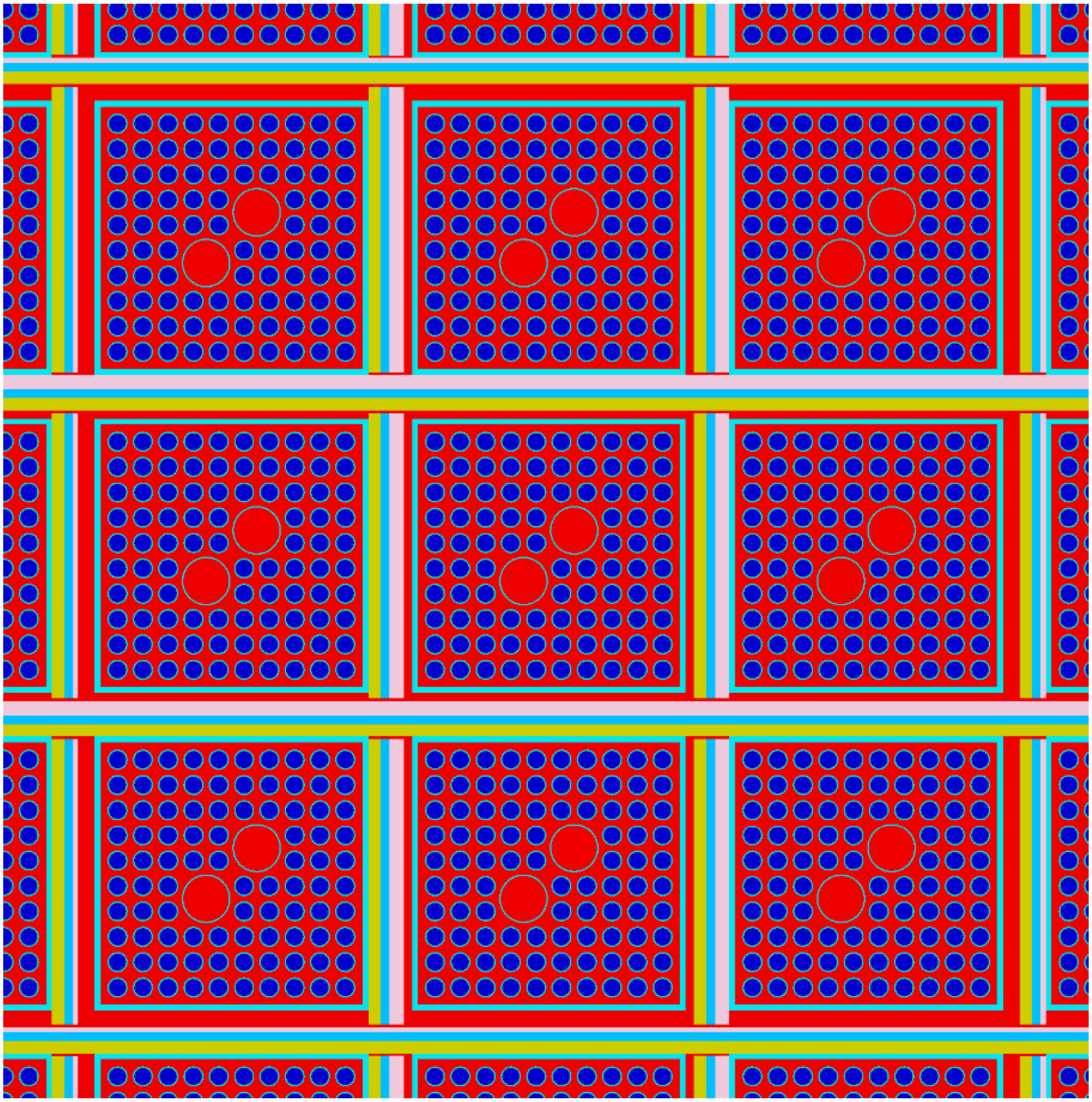

Figure 7-22
Radial Cross Section of Center 9 Fuel Compartments (Fuel Assemblies Placed Inwardly)

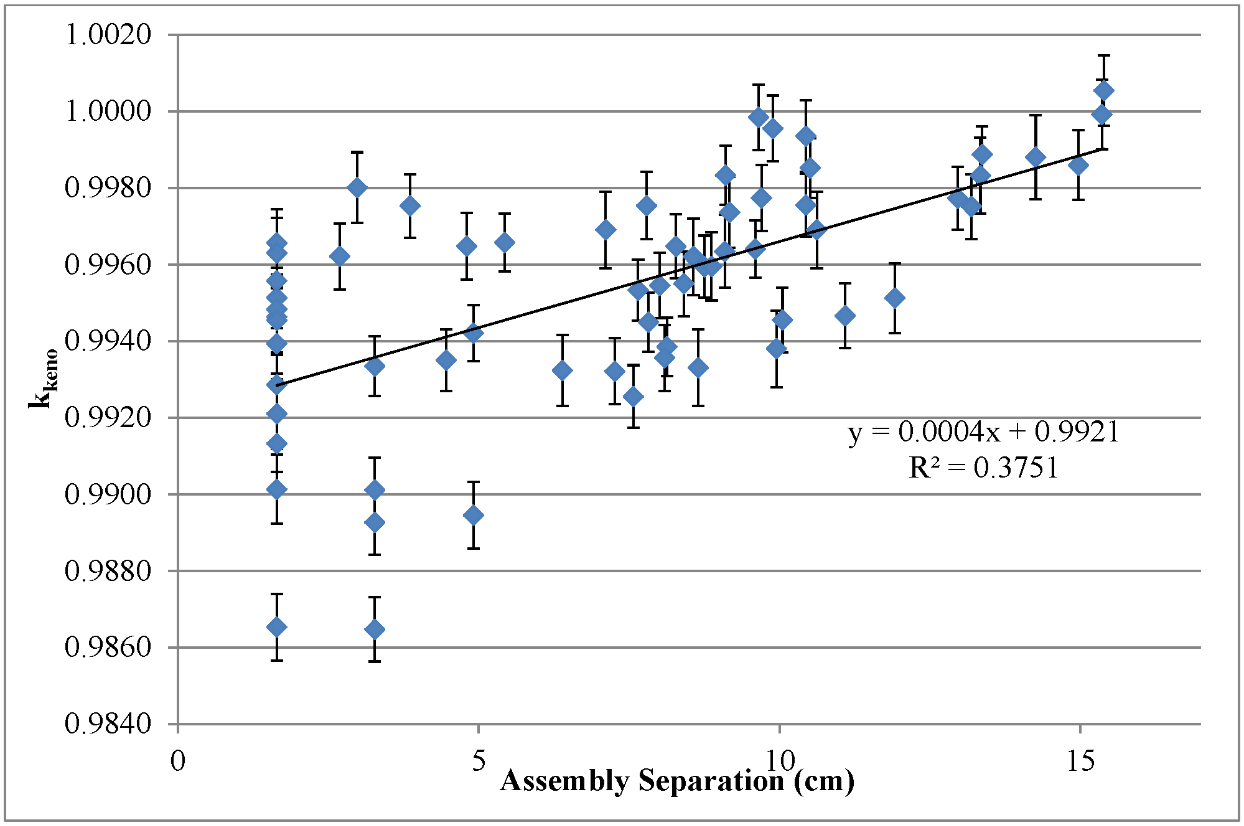

Figure 7-23
k_{KENO} versus Assembly Separation (cm) for Fresh Fuel Analysis

CHAPTER 8
MATERIALS EVALUATION

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8. MATERIALS EVALUATION

8.1 General Information

8.1.1 NUHOMS® EOS System Materials

This chapter provides the materials evaluation for the NUHOMS® EOS System in accordance with the guidance outlined in NUREG-1536, Revision 1 [8-1]. Steel materials employed in the various components of the NUHOMS® EOS System, particularly those that are relied on for structural integrity, are based on American Society for Testing and Materials (ASTM) and American Society of Mechanical Engineers (ASME) specifications. Horizontal storage module (HSM) concrete is based on American Concrete Institute (ACI) specifications. Neutron and gamma shielding materials are relied on for their nuclear properties and are not credited in the structural analysis. Similarly, the neutron absorber and the aluminum plates in the basket are relied on for their neutron absorption and thermal conductivity and are not credited in the structural analysis.

8.1.2 Environmental Conditions

The dry shielded canister (DSC) and HSM are exposed to the ambient weather conditions at the licensee site for the duration of the licensing period. Depending on the licensee local conditions, the environment may include chloride aerosols, precipitation, and freezing temperatures. The roof, front wall, door, and shield walls of the HSM concrete are directly exposed to the weather. The HSM side and rear walls, interior, and the DSC exterior surfaces are sheltered from direct effects of weather, though moisture and aerosols present in the air pass through the HSM interior via natural convention. Material temperatures of the storage system components are presented in Chapter 4.

During loading and unloading, the DSC is placed in the fuel pool, inside the transfer cask (TC). The annulus between the TC and DSC is filled with demineralized water and an inflatable seal is used to cover the annulus between the DSC and the cask. The exterior of the DSC is not exposed to pool water. The interior of the DSC and the exterior of the TC are exposed to either demineralized water (boiling water reactor (BWR)) or diluted boric acid (pressurized water reactor (PWR)). The TC and DSC are only kept in the spent fuel pool for a short period of time, typically less than 24 hours.

The radial neutron shield of the TC is filled with potable water. The removable neutron shield of the TC108 is not immersed in the fuel pool. It is installed onto the TC after the cask is removed from the pool.

During storage, the interior of the DSC is exposed to an inert helium environment. The DSC is vacuum dried and backfilled with helium after loading the fuel and welding the inner top cover plate.

8.1.3 Engineering Drawings

The drawings for the EOS-37PTH and EOS-89BTH DSCs, TCs, and HSMs are provided in Chapter 1, Section 1.3. The material specification, governing code, and quality category are specified in the parts list for each component.

8.2 Materials Selection

This section discusses the materials used in the main NUHOMS® EOS System components. Table 8-1 through Table 8-3 summarize the materials selected for the EOS-DSCs, -HSMs, and -TCs, respectively. Temperature-dependent mechanical and thermal properties for the materials listed in Table 8-1 through Table 8-3 are presented in Table 8-4 through Table 8-19 and in Table 8-20 through Table 8-22 for the main structural and non-structural materials, respectively. Table 8-23 and Table 8-24 present the temperature-dependent properties for reinforced concrete. The fuel cladding temperature-dependent properties are presented in Table 8-25 and Table 8-26. Properties of helium and air used in the thermal evaluations are presented in Table 8-27 and Table 8-28, respectively. Mechanical and thermal properties of the solid neutron shielding material and material compositions for the materials used in the shielding analysis Monte Carlo N-Particle (MCNP) models are shown in Table 8-29 through Table 8-31. The material properties used for the criticality analysis, including the minimum B-10 content used in the neutron poison plates criticality evaluations, are shown in Table 8-32 through Table 8-34. Emissivity requirements for the [ ] basket plates are provided in Table 8-35.

8.2.1 Applicable Codes and Standards and Alternatives

8.2.1.1 EOS-37PTH and EOS-89BTH DSC

The EOS-37PTH and EOS-89BTH DSC confinement boundary is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code [8-2], Section III, Division 1, Subsection NB and the alternative provisions to the ASME Code as described in Section 4.4.4 of the Technical Specifications (TS) [8-41].

The confinement boundary materials are ASME-approved for Class 1 Components, excepting duplex stainless steels SA-240 UNS S31803 and Type 2205. The higher chromium, molybdenum and nitrogen contents give these materials significantly improved resistance to localized corrosion including intergranular, pitting and crevice corrosion, and chloride-induced stress corrosion cracking (CISCC). Therefore, these materials are used when enhanced long-term resistance to CISCC is required. ASME Code Case N-635-1 [8-3], which has been endorsed by Regulatory Guide 1.84 [8-5], is used as a basis for including duplex stainless steels as alternate DSC confinement boundary materials.

For the DSC surfaces that are exposed to the ambient atmosphere, if the standard grade Type 304 or 316 is specified, the carbon content is limited to 0.03% in order to get the tensile strength of the standard grade, with the sensitization resistance of the 304L or 316L grade material.

The primary structural material for the basket is a high-strength low-alloy (HSLA) steel, which is used for fabrication of the fuel compartments that provide structural support to the fuel assemblies (FAs). Basket component stress intensity allowables used for evaluation of normal and off-normal conditions (ASME Code Service Level A and B) are developed based on the mechanical properties (S_u and S_y) listed in Table 8-10. A strain-based criterion is used for evaluation of the basket for accident conditions (Service Level D). Thus, the basket is regarded as a non-ASME Code component. Specification and acceptance testing of the HSLA steel is included in Chapter 10 and Section 4.3.2 of the TS.

The aluminum plates in the basket perform only a heat conducting function with no credit taken for their strength. The aluminum 6061 peripheral transition rails are entrapped between the fuel compartment structure and the DSC shell. For normal and off-normal loading conditions the primary stresses are limited to S_y. For accident conditions, qualification of the fuel compartment demonstrates that the rails perform their structural support safety function. The transition rails are specified as ASTM B221 Alloy 6061. The important-to-safety (ITS) Cat C rail fasteners are specified as ASTM A193 Gr B7 material.

The fixed neutron absorber plates are composed of boron carbide/aluminum metal matrix composite or BORAL® (EOS-89BTH DSC only). These materials perform no structural function. They are subject to AREVA specification and acceptance testing described in Chapter 10 and Section 4.3.1 of the TS.

8.2.1.2 EOS-TC Transfer Cask

The TC body is designed to the stress criteria of the ASME Code, Section III, Division 1, Subarticle NF-3200. The upper lifting trunnions *and trunnion welds* are designed in accordance with the ANSI N14.6 [8-6] stress allowables for a non-redundant lifting device. The TC neutron shields are designed to the stress criteria of Subarticle ND-3200.

The TC structural body is composed of carbon and low-alloy steel using ASME materials. The TC top cover plate (lid) may be made of aluminum or carbon steel using ASME materials. The trunnions are ASTM martensitic stainless steel. The radial neutron shell is carbon steel for the 125 and 135 ton TCs and aluminum for the removable neutron shield on the 108-ton TC. ASTM materials are used for the neutron shield shell.

The shielding materials in the TC perform no structural function, and therefore are not subject to any design code. The gamma shield is specified as ASTM B29 lead (any grade) and the axial bottom neutron shield as 5% boron high-density polyethylene.

8.2.1.3 EOS-HSM Horizontal Storage Module

The applicable codes for HSM(s) are:

- Concrete construction per ACI-318-08 [8-7].

- Concrete Design per ACI-349-06 [8-8].
- DSC support structure design per AISC Manual of Steel Construction [8-11].

Cement, aggregate, reinforcing steel, and DSC support structure steel conform to ASTM specifications.

The EOS-HSM concrete subcomponents are designed and constructed using a specified 28-day compressive strength of 5,000 psi, normal weight concrete. The cement is Type II or Type III Portland cement meeting the requirements of ASTM C150. The concrete aggregate meets the specifications of ASTM C33. The reinforcing steel is ASTM A615 or A706 Gr 60 deformed bars placed vertically and horizontally at each face of the walls, roof and floor.

The concrete surface temperature limits criteria are based on the provisions in Section 3.5.1.2 of NUREG-1536, as follows:

- If concrete temperatures in general or local areas are at or below 200 °F for normal/off-normal conditions/occurrences, no tests to prove capability at elevated temperatures or reduction of concrete strength are required.
- If concrete temperatures in general or local areas exceed 200 °F but do not exceed 300 °F, no tests to prove capability at elevated temperatures or reduction of concrete strength are required if the aggregates have a coefficient of thermal expansion (CTE) no greater than 6×10^{-6} in/in/°F, or are one of the following materials: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.

The above criteria in lieu of the ACI 349-06 requirements do not extend above 300 °F for normal/off-normal conditions and do not modify the ACI 349-06 requirements for accident conditions. Per E.4.2 of ACI 349-06 [8-8], the accident conditions or short-term period (i.e., blocked vent accident transient) concrete temperatures are limited to 350 °F. Higher temperatures are allowed per E.4.3 if tests are provided to evaluate the reduction in strength and this reduction is applied to design allowables. HSM concrete compressive tests are performed on specimens heated to or above that maximum accident temperature for no less than 40 hours. HSM concrete temperature testing is performed whenever there is a significant change in the cement, aggregate, or water-cement ratio of the concrete mix design. See Section 5.3 of the TS.

Alternatively, per the ACI 349-13 [8-26] commentary Section RE.4, the specified 28-day compressive strength can be increased to 7,000 psi for HSM fabrication, in lieu of the above aggregate types or CTE requirements, so that any losses in properties (e.g., compressive strength, modulus of elasticity) resulting from long-term thermal exposure will not affect the safety margins based on the specified 5,000 psi compressive strength used in the design calculations. Additionally, also as indicated in Section RE.4, short, randomly oriented steel fibers may be used to provide increased ductility, dynamic strength, toughness, tensile strength, and improved resistance to spalling. See Section 4.4.4 of the TS.

The EOS-DSC support structure is fabricated from ASTM A913 Gr 70 coated with an inorganic zinc-rich primer and a high build epoxy enamel finish. A corrosion allowance of 1/16 inch is used in the design calculations. Welding procedures are in accordance with ASME Code Section IX or AWS D1.1 [8-27].

At coastal sites with operational experience of corrosion due to atmospheric chlorides, the DSC support structure steel and weld filler metal have a minimum of 0.20% copper content. Weld material with 1% or more nickel is acceptable in lieu of 0.20% copper content. The copper content is equivalent to weathering steel [8-29], and nickel-bearing weld materials show equivalent corrosion resistance [8-30].

8.2.2 Material Properties

The material properties used in the NUHOMS® EOS System design analyses are listed in Table 8-4 through Table 8-35. Each table cites the source for the properties. Table 8-1 to Table 8-3 tie these materials to the individual components. Emissivity values for the thermal analysis are provided in Section 4.2.1(13).

8.2.2.1 EOS-37PTH and EOS-89BTH DSC

The structural material used in the baskets is a high strength low-alloy (HSLA) steel. The material properties shown in Table 8-10 are used in the structural analysis. If ASTM 829 Gr 4130 is used, AREVA test report [8-24] determines the optimum tempering for the desired toughness and the corresponding minimum yield and tensile strength. The A829 Gr 4130 steel plates are heat-treated and tempered per [ ] The requirements and acceptance criteria for HSLA steel are specified in Section 10.1.7.

The mechanical properties for the aluminum 6061 used for the basket transition rails are taken in the annealed (T0) condition to consider the effect of overaging at the service temperature near 400 °F. Therefore, the material may be supplied in any temper condition. Creep behavior of these rails is discussed in Section 8.2.6.

8.2.2.2 EOS-TC Transfer Cask

Material properties for the transfer cask body and fixed radial neutron shield shell are provided in Table 8-9 (SA-350 Gr LF3) and Table 8-11 (SA-516 Gr 70). Table 8-12 provides properties for the A182 Gr F6NM trunnion material. The mechanical properties at the welds and heat affected zones of the TC108's removable aluminum neutron shield are taken at the T0 condition per Table 8-18 to account for annealing due to welding heat. The balance of the shell is analyzed and specified at the T6 condition per Table 8-17.

The lead shielding is placed in the TC as bricks or sheet. The structural properties of lead used in the drop analysis are provided in Table 8-20(a) and Table 8-20(b). The effective thermal conductivity of the layered lead brick or sheet is calculated in Chapter 4 from the thermal conductivity of lead provided in Table 8-21.

8.2.2.3 EOS-HSM Horizontal Storage Module

In accordance with ACI 349-06, Section E.4.3, the strength properties of the concrete and reinforcing steel used in the HSM structural analysis are taken at the maximum calculated temperature. Temperature dependent mechanical properties of concrete and reinforcing steel are taken from [8-4] and presented in Table 8-23 and Table 8-24.

The material properties of the ASTM A913 Gr 70 steel used for the DSC support structure are listed in Table 8-15. The material properties used for the Type 304 stainless steel used for the heat shields are provided in Table 8-5.

8.2.2.4 NUHOMS® EOS System Materials Employed in the Shielding Analysis

Shielding properties of steel and concrete are obtained from [8-10] and are summarized in Table 8-30. Simple materials consisting of one element are not listed in Table 8-30. Such materials include lead, which is modeled at 11.18 g/cm^3, 98.6% of theoretical density of pure lead. Aluminum is used in the basket plates with a density of 2.7 g/cm^3. The metal matrix composite (MMC) poison is modeled as pure aluminum (no boron) with a density of 2.56 g/cm^3.

Borated polyethylene is used at the bottom of the EOS-TC for neutron shielding. Boron suppresses secondary gamma radiation from hydrogen capture of neutrons. The material is 5% boron by weight. The neutron shielding performance is more sensitive to hydrogen content than boron content, so the atom density of hydrogen is reduced by 15% to account for potential hydrogen loss due to aging, although the TC components are exposed to temperature and radiation only intermittently. The borated polyethylene composition used in the EOS-TC models is provided in Table 8-31.

Concrete used in the EOS-HSM is modeled without steel rebar at a density of 140 pcf (2.243 g/cm^3).

8.2.2.5 NUHOMS® EOS System Materials Employed in the Criticality Analysis

The Oak Ridge National Laboratory (ORNL) SCALE code package [8-15] contains a standard material data library for common elements, compounds, and mixtures. All materials used for the TC and canister analyses are available in this data library.

A complete list of all the relevant materials used for the criticality evaluation is provided in Table 8-33.

8.2.3 Materials for ISFSI Sites with Experience of Atmospheric Chloride Corrosion

As noted above, austenitic stainless steels for the DSC shell are procured with a maximum 0.03% carbon for reduced sensitization of the heat affected zones. Duplex stainless steels can be substituted as alternate materials when superior resistance to atmospheric chloride induced stress corrosion cracking is required. Fabrication specifications for duplex stainless steel DSCs shall use the latest edition of API 938-C [8-46] to establish requirements for material procurement, weld qualification, and weld process specifications in addition to those required by ASME Code Sections III and IX.

DSC support structures at sites with operational experience of corrosion caused by atmospheric chlorides are fabricated from steels equivalent to weathering steel.

8.2.4 Weld Design and Inspection

The primary confinement boundary consists of the DSC shell, the inner top cover plate, the inner bottom cover plate, the siphon and vent port covers, and the associated welds.

The confinement boundary welds made during fabrication of the DSC include the weld of the inner bottom cover plate to the shell and the circumferential and longitudinal seams of the shell. These welds are inspected (radiographic or ultrasonic inspection, and liquid penetrant inspection) in accordance with the requirements of Subsection NB of the ASME Code. The welds applied to the vent and siphon port covers and the inner top cover plate during closure operations define the confinement boundary at the top end of the DSC. These welds are examined by multi-level penetrant testing (PT) in accordance with NUREG 1536.

Both shop and field confinement boundary welds are pressure tested and leak tested as described in Chapter 10.

The welds of the TC structural body are designed to the stress limits for ASME Subsection NF for Class 1 supports. Weld inspections are performed by magnetic particle inspection (MT) as specified on the drawings in Chapter 1 with acceptance criteria of ASME Subarticle NF-5340.

The DSC support structure is bolted inside the HSM. The welds of the DSC support structure are designed in accordance with the Manual of Steel Construction [8-11], and visually inspected in accordance with AWS D1.1 with acceptance criteria for statically loaded non-tubular structures.

8.2.5 Galvanic and Corrosive Reactions

Potential sources of chemical or galvanic reactions are the interaction between the aluminum, neutron absorber, and HSLA steel while the DSC is immersed in the pool.

8.2.5.1 Behavior of Aluminum and Neutron Absorbers in Water and Boric Acid

The aluminum component of the MMC and BORAL® is a ductile metal having a high resistance to corrosion. Its corrosion resistance is provided by the buildup of a protective oxide film on the metal surface when exposed to a corrosive environment. As for aluminum, once a stable film develops, the corrosion process is arrested at the surface of the metal. The film remains stable over a pH range of 4.5 to 8.5. There are no chemical, galvanic, or other reactions that could reduce the areal density of boron in the neutron poison plates with either of the poison plate materials.

The pores in the core material exposed at the edges of BORAL® can retain water, which can cause delamination of the skin from the core during drying or storage. This has been evaluated and determined to have no effect on the criticality control function of BORAL® [8-31]. Metal matrix composites are tested to verify they will not be subject to this phenomenon.

The period of immersion is insufficient to cause significant localized corrosion such as pitting or crevice corrosion in the aluminum.

8.2.5.2 Behavior of Stainless Steel in Deionized Water and Weak Boric Acid

The DSC shell and cover plates are made from Type 304 or 316 or duplex stainless steels. Stainless steel does not exhibit general corrosion when immersed in deionized water. Reference [8-43] reports testing type 304 stainless steel for corrosion in saturated boric acid at 70 °F (5% boric acid, 8750 ppm boron) and 140 °F (13% boric acid, 22750 ppm boron). At 70 °F, there was no measureable corrosion, and at 140 °F, corrosion was measured at 7x10-4 inch/year (0.018 mm/year) for consumable electrode welds, and no measureable corrosion for other weld and plate conditions. Typical conditions for pool during loading of the EOS DSC would be up to 3000 ppm boron, water temperature <140 °F in the pool increasing until the time of draining, and duration usually <72 hours, including both immersion in the pool and the time until draining and drying the DSC. Considering the short time and the low boric acid concentration compared to the testing, stainless steel type 304 would show no measureable corrosion, and duplex 2205 would have less corrosion due to its higher chromium content and its molybdenum.

Under PWR reactor operating conditions, stress corrosion cracking has occurred in piping containing stagnant, high concentration boric acid, and stainless steel cladding has cracked [8-44], but the time, temperature, and concentration conditions described in the foregoing paragraph are insufficient to initiate stress corrosion cracking in the stainless steel shell during DSC loading. Galvanic corrosion could occur at contact between the basket perimeter's aluminum rails and the stainless steel DSC shell, with the aluminum corroding sacrificially, but this is mitigated by the passivity of the aluminum and the stainless steel in the short time the pool water is in the DSC. Also, the low conductivity of the pool water tends to minimize galvanic reactions.

8.2.5.3 Behavior of Low-Alloy Steel in Deionized Water and Weak Boric Acid

EPRI-1000975, Boric Acid Corrosion Guidebook, Revision 1 [8-42] provides available boric acid corrosion test data in aerated and deaerated water at various temperatures and soluble boron concentrations.

As noted in Section 8.1.2, the TC and DSC are only kept in the spent fuel pool for a short period of time, typically less than 24 hours. The spent fuel pool temperature is controlled during the loading operation and the pool water remains open to the atmosphere during loading. As noted in NUREG-1536 [8-1], Section 4.5.3, the maximum temperature limit of a spent fuel pool is typically 46 °C (115 °F).

The soluble boron concentration required for loading of PWR fuel assemblies in the NUHOMS® EOS system varies between 2,000 and 2,500 ppm as shown in Chapter 7, Table 7-3.

EPRI-1000975 [8-42] Figure 4-3 and Table 4-3 provide a summary of the available boric acid corrosion test data. The relevant corrosion rates retrieved from Figure 4-3 and Table 4-3 of the EPRI report are summarized in Table 8-36.

The highest corrosion rate among the relevant data shown in Table 8-36 is 0.015 in/yr (0.38 mm/yr). To account for uncertainties related to an initially high corrosion rate during a shorter period of time and assuming a loading time of 100 hours (approximately four times larger than the typical period),the thickness reduction is 0.00017 inch (4.3x10^{-3} mm) based on a maximum corrosion rate of 0.015 in/yr. The evaluated thickness reduction is much lower than the standard plate tolerance of ±0.02 inch. Therefore, the amount of the evaluated thickness reduction would have no effect on the structural performance.

The basket's steel plates are [

] will provide short-term corrosion protection, sufficient for the manufacturing process and short-term immersion in the pool. It can be expected that a small amount of rust will form, but this will be insufficient to affect the performance of design functions or to cause turbidity in pool water during loading operations.

During storage, the interior of the DSC is exposed to an inert helium environment. The helium environment does not support the occurrence of chemical or galvanic reactions because both moisture and oxygen must be present for a reaction to occur.

8.2.5.4 Lubricants and Cleaning Agents

The DSC is cleaned in accordance with approved procedures to remove cleaning residues prior to shipment to the storage site. The basket is also cleaned prior to installation in the DSC. Decontamination agents may be applied on the TC surfaces at each use. A graphite-based dry lubricant is applied to the surface of the Nitronic 60 sliding surfaces in the TC and on the HSM's DSC support structure. Lubricants may be used on the TC cover bolts and on the trunnion and lower rotating socket bearing surfaces. The cleaning agents have no adverse effect on the DSC materials and their design functions. Lubricants used on bolt threads and trunnions have no adverse effect on materials or design functions.

The graphite lubricant used on the DSC support structure rails is noble relative to the rail face and to the canister shell; therefore, it could induce galvanic corrosion of DSC shell and Nitronic 60 support rail. While galvanic corrosion is possible under the correct conditions, the conditions of dry storage in NUHOMS are very unlikely to result in loss of materials due to galvanic corrosion at the rail-DSC interface based on the results of galvanic corrosion tests for stainless steel Type 304 coupled with graphite in reference [8-45]. The corrosion rate of the stainless steel Type 304 was reported to be "nil," and no pitting was observed with a cathode-to-anode ratio of about unity in a 3.5 percent NaCl solution. The dry conditions and high anode-to-cathode ratio in the NUHOMS® system would result in an even lower corrosion rate than these experimental conditions. Unlike failures of high strength martensitic stainless steel valve components in reactor operation [8-44], there is no operating experience for galvanic corrosion of dry graphite lubricants on stainless steel in dry conditions. Therefore, the graphite lubricant does not have an adverse effect on materials or design functions.

8.2.5.5 Corrosion of Canister Shell During Storage

The DSC external surfaces are protected from direct exposure to precipitation, and are exposed only to the humidity and aerosols in the cooling air that flows through the HSM. The stainless steel outer surfaces and the rails surfaces upon which the DSC rests are not subject to general corrosion. Sites near the coast or other sources of chloride aerosols could experience pitting, stress corrosion cracking, or crevice corrosion [8-39] during long term service, but neither the accumulation of deposits nor the canister surface temperatures are expected to be sufficiently aggressive conditions of chloride concentration and deliquescence required for these corrosion mechanisms during the initial 20 year license period.

For sites near chloride aerosol sources, the NUHOMS® EOS System provides options for canisters made of duplex stainless steel, which, due to its partially ferritic grain structure, is very resistant to localized corrosion of pitting and crevice corrosion and stress corrosion cracking. An option for inspection port on the front of the HSM provides for improved aging management beyond the 20-year initial license period.

8.2.5.6 Corrosion of DSC Support Structure

The DSC support structure is protected from direct exposure to precipitation, and is exposed only to the humidity and aerosols in the cooling air that flows through the HSM. Epoxy enamels such as Carboguard® 890 are suitable for continuous service to 300 °F, while inorganic zinc primers such as Carbozinc 11 have much higher temperature resistance. The maximum temperature on the top of the support beam is about 270 °F per Figure 4-12. The top coat is expected to experience chalking and other effects of radiation over 10^6 rad, but the inorganic primer coat is insensitive to radiation. Inspections for license extension [8-32, 8-33] have found only minor local rusting. Nonetheless, the stress analysis removes 1/16 inch from all surfaces to account for corrosion. At ISFSIs with operational experience of corrosion with atmospheric chlorides, additional protection is provided by specifying a minimum 0.2% copper content, which results in an adherent self-protecting oxide layer equivalent to weathering steel [8-29].

8.2.5.7 Corrosion of Transfer Cask

Protective coatings, periodic inspection, and maintenance as specified by Section 10.2.1 prevent corrosion of the TC from affecting design functions.

8.2.6 Creep Behavior of Aluminum

The structural analysis of the EOS-37PTH and EOS-89BTH DSC described in Chapter 3 does not credit the neutron poison material to demonstrate mechanical integrity. For the steel components, creep is not a concern due to the operating temperature and mechanical and thermal properties of materials.

From [8-28], the allowable bearing stresses in the basket aluminum components, to limit creep strain to 0.01 in 550,000 hours, are as follows:

- 0.254 ksi in the hottest aluminum plate, with a starting temperature of 680 °F.
- 0.758 ksi in the hottest R90 rail, with a starting temperature of 470 °F.
- 0.876 ksi in a less than hottest R90 rail, based on a starting temperature of 440 °F.

Although 550,000 hours is approximately 63 years, the creep strain time curve is very flat at 550,000 hours, such that the change in allowable bearing stress for 80 years is insignificant.

In addition, from Chapter 4, for normal conditions (applicable to long-term storage conditions) at the hottest cross-section of the EOS-37PTH and EOS-89BTH baskets, the average R90 transition rail temperature is not more than 469 °F, which is less than the above temperature of 470 °F for the hottest R90 rail. Similarly, from Chapter 4, for normal conditions, the hottest basket plate temperature is not more than 676 °F, which is less than the above temperature of 680 °F for the hottest aluminum plate. Based on this favorable comparison of starting temperatures, and since the heat dissipation rate of the EOS-37PTH and EOS-89BTH baskets is better than the temperature data for the baskets evaluated in [8-28] (i.e. more favorable temperature versus time values), the allowable creep stresses given above are applicable to the aluminum components of the EOS-37PTH and EOS-89BTH baskets.

8.2.7 Bolt Applications

There are no bolts in the NUHOMS® EOS System associated with confinement, and no bolts performing quality category A functions. Bolts that perform ITS Category B and C functions are described here. No specific preload of any bolt is required by the design analysis. Therefore, all bolts are installed snug tight without a torque specification except as noted below.

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The DSC support structure inside the HSM uses SA-193 Gr B7 bolts. The heat shields are fastened to the HSM base and roof with SA-193 Gr B7 bolts. The roof, door and wall assemblies of the HSM use SA-193 Gr B7 bolts. These bolts are zinc-coated for corrosion resistance.

The top cover and bottom ram access port covers of the TC cask are retained by SA-540 Gr B23 bolts. The bolts may be plated or coated for corrosion protection. Torque values on the drawings in Chapter 1 are recommendations for assembly, not requirements based on the design analysis.

8.2.8 Protective Coatings and Surface Treatments

No coatings are applied to the DSC surface. The top shield plug of the DSCs is coated with an electroless nickel [

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The exposed carbon steel surfaces of the EOS-TCs are coated with a painting system suitable for spent fuel pool immersion, and withstanding long-term exposure to the elevated temperatures of the TC. Stainless steel surfaces, that is, trunnions, sliding rails, and the stainless steel overlay for the ram access port sealing surfaces are not coated. The removable aluminum neutron shield shell for the TC108 is painted only on the outer diameter. The following finish enamels are used on the transfer cask:

- PPG Amerishield™ enamel or Carboline Carboguard® 890, color white, is used for the EOS-TC exterior surfaces.
- PPG Amerishield™, color white, is used for coating the exterior of the removable EOS-TC108 neutron shield. This neutron shield is not immersed.
- Carboline Thermaline® 450-EP PPG or Amercoat® 91 is used for coating the EOS-TC interior surfaces, which are exposed to higher service temperatures up to 373 °F (Tables 4-26 and 4-27).

Manufacturer's recommendations are followed for surface preparation, primer coat selection, and coating application.

Alternate coatings that are accepted by licensees for spent fuel pool immersion, and whose short-term service temperature is above the normal condition TC surface temperatures may be used. For solar absorptivity, white color must be maintained where specified.

The DSC support structure in the HSM is coated with an inorganic zinc-rich primer and a high build epoxy enamel finish, for example, Carboline Carbozinc® 11 primer with Carboguard® 890 enamel. Embedments and fasteners are galvanized.

Coatings are not important to safety.

8.2.9 Neutron Shielding Materials

During storage, all neutron shielding is provided by the concrete of the HSM. No polymeric neutron absorbers are used. Boron is not added to the concrete.

During transfer of the DSC from the pool to the HSM, neutron shielding is provided by water in a radial neutron shielding jacket of the TC. The bottom end and lid of the TC include a layer of solid neutron shielding consisting of borated high-density polyethylene, Quadrant Borotron® types HD050 [8-9], UH050, and HM050 or similar material.

Because of the short duration of the transfer operations, this material is not subjected to significant thermal or radiation-induced degradation. The shielding analysis uses a reduced hydrogen content to bound any degradation as discussed in Section 8.2.2.4.

8.2.10 Materials for Criticality Control

The EOS-37PTH uses an Aluminum/B4C MMC poison plate material, which is suitable for long-term use in radiation and thermal environments of a dry cask storage system. The NUHOMS® EOS-37PTH DSC has two neutron poison loading options A and B that correspond to minimum B-10 loadings in mg B-10/cm^2, as shown in Table 8-32. There are three neutron poison loading options specified for the EOS-89BTH corresponding to the type and boron content of the poison plates. The baskets manufactured with MMC are designated as A and B, while the basket manufactured with BORAL® plates is designated as C, as shown in Table 8-32. In criticality evaluations, credit is taken only for 90% of B-10 areal density in the MMC and 75% in the BORAL® poison plates.

The control method used to prevent criticality is incorporation of neutron absorber material in the basket material and favorable geometry. The quantity and distribution of boron in the poison material is controlled by specific manufacturing and acceptance criteria of the poison plates.

Chapter 10 provides detailed information of the material specifications, qualification, and acceptance testing for the neutron absorbing materials. The essential requirements are included in Chapter 13.

8.2.11 Concrete and Reinforcing Steel

The concrete and reinforcing steel are described in Section 8.2.2.

8.2.12 Seals

The DSCs do not employ mechanical seals to demonstrate the integrity of the confinement boundary.

The only mechanical seal in the storage system is the elastomer o-ring for the bottom ram access penetration closure plate of the TC, which isolates the DSC-TC annulus water from the pool water during DSC loading in the fuel pool.

8.2.13 Low Temperature Ductility of Ferritic Steels

8.2.13.1 DSC Basket

The materials test report [8-24] evaluates several tempering cycles for tensile properties and fracture toughness. In accordance with the recommendation of that report and Regulatory Guide 7.11 Table 4 [8-40], the material is subject to dynamic tear testing as specified in Section 10.1.7, and on the basket parts list drawings in Chapter 1. No welding is performed on this material.

8.2.13.2 Transfer Cask

The materials in the lifting load path of the TC are the SA-350 Gr LF3 top and bottom rings, SA-516 Gr 70 inner and outer structural shells and bottom plate, and ASTM A182 Gr F6NM martensitic stainless trunnions. The minimum service (material) temperature of the TC is 0 °F. The SA-516 Gr 70 material and the procedures for all welds in the load path are subject to Charpy impact testing at 0 °F in accordance with ASME Code Section III, Article NF-2300 for Class 1 supports. SA-350 Gr LF3 is a cryogenic steel that has been used on AREVA Inc. metal casks and has consistently demonstrated nil ductility transition temperatures below -80 °F. Therefore, this material need not be impact tested for the TC application. The trunnion materials, a martensitic stainless, are subject to drop weight testing at -40 °F per ANSI N14.6 or to Charpy impact testing at 0 °F in accordance with ASME Code Section III, Article NF-2300 for Class 1 supports. More recent editions of the ASTM test specifications than those invoked by ANSI N14.6 may be used.

8.3 Fuel Cladding

8.3.1 Fuel Burnup

The limit for fuel burnup is 62 GWd/MTU.

8.3.2 Cladding Temperature Limits

The thermal design criteria for cladding integrity are described in Chapter .4, Section 4.2. The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short-term fuel loading and transfer operations including vacuum drying and helium backfilling of the EOS-37PTH and EOS-89BTH DSC per NUREG 1536. In addition, NUREG 1536 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65 °C (117 °F) during drying and backfilling operations. A maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off- normal thermal transients.

In addition, Chapter 4, provides the evaluation of thermal cycling during loading and vacuum drying operations. The thermal analysis of the EOS37PTH and EOS-89BTH DSC during the blowdown (draining) operation assumes helium is used to drain the water from the EOS-37PTH and EOS-89BTH DSC cavity and subsequent vacuum drying occurs with a helium environment. Further, the water level in the annulus between the DSC and TC is monitored and replenished, if needed, during vacuum drying. This configuration provides adequate cooling to eliminate the thermal cycling of fuel cladding during helium backfilling of the EOS-37PTH and EOS-89BTH DSC subsequent to vacuum drying and to eliminate the need for a time limit on the vacuum drying operation. Further, the water level in the annulus between the DSC and TC is monitored and replenished, if needed, during vacuum drying. The maximum fuel cladding temperature limit of 400 °C (752 °F) in NUREG 1536 is satisfied for the EOS-37PTH and EOS89-BTH DSC.

8.4 Prevention of Oxidation Damage During Loading of Fuel

The operations described in Chapter 9 require that the canister is filled with helium as the water is pumped below the top of the fuel rods. Subsequent operations alternate evacuation and helium backfill. The use of helium will prevent oxidation of fuel cladding. The final condition of helium purity for storage is controlled by the vacuum drying acceptance criteria in Chapter 9 and Section 3.1.1 of the TS.

8.5 Flammable Gas Generation

Experience with aluminum in dry storage indicates that it will generate some hydrogen as the surface oxide layer is formed. The reaction is more pronounced in pure water (BWR pools) than in PWR pools. The general corrosion of aluminum and aluminum-based neutron absorbers is self-limiting, and does not reduce the integrity of the materials. [

] Galvanic coupling between aluminum and stainless steel has not proven to be a problem in many years of design with these materials. Aluminum and carbon steel are closer on the galvanic scale than aluminum and stainless steel. The hydrogen generation is monitored and controlled prior and during welding operations in accordance with the operations instructions in Chapter 9.

8.6 DSC Closure Weld Testing

The field closure weld of the inner top cover plate to the shell is pressure tested as described in Section 10.1.1.1. The confinement boundary field welds are examined by root and final PT, and the structural weld of the outer top cover plate to the shell is examined by progressive PT as shown on the drawings in Chapter 1, and as described in the Code Alternatives, Section 4 of the Technical Specifications. PT acceptance criteria are those of ASME Code, Section III, Subsection NB, Subarticle 5350.

The weld between the DSC shell and inner top cover and the siphon and vent cover welds are leak tested to an acceptance criterion of 1×10^{-7} ref cm^3/s at the field after the FAs are loaded in the canister. The testing is described in more detail in Chapter 10.

8.6.1 Periodic Inspections

The NUHOMS® EOS System is designed to be totally passive with minimal maintenance requirements. During fuel storage, the system requires only periodic inspection of the air inlets and outlets to ensure that no blockage has occurred, unless a means of thermal performance monitoring is employed.

The TC is designed to require only minimal maintenance. Transfer cask maintenance is limited to periodic inspection of critical components, inspection and repair of coatings, and replacement of damaged or nonfunctioning components.

8.7 References

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8-46 API Technical Report 938-C, "Use of Duplex Stainless Steels in the Oil Refining Industry," Second Edition, American Petroleum Institute, April 2011.

Table 8-1
DSC Materials

Shell Assembly Subcomponents	**Material**
Cylindrical Shell, Inner Top Cover Plate, Inner Bottom Cover Plate (Confinement Boundary), Outer Top Cover Plate	Stainless steel ASME SA-240 Type 304, 316, 2205 or UNS S31803
Drain Port Cover Plate (Confinement Boundary)	ASME SA-240 Type 304
Vent Port Plug (Confinement Boundary)	ASME SA-240 Type 304 or SA-479 Type 304
Outer Bottom Cover Plate, Lifting Lug, Lifting Lug Plate	Stainless steel ASTM A240 Type 304, 316, or 2205 or UNS S31803
Grapple Ring, Grapple Ring Support	Stainless steel ASTM A240 Type 304, 316, or 2205 or UNS S31803, or ASTM A182 Gr F304, F316, F51, or F60
Siphon Bracket, Basket Key	Stainless steel ASTM A240 Type 304 or 316
Test Port Plug	ASTM A240, A276, or A479, Type 304, 316, UNS S31803, or UNS S32205
Top and Bottom Shield Plugs	Carbon steel ASTM A36
Miscellaneous parts (Drain Tube, Siphon Block, Port Adapter, Quick Connect, Reducing Bushing)	Stainless Steel
Basket Assembly Subcomponents	**Material**
Basket Steel Plates	Low-alloy high strength steel such as ASTM A829 Gr 4130
Basket Aluminum Plates	Aluminum ASTM B209 Alloy 1100
Transition Rails: R90 and R45 Aluminum Extrusions R45 Angle Plates Tie Rod/Flat Head Screws Hex Nuts Miscellaneous Washers	 ASTM B221 Alloy 6061 ASTM A516 Gr 70 ASTM A193 Gr B7 ASTM A194 Gr 7 Carbon Steel
Neutron absorber plates	MMC or BORAL® (89BTH only)

Table 8-2
HSM Materials

HSM Subcomponents	Material
HSM walls, roof, floor, end shield walls, rear shield walls	Reinforced concrete with ASTM A615 or A706 Gr 60 reinforcing steel
DSC Support Structure Assembly	ASTM A913 Gr 70
Sliding Rail	Nitronic® 60 stainless steel, ASTM A240 UNS S21800
HSM Door Door Steel Liner Assembly Threaded Inserts Inspection Penetration Sleeve Door	Reinforced concrete Steel Steel Stainless Steel
Axial retainer assembly: Tube Steel Miscellaneous (plate embedment, DSC axial retainer)	 ASTM A500 Carbon steel
HSM Heat Shields	Stainless steel ASTM A240 Type 304 or 316
HSM Roof Attachment Angles and Stiffener Plates	Carbon steel
HSM Outlet Vent Cover Outlet Vent Cover Steel Liner HSM Inlet Vent Screen Assembly	Reinforced concrete ASTM A36 Carbon Steel
Bird Screens and Dose Reduction Hardware	Stainless steel
Wind deflectors	Aluminum
Segmented HSM Connecting Hardware	ASTM A722 Gr 150
Fasteners: Bolts Washers Nuts	 ASTM A193 Gr B7/ A325/A563/A490/A108 ASTM A36/F436/F844/ Stainless Steel ASTM A194/A563/A194/ Carbon Steel
Threaded Embedments: Stud Bolt Sleeve Nut Nut	 ASTM A193-B8 CL 2 or ASTM A193-B8M CL 2 ASTM A194 Gr 2H or A563 Gr A ASTM A194 Gr 8M or A563 Gr A

Table 8-3
Transfer Cask Materials

TC Subcomponents	Material
Top Cover Plate	ASME SA-516 GR 70
Top Cover Plate	ASTM B209 Alloy 6061- T6
Bottom Cover Plate	ASTM A516 Gr 70 or
Top and Bottom Rings	ASME SA-350 Gr LF3
Inner and Outer Shells	ASME SA-516 Gr 70
Bottom End Plate	ASME SA-516 Gr 70
RAM Access Penetration Ring	ASME SA-516 Gr 70
Wedge Plates	ASTM A516 Gr 70
Upper Trunnion	ASTM A182 Gr F6NM
Gamma Shielding	ASTM B29
Neutron Shielding Side Bottom and Top Cover	 Water HDPE 5%B
Neutron Shield Panels (NSP), NSP Top and Bottom Support Rings, Bottom Neutron Shield Plate Outer Panel (TC 125 and 135)	ASTM A516 GR 70
Neutron Shield I Beams (TC 125 and 135)	ASTM A36
Removable Neutron Shield Assembly (TC 108 only)	ASTM B209 Alloy 6061-T6
Canister Rails	ASTM A240 UNS S21800 (Nitronic® 60)
Cask Bottom Cover Plate O-Ring	Ethylene Propylene
Inlet-Outlet Covers	ASTM A513 Type 1020
Threaded Fasteners: Cask Top Cover Closure Bolts Bottom Cover Plate Closure Screws	 SA540 GR B23 Cl 1 SA540 GR B23 Cl 1

Table 8-4
Material Properties, SA-36

Temp (°F)	E (103 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} (10-6 °F^{-1})	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20		19.3	36.0	58.0		0.280		
70	29.4	19.3	36.0	58.0	6.4		34.9	0.103
100		19.3	36.0	58.0	6.5		34.7	0.106
150			33.8		6.6		34.2	0.110
200	28.8	19.3	33.0	58.0	6.7		33.7	0.114
250			32.4		6.8		33.0	0.117
300	28.3	19.3	31.8	58.0	6.9		32.3	0.119
Xm 350					7.0		31.6	0.122
400	27.9	19.3	30.8	58.0	7.1		30.9	0.124
450					7.2		30.1	0.126
500	27.3	19.3	29.3	58.0	7.3		29.4	0.128
550					7.3		28.7	0.131
600	26.5	18.4	27.6	58.0	7.4		28.0	0.134
650		17.8	26.7	58.0	7.5		27.3	0.136
700	25.5	17.3	25.8	58.0	7.6		26.6	0.140
750			24.9	57.3	7.7		26.0	0.143
800	24.2		24.1	53.3	7.8		25.3	0.147
850			23.4	48.5	7.9		24.6	0.151
900	22.5		22.8	43.3	7.9		23.8	0.155
950			22.1	38.0	8.0		23.1	0.159
1000	20.4		21.4	33.4	8.1		22.4	0.164
ASME	Table TM-1for Carbon Steel with C≤0.3% p. 738	Table 2A Line 20 p. 270	Table Y-1 Line 2 pp. 542-543	Table U Line 46 pp. 460-461	Table TE-1 p. 708 Group 1	Table PRD p.744	Calculated based on Table TCD p. 726, Group A	

Source: ASME Section II, Part D

Table 8-5
Material Properties, SA-240 Type 304 / SA-182 Gr F304 (≤ 5” thk.)

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} ($10^{-6}\ {}^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20		20.0	30.0	75.0		0.290		
70	28.3	20.0	30.0	75.0	8.5		8.6	0.114
100		20.0	30.0	75.0	8.6		8.7	0.114
150			26.7		8.8		9.0	0.117
200	27.5	20.0	25.0	71.0	8.9		9.3	0.119
250			23.6		9.1		9.6	0.121
300	27.0	20.0	22.4	66.2	9.2		9.8	0.122
350					9.4		10.1	0.124
400	26.4	18.6	20.7	64.0	9.5		10.4	0.126
450					9.6		10.6	0.127
500	25.9	17.5	19.4	63.4	9.7		10.9	0.129
550					9.8		11.1	0.129
600	25.3	16.6	18.4	63.4	9.9		11.3	0.130
650		16.2	18.0	63.4	9.9		11.6	0.131
700	24.8	15.8	17.6	63.4	10.0		11.8	0.132
750		15.5	17.2	63.3	10.0		12.0	0.132
800	24.1	15.2	16.9	62.8	10.1		12.3	0.133
850			16.5	62.0	10.2		12.5	0.134
900	23.5		16.2	60.8	10.2		12.7	0.134
950			15.9	59.3	10.3		12.9	0.135
1000	22.8		15.5	57.4	10.3		13.1	0.135
ASME	Table TM-1, Group G p. 738	Table 2A Lines 17, 19, 26 p. 306	Table Y-1 Lines 32, 38 pp. 610-611, Line 10 p. 612	Table U Lines 16,22,39 pp. 492-493	Table TE-1 p. 711 Group 3	Table PRD p. 744	Calculated based on Table TCD p. 727, Group J	

Source: ASME Section II, Part D

Note: Thermal conductivity is provided for information only; the design analysis is based on the thermal conductivity of SA-240 Type 316, which is lower than Type 304.

Table 8-6
Material Properties, SA-240 Type 316 / SA-182 Gr F316 (≤ 5" thk.)

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} (10^{-6} $^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20		20.0	30.0	75.0				
70	28.3	20.0	30.0	75.0	8.5		8.2	0.118
100		20.0	30.0	75.0	8.6		8.3	0.118
150			27.4		8.8		8.6	0.121
200	27.5	20.0	25.9	75.0	8.9		8.8	0.121
250			24.6		9.1		9.1	0.124
300	27.0	20.0	23.4	72.9	9.2		9.3	0.124
350					9.4		9.5	0.125
400	26.4	19.3	21.4	71.9	9.5		9.8	0.126
450					9.6		10.0	0.127
500	25.9	18.0	20.0	71.8	9.7	0.290	10.2	0.127
550					9.8		10.5	0.129
600	25.3	17.0	18.9	71.8	9.9		10.7	0.129
650		16.6	18.5	71.8	9.9		10.9	0.130
700	24.8	16.3	18.2	71.8	10.0		11.2	0.131
750		16.1	17.9	71.5	10.0		11.4	0.132
800	24.1	15.9	17.7	70.8	10.1		11.6	0.132
850			17.5	69.7	10.2		11.9	0.134
900	23.5		17.3	68.3	10.2		12.1	0.134
950			17.1	66.5	10.3		12.3	0.135
1000	22.8		17.0	64.3	10.3		12.5	0.136
ASME	Table TM-1 p. 738 Group G	Table 2A, p. 298 Lines 24, 26, and 33	Table Y-1, pp. 602-603 Lines 7, 10, and 19	Table U, p. 487, Lines 43 and 46; p. 489 Line 9	Table TE-1 p. 711 Group 3	Table PRD p. 744	Calculated based on Table TCD p. 728 Group K	

Source: ASME Section II, Part D
Note: Mechanical properties are provided for information only; the design analysis is based on the mechanical properties of SA-240 Type 304, which are lower than Type 316.

Table 8-7
Material Properties, SA-240 Type 2205 / SA-182 Gr F60

Temp (°F)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} (10^{-6} °F^{-1})	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20		37.5	65.0	90.0				
70	29.0	37.5	65.0	90.0	7.0		8.2	0.122
100		37.5	65.0	90.0	7.1		8.3	0.122
150		37.5	60.5		7.2		8.6	0.123
200	28.2	37.5	57.8	90.0	7.3		8.8	0.125
250		37.0	55.5		7.3		9.1	0.128
300	27.5	35.8	53.7	86.8	7.4		9.3	0.128
350		34.9			7.5		9.5	0.129
400	27.0	34.2	51.2	83.5	7.6		9.8	0.131
450					7.6		10.0	0.132
500	26.4		49.6	81.6	7.7	0.280	10.2	0.132
550					7.8		10.5	0.134
600	26.0		47.9	80.7	7.8		10.7	0.134
650			46.9	80.5	7.9		10.9	0.135
700	25.5				7.9		11.2	0.136
750					8.0		11.4	0.137
800	25.1				8.0		11.6	0.137
850					8.1		11.9	0.139
900					8.1		12.1	0.139
950					8.2		12.3	0.140
1000					8.2		12.5	0.140
ASME	Table TM-1 p. 738 Group H	Table 5A p. 420, Lines 29, 30	Table Y-1 p. 632 Lines 1, 2, 3	Table U, pp. 502-503, Lines 7, 8, 9	Table TE-1 p. 708 Group 2	Table PRD p. 744	Calculated based on Table TCD p. 728 Group K	

Source: ASME Section II, Part D

Note: These properties are provided for information only; the design analysis is based on the mechanical properties and thermal conductivity of SA-240 Types 304 and 316, which bound the duplex steel

Table 8-8
Material Properties, SA-240 UNS S31803 / SA-182 Gr F51

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} (10^{-6} $^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20				90				
70	29.0			90	7.0		8.2	0.122
100		30	65	90	7.1		8.3	0.123
150					7.2		8.6	0.125
200	28.2	30	57.8	90	7.3		8.8	0.125
250					7.3		9.1	0.128
300	27.5	28.9	53.7	86.8	7.4		9.3	0.128
350					7.5		9.5	0.129
400	27.0	27.8	51.2	83.5	7.6		9.8	0.131
450					7.6		10.0	0.132
500	26.4	27.2	49.6	81.6	7.7	0.280	10.2	0.132
550					7.8		10.5	0.134
600	26.0	26.9	47.9	80.7	7.8		10.7	0.134
650					7.9		10.9	0.135
700	25.5				7.9		11.2	0.136
750					8.0		11.4	0.137
800	25.1				8.0		11.6	0.137
850					8.1		11.9	0.139
900					8.1		12.1	0.139
950					8.2		12.3	0.140
1000					8.2		12.5	0.140
ASME	Table TM-1 p. 738 Group H	ASME Code Case N-635-1	ASME Code Case N-635-1	Table U	Table TE-1 p. 708 Group 2	Table PRD p. 744	Calculated based on Table TCD p. 728 Group K	

Sources: ASME Section II, Part D and ASME Code Case N-635-1

Note: These properties are provided for information only; the design analysis is based on the mechanical properties and thermal conductivity of SA-240 Types 304 and 316, which bound the duplex steels.

Table 8-9
Material Properties, SA-350 Gr LF3

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_u (ksi)	S_y (ksi)	α_{AVG} ($10^{-6}\ {}^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-100	28.6							
-20		23.3	70.0	37.5				
70	27.8	23.3	70.0	37.5	6.4		23.7	0.107
100		23.3	70.0	37.5	6.5		23.6	0.108
150				35.3	6.6		23.5	0.111
200	27.1	22.9	70.0	34.3	6.7		23.5	0.115
250				33.7	6.8		23.4	0.117
300	26.7	22.1	70.0	33.2	6.9		23.4	0.121
350					7.0	0.280	23.3	0.123
400	26.2	21.4	70.0	32.0	7.1		23.1	0.126
450					7.2		23.0	0.129
500	25.7	20.3	70.0	30.4	7.3		22.7	0.131
550					7.3		22.5	0.134
600	25.1	18.8	70.0	28.2	7.4		22.2	0.137
650		17.9	70.0	26.8	7.5		21.9	0.139
700	24.6	16.9	66.5	25.3	7.6		21.6	0.142
ASME	Table TM-2	Table 2A	Table U	Table Y-1	Table TE-1, Group 1	Table PRD	Calculated based on Table TCD group C	

Source: ASME Section II, Part D

Table 8-10
Material Properties, High Strength Low Alloy Steel

Temp (°F)	$E^{(1)}$ (10^3 ksi)	$S_y^{(1)(2)}$ (ksi)	$S_u^{(1)(3)}$ (ksi)	Thermal Expansion 10^{-6} in/(in-°F) (1)	Thermal Conductivity Btu/(hr-ft-°F) (1)	Specific Heat Btu/(lb-°F) (1)	Density lb/in^3
-20	29.3	100.2	105.2				0.283 (1)
70	29.0	96.4	101.2				
100	28.9	95.4	100.2	6.60	23.8		
200	28.4	91.6	96.2	6.90	24.4	0.110	
300	28.0	87.7	92.1	7.20	24.6		
400	27.6	83.9	88.1	7.40	24.4	0.120	
500	27.0	80.0	84.0	7.55	24.0		
600	26.2	76.1	79.9	7.70	23.3	0.130	
700	25.2	71.3	74.9	7.80	22.7		
800	24.1	66.0	69.3	7.88	22.0	0.145	

Notes:

1. Listed values for yield stress calculated from rate of reduction provided in Figure 2.3.1.1.1, Figure 2.3.1.1.4 for modulus of elasticity, and Figure 2.3.1.0 for thermal properties from Reference [8-17].
2. Listed values based on Reference [8-17].
3. Yield stress values calculated based on 80 ksi at 500 °F.
4. Ultimate strength based on 1.05 Sy.
5. Specified minimum room temperature tensile test acceptance criteria for ASTM A829 Gr 4130 steel are
   $S_y \geq 103.6$ksi
   $S_u \geq 123.1$ ksi
   The acceptance criteria for testing at room temperature for other HSLA steels shall be identified using a similar method described in [8-24].

Table 8-11
Material Properties, SA-516 Gr 70 and ASTM A516 Gr 70

Temp (°F)	S_m (ksi)	S_y (ksi)	S_u (ksi)	E (10^3 ksi)	α_{INST} (10^{-6} °F^{-1})	α_{AVG} (10^{-6} °F^{-1})	ρ (lb/in³)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-100	--	--	--	30.3	--	--	--	--	--
-20 - 100	23.3	38.0	70.0	--	--	--	0.280		
70	--	--	--	29.4	6.4	6.4		34.9	0.103
100	--	--	--	--	6.6	6.5		34.7	0.106
150	23.3	35.7	--	--	6.8	6.6		34.2	0.110
200	23.2	34.8	70.0	28.8	7.0	6.7		33.7	0.114
250	--	34.2	--	--	7.2	6.8		33.0	0.117
300	22.4	33.6	70.0	28.3	7.3	6.9		32.3	0.119
350	--	--	--	--	7.5	7.0		31.6	0.122
400	21.6	32.5	70.0	27.9	7.7	7.1		30.9	0.124
450	--	--	--	--	7.8	7.2		30.1	0.126
500	20.6	31.0	70.0	27.3	8.0	7.3		29.4	0.128
550	--	--	--	--	8.2	7.3		28.7	0.131
600	19.4	29.1	70.0	26.5	8.3	7.4		28.0	0.134
650	18.8	28.2	70.0	--	8.5	7.5		27.3	0.136
700	18.1	27.2	70.0	25.5	8.7	7.6		26.6	0.140
750	--	26.3	69.1	--	8.8	7.7		25.3	0.143
800	--	25.5	58.6	24.2	9.0	7.8		24.6	0.147
ASME	Table 2A	Table Y-1	Table U	Table TM-1 C≤0.30%	Table TE-1 Group 1		Table PRD	Calculated from Table TCD Group A	

Source: ASME Section II, Part D [8-2]

Table 8-12
SA-182, Type F6NM

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} ($10^{-6}\ ^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20								
70	29.4	38.3	90.0	115.0	5.9		14.2	0.106
100					6.0		14.2	0.108
150					6.1		14.3	0.112
200	28.7	38.3	86.5	115.0	6.2		14.3	0.114
250					6.2		14.4	0.116
300	28.1	38.3	84.6	115.0	6.3		14.4	0.119
350					6.4		14.4	0.121
400	27.5	37.9	82.8	113.7	6.4	.280	14.5	0.124
450					6.4		14.5	0.126
500	26.9	36.5	80.8	109.5	6.5		14.5	0.130
550					6.5		14.6	0.134
600	26.3	35.0	78.5	105.1	6.5		14.6	0.137
650					6.6		14.6	0.140
700	25.7	33.4	75.7	100.3	6.6		14.6	0.144
750					6.6		14.6	0.147
800					6.7		14.6	0.151
ASME	Table TM-1, S13800	Table 2A	Table Y-1	Table U,	Table TE-1, 13 Cr	Table PRD	Calculated based on Table TCD Group G	

Source: ASME Section II, Part D [8-2]

Table 8-13
Material Properties, SA-193 Gr B7 Bolting ≤2 ½ inch

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} ($10^{-6}\ {}^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20	30.5	35.0	105.0	125.0				
70	29.6	35.0	105.0	125.0	6.4		23.7	0.114
100		35.0	105.0	125.0	6.5		23.6	0.114
150				125.0	6.6		23.5	0.117
200	29.0	32.6	98.0	125.0	6.7		23.5	0.119
250				125.0	6.8		23.4	0.121
300	28.5	31.4	94.1	125.0	6.9		23.4	0.122
350				125.0	7.0		23.3	0.124
400	28.0	30.5	91.5	125.0	7.1	0.280	23.1	0.126
450				125.0	7.2		23	0.127
500	27.4	29.5	88.5	125.0	7.3		22.7	0.129
550				125.0	7.3		22.5	0.129
600	26.9	28.4	85.3	125.0	7.4		22.2	0.13
650		27.7	83.0	124.4	7.5		21.9	0.131
700	26.2	26.9	80.6	119.6	7.6		21.6	0.132
750		25.9	77.6	114.3	7.7		21.3	0.132
800	25.6	24.6	73.9	108.4	7.8		21	0.133
ASME	Table TM-1 Group C	Table 4	Table Y-1	Table U	Table TE-1 Group 1	Table PRD	Calculated based on Table TCD Group C	

Source: ASME Section II, Part D [8-2]

Table 8-14
Material Properties, SA-540 Grade B23 Class 1 Bolting

Temp (^oF)	E (10^3 ksi)	S_m (ksi)	S_y (ksi)	S_u (ksi)	α_{AVG} ($10^{-6}\ {}^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-20	--	50.0	150.0	165.0	--	0.280	21.0	0.106
70	27.8	50.0	150.0	165.0	6.4		21.0	0.106
100	--	50.0	150.0	165.0	6.5		21.0	0.108
150	--	--	--	--	6.6		21.2	0.112
200	27.1	47.8	144.0	165.0	6.7		21.3	0.115
250	--	--	--	--	6.8		21.4	0.117
300	26.7	46.2	140.3	165.0	6.9		21.5	0.120
350	--	--	--	--	7.0		21.5	0.122
400	26.2	44.8	137.9	165.0	7.1		21.5	0.124
450	--	--	--	--	7.2		21.5	0.127
500	25.7	43.4	136.0	165.0	7.3		21.4	0.129
550	--	--	--	--	7.3		21.3	0.132
600	25.1	41.4	133.4	165.0	7.4		21.1	0.135
650	--	--	131.4	--	7.5		20.9	0.138
700	24.6	--	129.0	158.6	7.6		20.7	0.141
750	--	--	126.0	--	7.7		20.5	0.144
800	--	--	122.3	--	7.8		20.2	0.147
ASME	Table TM-1 Group B	Table 4	Table Y-1	Table U	Table TE-1 Group 1	Table PRD	Calculated from Table TCD Group D	

Source: ASME Section II, Part D [8-2]

Table 8-15
Material Properties, ASTM A913 Grade 70 High Strength Low-Alloy Steel

Temp (°F)	E^2 (10^3 ksi)	S_y [3] (ksi)	S_u [4] (ksi)	α_{AVG} [5] (10^{-6} F^{-1})	γ (lb/in^3)	K (BTU/hr-ft-°F)	C_p (BTU/lb-°F)
-20					0.280		
70	29.0	70.0 (1)	90.0(1)			27.3	0.106
100	29.0	67.9	90.0	6.3		27.6	0.110
150						27.8	0.114
200	28.4	64.4	88.2	6.5		27.8	0.118
250						27.6	0.121
300	27.8	61.6	90.0	6.7		27.3	0.124
350						26.9	0.126
400	27.3	59.5	90.0	6.9		26.5	0.129
450						26.1	0.131
500	26.7	58.1	90.0	7.1		25.7	0.133
550						25.3	0.135
600	26.1	57.4	86.4	7.2		24.9	0.138
650						24.5	0.141
700	25.5	56.0	81.0	7.4		24.1	0.144

Notes

1. Reference [8-14].
2. Based on lowest rate of reduction provided in [8-12] Figure 7.5.
3. Based on lowest rate of reduction provided in 8-12] Figure 7.3.
4. Based on lowest rate of reduction provided in [8-12] Figure 7.4.
5. Based on lowest rate of reduction provided in [8-12] Figure 7.6.
6. ASME Section II Part D, Table PRD.
7. ASME Section II Part D, Table TCD, Material Group B.

Table 8-16
Material Properties, Aluminum ASTM B221 Alloy 6061-O

Temp (°F)	E (10^3 ksi)	Elongation in 4D, %	S_u (ksi)	S_y (ksi)	α_{AVG} (10^{-6} °F^{-1})	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-100								
-20								
70					12.1		96.1	0.213
75	9.9	30	18.0	8.0				
100					12.4		96.9	0.215
150					12.7		98.0	0.218
200					13.0		99.0	0.221
212	9.5	30	18.0	8.5				
250					13.1	0.098	99.8	0.223
300	9.1	35	15.0	9.5	13.3		100.6	0.226
350	8.9	45	12.0	8.5	13.4		101.3	0.228
400	8.6	60	10.0	7.5	13.6		101.9	0.230
450	8.3	75	8.5	6.0	13.8			
500	7.9	80	7.0	5.5	13.9			
550					14.1			
600	6.8	80	5.0	4.2	14.2			
ASME	Kaufman, p. 163(1)	Kaufman, p. 163(1)	Kaufman, p. 163(1)	Kaufman, p. 163(1)	Table TE-2 p. 714 Aluminum Alloys	Table PRD p. 744	Calculated based on Table TCD p. 735, group A96061	

Source: ASME Section II, Part D, except as noted

Notes:

1. Annealed values used for analysis are typical tensile properties from [8-16] p. 163.
2. Mechanical properties are used for design of the basket peripheral transition rails. The thermal design analysis uses values for density, thermal conductivity, and heat capacity that are lower than those in this table.
3. Thermal conductivity and heat capacity used in the design analyses are lower than those shown in the table.

Table 8-17
Material Properties, Aluminum ASTM B209 Alloy 6061-T6

Temp (^oF)	E (10^3 ksi)	S (ksi)	S_u (ksi)	S_y (ksi)	α_{AVG} ($10^{-6}\ {}^oF^{-1}$)	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-100	10.5							
-20		12.0	41.0	35.0		0.098		
70	10.0	12.0	40.0	35.0	12.1		96.1	0.213
100		12.0	39.6	35.0	12.4		96.9	0.215
150		12.0		34.6	12.7		98.0	0.218
200	9.6	12.0	37.8	33.7	13.0		99.0	0.221
250		9.9		32.4	13.1		99.8	0.223
300	9.2	8.4	31.0	27.4	13.3		100.6	0.226
350		6.3		20.0	13.4		101.3	0.228
400	8.7	4.5	15.0	13.3	13.6		101.9	0.230
450					13.8			
500	8.1		5.0		13.9			
550					14.1			
600			2.7		14.2			
ASME	Table TM-2	Table 1B	Ref. [8-13], p. 117	Table Y-1	Table TE-2	Table PRD	Calculated based on Table TCD group A96061	

Source: ASME Section II, Part D [8-2], except as noted.

Table 8-18
Material Properties, Aluminum ASTM B209 Alloy 6061-T0
(Weld and Heat Affected Zone)

Temp (°F)	E (10^3 ksi)	S (ksi)	S_u (ksi)	S_y (ksi)	α_{AVG} (10^{-6} °F^{-1})	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-100	10.5							
-20		6.0	24.0	9.0		0.098		
70	10.0	6.0	24.0	9.0	12.1		96.1	0.213
100		6.0	19.1	9.0	12.4		96.9	0.215
150		6.0	19.1	9.0	12.7		98.0	0.218
200	9.6	6.0	19.1	9.0	13.0		99.0	0.221
250		5.9	18.8		13.1		99.8	0.223
300	9.2	5.5	17.5	9.5	13.3		100.6	0.226
350		4.6	14.6	8.5	13.4		101.3	0.228
400	8.7	3.5	11.1	7.5	13.6		101.9	0.230
450					13.8			
500	8.1				13.9			
550					14.1			
600					14.2			
ASME	Table TM-2	Table 1B	Calculated based on Appendix 1(1)		Table TE-2	Table PRD	Calculated based on Table TCD group A96061	

Source: ASME Section II, Part D [8-2], except as noted

Notes:

1. Calculated values based on S provided in Table 1B using minimum room temperature tensile stress (24 ksi) and minimum room temperature yield stress (20 ksi) or welded aluminum alloys as provided by [8-25], p. 435.
2. Annealed values for 6061 at 300 °F and above are typical tensile properties from [8-16], p. 163.
3. These properties are used for evaluation of the TC108 removable neutron shield welds and heat affected zone stresses.

Table 8-19
Material Properties, Aluminum ASTM B209 Alloy 1100

Temp (°F)	S_u (ksi)	S_y (ksi)	E (10^3 ksi)	α_{AVG} (10^{-6} °F^{-1})	ρ (lb/in^3)	K (Btu/hr-ft-°F)	C_p (Btu/lb-°F)
-100			10.5		0.098		
70	13.0	5.0	10.0	12.1		133.1	0.214
100				12.4		131.8	0.216
150				12.7		130.0	0.219
200			9.6	13.0		128.5	0.222
212	11.0	4.6					
250				13.1		127.3	0.225
300	8.5	4.2	9.2	13.3		126.2	0.227
350	7.5	3.8		13.4		125.3	0.229
400	6.0	3.5	8.7	13.6		124.5	0.232
450	5.0	3.1		13.8			
500	4.0	2.6	8.1	13.9			
550				14.1			
600	2.9	2.0		14.2			
	[8-16] p.9(1)	[8-16] p.9(1)	Table TM-2(2)	Table TE-2(2) Al Alloys	Table PRD(2)	Calculated based on Table TCD(2) group A91100	

Notes:

1. Data from [8-16] p 171, typical tensile properties. Values taken for h=10,000, unless otherwise specified. Tensile properties are for information only; they are not credited in the structural analysis.
2. ASME Section II, Part D [8-2].

Table 8-20(a)
Static Mechanical Properties, ASTM B29 Lead

Temp. (°F)	Static Stress Properties (ksi)			E (10^6 psi)	Coefficient of Thermal Exp (10^{-6} in/in/°F)
	Yield (S_y)		Ultimate (S_u)		
	Tension	Compression	Tension		
-99	-	-	-	2.50	15.28
70	-	-	-	2.34	16.07
100	0.584	0.490	1.570	2.30	16.21
175	0.509	0.428	1.162	2.20	16.58
250	0.498	0.391	0.844	2.09	16.95
325	0.311	0.320	0.642	1.96	17.54
440	-	-	-	1.74	18.50
620	-	-	-	1.36	20.39

Notes

1. Sources: [8-18] and [8-19].

Table 8-20(b)
Dynamic Mechanical Properties, ASTM B29 Lead

Strain (in/in)	Stress (ksi)				
	At 100 °F	At 230 °F	At 300 °F	At 350 °F	At 500 °F
0.000485	1.14	1.06	1.00	0.97	0.86
0.03	2.2	2.0	1.7	1.5	1.1
0.1	3.3	2.8	2.38	2.1	1.26
0.3	4.9	3.2	2.72	2.4	1.44
0.5	5.6	3.6	3.06	2.7	1.62

Notes

1. Source: [8-20].

Table 8-21
Lead Properties

Temp. (°F)	Density (lb/in^3)	Coefficient of Thermal Exp (10^{-6}in/in/°F)	Thermal Conductivity W/mK
-99	0.41	15.28	
70		16.07	
80			35.3
100		16.21	
175		16.58	
250		16.95	
260			34.0
325		17.54	
440			32.8
500		19.14	

Notes:

Sources: [8-19] and [8-35].

Table 8-22
Not Used

Table 8-23
Material Properties, Concrete

Temp (°F)	f_c' (ksi)(1)	Modulus of Elasticity (10^3 ksi) (1)	a_{AVG} (10^{-6} F^{-1}) (1)	Density (lb/ft^3) (2)	Specific Heat (3) (Btu/lb-°F)	Thermal Conductivity(3) (Btu/hr-in-°F)
70						0.0958
100	5.0	4.0	5.5	150	0.22	
200	5.0	3.6	5.5		0.22	
300	5.0	3.3	5.5		0.22	
400	4.5	3.0	5.5		0.22	
500	4.5	2.9	5.5		0.22	
1382						0.048

Notes

1. Reference [8-12].
2. Reference [8-8]. The shielding analysis uses a density of 140 pcf.
3. Reference [8-23].

Table 8-24
Material Properties, ASTM A615 Grade 60 and ASTM A706 Grade 600 Reinforcing Steel

Temp (°F)	Yield Strength (ksi) (1)	Modulus of Elasticity (10^3 ksi) (1)	Density (lb/ft^3) (2)
100	60.0	29.0	490
200	57.0	28.4	
300	54.0	27.8	
400	51.0	27.3	
500	51.0	27.0	

Notes

1. Reference [8-12].
2. Reference [8-11].

Table 8-25
Materials Properties, Zircaloy-2

Temperature °F	Young's Modulus[(1)] E, (psi)	Yield Stress[(1)] S_y (psi)
200[(2)]	1.48E+07	119,601
300	1.43E+07	110,516
400	1.38E+07	101,431
500	1.33E+07	93,451
600	1.28E+07	85,810
700	1.23E+07	77,906
750	1.21E+07	73,712

Notes:

1. Values in this Table are derived from the equations provided in Section 2 of Reference [8-21] with the following values
   - Strain rate of 0.5 s^{-1}
   - Cold work (CW) ratio of 0.0
   - Oxygen content (Δ) ratio of 0.0012
   - Fast neutron fluence (Φ) of 1.2 x 10^{26} n/m^2
   - Uncertainty of 67 MPa (9.7 ksi) is not included in the Yield Stress value
2. E and S_y values at this temperature are obtained by linear extrapolation of these values at 300 and 400 ºF.

Table 8-26
Materials Properties, Zircaloy-4

Temperature (°F)	PWR, Zircaloy-4[1]	
	E (psi)	Yield Stress (psi)
300	1.22E+07	126102
350	1.19E+07	120769
400	1.17E+07	116272
450	1.14E+07	112390
500	1.12E+07	108921
550	1.09E+07	105683
600	1.07E+07	102512
625	1.06E+07	100904
650	1.04E+07	99259
675	1.03E+07	97560
700	1.02E+07	95793
725	1.01E+07	93944
750	9.93E+06	92000

Notes:

1. Values in this Table are derived from the equations provided in Section 2 of Reference [8-21] with the following values

   - Strain rate of 0.5 s^{-1}
   - Cold work (CW) ratio of 0.0
   - Oxygen content (Δ) ratio of 0.0012
   - Fast neutron fluence (Φ) of 1.2 x 10^{26} n/m^2
   - Uncertainty of 67 MPa (9.7 ksi) is not included in the Yield Stress value

Table 8-27
Material Properties, Helium

Temperature (K)	Thermal conductivity (W/m-K)	Temperature (°F)	Thermal conductivity (Btu/hr-in-°F)
300	0.1499	80	0.0072
400	0.1795	260	0.0086
500	0.2115	440	0.0102
600	0.2466	620	0.0119
800	0.3073	980	0.0148
1000	0.3622	1340	0.0174
1050	0.3757	1430	0.0181

The above data are calculated based on the following polynomial function from [8-22].

$k = \sum C_i \; T^i$ for conductivity in(W/m-K) and T in (K)

For 300 < T < 500 K	
C0	-7.761491E-03
C1	8.66192033E-04
C2	-1.5559338E-06
C3	1.40150565E-09
C4	0.0E+00

for 500< T < 1050 K	
C0	-9.0656E-02
C1	9.37593087E-04
C2	-9.13347535E-07
C3	5.55037072E-10
C4	-1.26457196E-13

No heat capacity or density is considered for helium.

Table 8-28
Material Properties, Air

Temperature (K)	Thermal conductivity (W/m-K)	Temperature (°F)	Thermal conductivity (Btu/hr-in-°F)
250	0.02228	-10	0.0011
300	0.02607	80	0.0013
400	0.03304	260	0.0016
500	0.03948	440	0.0019
600	0.04557	620	0.0022
800	0.05698	980	0.0027
1000	0.06721	1340	0.0032

The above data are calculated based on the following polynomial function from [8-22].

$k = \sum C_i\ T^i$ for conductivity in (W/m-K) and T in (K)

For 250 < T < 1050 K	
C0	-2.2765010E-03
C1	1.2598485E-04
C2	-1.4815235E-07
C3	1.7355064E-10
C4	-1.0666570E-13
C5	2.4766304E-17

$c_P = \sum A_i\ T^i$ for specific heat in (kJ/kg-K) and T in (K)

For 250 < T < 1050 K	
A0	0.103409E+1
A1	-0.2848870E-3
A2	0.7816818E-6
A3	-0.4970786E-9
A4	0.1077024E-12

$\mu = \sum B_i\ T^i$ for viscosity (N-/m2)×106 and T in (K)

For 250 < T < 600 K	
B0	-9.8601E-1
B1	9.080125E-2
B2	-1.17635575E-4
B3	1.2349703E-7
B4	-5.7971299E-11

For 600 < T < 1050 K	
B0	4.8856745
B1	5.43232E-2
B2	-2.4261775E-5
B3	7.9306E-9
B4	-1.10398E-12

$\rho = \dfrac{P}{RT}$ for density (kg/m^3)

P=101.3 kPa; R = 0.287040 kJ/kg-K; T = air temp in (K)

Table 8-29
Material Properties, Solid Neutron Shielding

Borated HDPE Mechanical Properties Reference [8-9]	
Density	.036 lb/in^3
Tensile Strength	2,407 psi
Modulus of Elasticity	111,200 psi
Flexural Strength	4,220 psi
Flexural modulus of Elasticity	126,000 psi
Flexural Yield Strength	4220 psi
Compressive Strength	957 psi

Notes:

1. All values are nominal.
2. Properties taken at 73 °F.

Borated HDPE Thermal Properties Reference [8-9]	
Coefficient of thermal expansion (-40°F to 30°F)	1.1 x 10^{-5}in./in./ °F
Melting Point	260 °F
Continuous service temperature in Air (max.)	180 °F

Table 8-30
MCNP Material Compositions (wt. %)

Element	Carbon Steel	Stainless Steel	Dry Air	Water	Regular Concrete	Soil
Hydrogen	-	-	-	11.1894	1.0	-
Helium	-	-	-	-	-	-
Boron	-	-	-	-	-	-
Carbon	0.5	0.04	0.0124	-	-	-
Nitrogen	-	-	75.5268	-	-	-
Oxygen	-	-	23.1781	88.8106	53.2	51.37
Sodium	-	-	-	-	2.9	0.614
Magnesium	-	-	-	-	-	1.33
Aluminum	-	-	-	-	3.4	6.856
Silicon	-	0.5	-	-	33.7	27.118
Phosphorous	-	0.023	-	-	-	-
Sulfur	-	0.015	-	-	-	-
Argon	-	-	1.2827	-	-	-
Potassium	-	-	-	-	-	1.433
Calcium	-	-	-	-	4.4	5.117
Titanium	-	-	-	-	-	0.461
Chromium	-	19	-	-	-	-
Manganese	-	1	-	-	-	0.0716
Iron	99.5	70.173	-	-	1.4	5.629
Cobalt	-	-	-	-	-	-
Nickel	-	9.25	-	-	-	-
Copper	-	-	-	-	-	-
Zirconium	-	-	-	-	-	-
Niobium	-	-	-	-	-	-
Molybdenum	-	-	-	-	-	-
Tin	-	-	-	-	-	-
Lead	-	-	-	-	-	-
Density (g/cm^3)	7.82	8.00	0.001205	(1)	2.243	1.52

Note:

1. 0.958 g/cm3 inside the DSC and 0.9982 g/cm3 inside the neutron shield.

Table 8-31
MCNP Borated Polyethylene Composition

Element	Atom Density (atom/b-cm)
Hydrogen	6.1848E-02
Boron-10	5.5978E-04
Boron-11	2.2532E-03
Carbon	3.6381E-02
Oxygen	4.2195E-03
Total	1.0526E-01

Table 8-32
Minimum B-10 Content in the Neutron Poison Plates

EOS-37PTH

Basket Type	Minimum Specified B-10 Areal Density for MMC (mg/cm^2)	B-10 Content Used in Criticality Evaluation (mg/cm^2)
A1 / A2 / A3	28.0	25.2
B1 / B2 / B3	35.0	31.5

EOS-89BTH

Basket Type	Minimum B-10 Areal Density (mg/cm^2)		B-10 Content Used in Criticality Evaluation (mg/cm^2)
	MMC	BORAL®	
A1 / A2 / A3	32.7	-	29.4
B1 / B2 / B3	41.3	-	37.2
C1 / C2 / C3	-	60.0	45.0

Table 8-33
Material Property Data for Criticality Analysis
(Part 1 of 2)

Material	ID	Density g/cm^3	Element	Wt. %	Atom Density (atoms/b-cm)
UO_2 (Enrichment – 1.0 to 5.0 wt. %) (1)	1	10.686	U-235	4.41	1.20668E-03
			U-238	83.74	2.26374E-02
			O	11.85	4.76881E-02
Zircaloy-4	2	6.56	Zr	98.23	4.2541E-02
			Sn	1.45	4.8254E-04
			Fe	0.21	1.4856E-04
			Cr	0.10	7.5978E-05
			Hf	0.01	2.2133E-06
Water (Pellet Clad Gap)	3	0.998	H	11.1	6.6769E-02
			O	88.9	3.3385E-02
Stainless Steel (SS304)	4	7.94	C	0.080	3.1877E-04
			Si	1.000	1.7025E-03
			P	0.045	6.9468E-05
			Cr	19.000	1.7473E-02
			Mn	2.000	1.7407E-03
			Fe	68.375	5.8545E-02
			Ni	9.500	7.7402E-03
Borated Water (2000 – 2500 ppm Boron)(2)	5	1.00	H	11.163	6.67515E-02
			O	88.587	3.33757E-02
			B-10	0.046	2.77126E-05
			B-11	0.204	1.11547E-04
$^{11}B_4C$ in CC	7	2.52	B-11	78.57	1.08305E-01
			C	21.43	2.70763E-02
Aluminum	8	2.702	Al	100.0	6.0307E-02
Water	10	0.998	H	11.1	6.6769E-02
			O	88.9	3.3385E-02
Lead	11	11.344	Pb	100.0	3.2969E-02

Table *8-33*
Material Property Data for Criticality Analysis
(Part 2 of 2)

Material	ID	Density g/cm^3	Compound	Wt. %	Element	Atom Density (atoms/b-cm)
EOS-37PTH						
MMC Poison Plate for Basket Type A1 / A2 / A3 (25.2 mg B-10/cm^2)	9	2.693	B_4C	15.57	B-10	3.63761E-03
					B-11	1.46418E-02
					C	4.56986E-03
			Al	84.43	Al	5.07473E-02
MMC Poison Plate for Basket Type B1 / B2 / B3 (31.5 mg B-10/cm^2)	9	2.693	B_4C	19.463	B-10	4.54713E-03
					B-11	1.83028E-02
					C	5.71247E-03
			Al	80.537	Al	4.84074E-02
EOS-89BTH						
MMC Poison Plate for Basket Type A1 / A2 / A3 (29.4 mg B-10/cm^2)	9	2.669	B_4C	17.18	B-10	3.97731E-03
					B-11	1.60092E-02
					C	4.99662E-03
			Al	82.82	Al	4.93353E-02
MMC Poison Plate for Basket Type B1 / B2 / B3 (37.2 mg B-10/cm^2)	9	2.660	B_4C	21.81	B-10	5.03252E-03
					B-11	2.02565E-02
					C	6.32227E-03
			Al	78.19	Al	4.64244E-02
BORAL Poison Plate for Basket Type C1 / C2 / C3 (45.0 mg B-10/cm^2)	9	2.450	B_4C	28.64	B-10	6.08772E-03
					B-11	2.45038E-02
					C	7.64789E-03
			Al	71.36	Al	3.90204E-02

Note:

1. The composition for maximum enrichment evaluate at 5.0 wt. % U-235 is provided.
2. Applies to EOS-37PTH only. EOS-89BTH evaluated with 100% internal moderator density. The composition for the maximum soluble boron concentration at 100 % internal moderator density is provided.

Table 8-34
Not Used

Table 8-35

Emissivity of [ ] on Basket Steel Plates

[ ]

Table 8-36
Summary of Relevant Corrosion Rates

Condition	***Soluble Boron Concentration (ppm)***	***Temperature °F (°C)***	***Corrosion Rate in/yr (mm/yr)***
Aerated Water	*2,500*	*70 (21)*	*0.002 (0.05)*
	2,000	*100 (38)*	*0.002 – 0.0045 (0.05 – 0.11)*
	2,500	*100 (38)*	*0.007 (0.18)*
	2,000	*104 (40)*	*0.007 (0.18)*
	2,500	*140 (60)*	*0.015 (0.38)*

Note: Data are taken from Figure 4-3 and Table 4-3 of EPRI-1000975 [8-42].

CHAPTER 9
OPERATING PROCEDURES

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9. OPERATING PROCEDURES

This chapter presents the operating procedures for the NUHOMS® EOS System described in previous chapters and shown on the drawings in Chapter 1, Section 1.3. The procedures include preparation of the NUHOMS® EOS System dry shielded canister (DSC) and fuel loading, closure of the DSC, transfer to the independent spent fuel storage installation (ISFSI) using the transfer cask (TC), DSC transfer into the horizontal storage module (HSM), monitoring operations, and DSC retrieval from the HSM. The NUHOMS® EOS transfer equipment, and the existing plant systems and equipment are used to accomplish these operations. Procedures are delineated here to describe how these operations are to be performed and are not intended to be limiting. Standard fuel and cask handling operations performed under the plant's 10 CFR Part 50 operating license are described in less detail. Existing operational procedures may be revised by the licensee and new ones may be developed according to the requirements of the plant, provided that the conditions specified in the technical specifications and the NUHOMS® EOS CoC are met. Temporary shielding may be used throughout as appropriate to maintain doses as low as reasonably achievable (ALARA). Helium is the only gas that is authorized inside the canister. After water is drained from the DSC, (Sections 9.1.2 and 9.1.3), the DSC shall be backfilled only with helium.

The following sections outline the typical operating procedures for the NUHOMS® EOS System. These generic NUHOMS® EOS procedures have been developed to minimize the amount of time required to complete the subject operations, to minimize personnel exposure, and to assure that all operations required for DSC loading, closure, transfer, and storage are performed safely. Plant specific ISFSI procedures are to be developed by each licensee in accordance with the requirements of 10 CFR 72.212 (b) and the guidance of Regulatory Guide 3.61 [9-4]. The generic procedures presented here are provided as a guide for the preparation of plant specific procedures and serve to point out how the NUHOMS® EOS System operations are to be accomplished. They are not intended to be limiting, in that the licensee may evaluate that alternate acceptable means are available to accomplish the same operational objective.

Pictograms of the NUHOMS® EOS System operations are presented in Figure 9-1. The location of the various operations may vary with individual plant requirements. The steps described in this document are the recommended generic operating procedures for the NUHOMS® EOS System.

See Chapter 1 for description of components.

The generic terms used throughout this section are as follows.

- TC, or transfer cask is used for the TC108 or TC125 or TC135 transfer cask.
- DSC is used for the EOS-37PTH DSC or EOS-89BTH DSC.
- HSM is used for the EOS-HSM or EOS-HSMS module.

Note: If applicable to the planned DSC heat zone loading configuration per Figure 1 or 2 of the Technical Specifications [9-5], the FC system should be installed and verified operational prior to initiating the transfer operations.

9.1 Procedures for Loading the DSC and Transfer to the HSM

The following steps describe the recommended generic operating procedures for the NUHOMS® EOS System. Since the design of the EOS-TC includes two primary variations – the TC108 (with a removable neutron shield jacket) and the TC125/TC135 (with an integral neutron shield), some steps have alternate steps specific to one of these EOS-TC designs. A pictorial representation of key phases of this process is provided in Figure 9-1.

9.1.1 TC and DSC Preparation

1. Prior to placement of the Authorized Contents in dry storage:

    a. The candidate fuel assemblies (FAs) and control components (CCs), if applicable, shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Section 2.1 (EOS-37PTH DSC) or Section 2.2 (EOS-89BTH DSC) of the Technical Specifications [9-5].

2. Prior to being placed in service: clean and/or decontaminate the TC as necessary to provide a surface contamination level of less than those specified in Section 3.3.1 of the Technical Specifications [9-5].

3. Place the TC in the vertical position in the designated area using the TC handling crane and the lifting yoke.

    a. TC125 or TC135: The neutron shield may need to be drained to meet the crane capacity when the loaded TC is pulled out of the pool.

    b. TC108: The neutron shield tank may be drained or removed from the TC108 and staged in an appropriate location.

4. Place scaffolding around the TC so that the top cover plate and surface of the TC are easily accessible to personnel.

    a. TC108 without neutron shield tank: Install protective cover around outer shell of the TC108 to minimize contamination of the outer shell of the TC.

5. Remove the TC top cover plate and examine the TC cavity for any physical damage and ready the TC for service.

    Note: Verify that a TC spacer of appropriate height is placed inside the TC to provide the correct airflow and interface at the top of the TC during loading, drying, and sealing operations for DSCs that are shorter than the TC cavity length.

6. Verify specified lubrication of the TC rails.

7. Examine the DSC for any physical damage that might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed.

8. Record the DSC serial number that is located on the grapple ring. Verify the DSC type and basket type against the DSC serial number. Verify that the DSC is appropriate for the specific fuel loading campaign per the criteria specified in Section 2.1 (EOS-37PTH DSC) or Section 2.2 (EOS-89BTH DSC) of the Technical Specifications [9-5].

9. Using a crane, lower the DSC into the TC cavity by the internal lifting lugs and rotate the DSC to match the TC and DSC alignment marks.

10. Fill the TC/DSC annulus with clean water. Place the inflatable seal into the upper TC liner recess and seal the TC\DSC annulus by pressurizing the seal with compressed air.

    Note: A TC/DSC annulus pressurization tank filled with clean water is connected to the top vent port of the TC via a hose to provide a positive head above the level of water in the TC/DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the TC/DSC annulus, provided a positive head is maintained at all times.

11. Fill the DSC cavity with water from the fuel pool or an equivalent source that meets the requirements of Section 3.2.1 of the Technical Specifications [9-5] for boron concentration, if applicable.

12. Place the top shield plug onto the DSC. Examine the top shield plug to ensure a proper fit. Optionally, the top shield plug, once fitted, may be removed and disconnected from the yoke. It may be installed later, once the DSC is loaded and prior to removing it from the pool.

13. Position the TC lifting yoke and engage the TC lifting trunnions and the rigging cables to the DSC top shield plug. Adjust the rigging cables, as necessary, to obtain even cable tension.

14. Visually inspect the yoke lifting arms to ensure that they are properly positioned and engaged on the TC lifting trunnions.

15. Move the scaffolding away from the TC as necessary.

16. Lift the TC just far enough to allow the weight of the TC to be distributed onto the yoke lifting arms. Reinspect the lifting arms to insure that they are properly positioned on the TC trunnions.

17. Optionally, secure a sheet of suitable material to the bottom of the TC to minimize the potential for ground-in contamination. This may also be done prior to initial placement of the TC in the designated area.

18. Prior to the TC being lifted into the fuel pool, the water level in the pool should be adjusted, as necessary, to accommodate the TC/DSC volume. If the water placed in the DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.

9.1.2 DSC Fuel Loading

1. Lift the TC/DSC and position it over the TC loading area of the spent fuel pool in accordance with the plant's 10 CFR Part 50 TC handling procedures.

2. Lower the TC into the fuel pool until the bottom of the TC is at the height of the fuel pool surface. As the TC is lowered into the pool, spray the exterior surface of the TC with clean water.

3. Place the TC in the location of the fuel pool designated as the TC loading area.

4. Disengage the lifting yoke from the TC lifting trunnions and move the yoke and the top shield plug clear of the TC. Spray the lifting yoke and top shield plug with clean water if it is raised out of the fuel pool.

5. The potential for fuel misloading is essentially eliminated through the implementation of procedural and administrative controls. The controls instituted to ensure that intact FAs and CCs, if applicable, are placed into a known cell location within a DSC, will typically consist of the following:
    - A TC/DSC loading plan is developed to verify that the intact fuel assemblies, and CCs, if applicable, meet the burnup, enrichment and cooling time parameters of Section 2.1 (EOS-37PTH DSC) or Section 2.2 (EOS-89BTH DSC) of the Technical Specifications [9-5].
    - The loading plan is independently verified and approved before the fuel load.
    - A fuel movement schedule is then written, verified, and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance of the fuel movement schedule.

6. Prior to loading of a FA and CC, if applicable, into the DSC, the identity of the assembly and CC, if applicable, is to be verified by two individuals using an underwater video camera or other means. Verification of CC identification is optional if the CC has not been moved from the host FA since its last verification. Read and record the identification number from the FA and CCs, if applicable, and check this identification number against the DSC loading plan, which indicates which FAs and CCs, if applicable, are acceptable for dry storage.

7. Position the FA for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Steps 6 and 7 for each FA loaded into the DSC. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly and CCs, if applicable, in the DSC.

8. After all the FAs and CCs, if applicable, have been placed into the DSC and their identities verified, position the lifting yoke and the top shield plug and lower the shield plug onto the DSC.

CAUTION: Verify that all the lifting height restrictions as a function of temperature specified in Section 5.2.1 of the Technical Specifications [9-5] can be met in the following steps that involve lifting of the TC.

9. Visually verify that the top shield plug is properly seated onto the DSC.

10. Position the lifting yoke with the TC trunnions and verify that it is properly engaged.

11. Raise the TC to the pool surface. Prior to raising the top of the TC above the water surface, stop vertical movement.

12. Inspect the top shield plug to verify that it is properly seated onto the DSC. If not, lower the TC and reposition the top shield plug. Repeat Steps 8 to 12 as necessary.

13. Continue to raise the TC from the pool and spray the exposed portion of the TC with clean water until the top region of the TC is accessible.

14. Disengage the rigging cables from the top shield plug and remove the eyebolts.

15. Drain any excess water from the top of the DSC shield plug back to the fuel pool.

16. Check the radiation levels at the center of the top shield plug and around the perimeter of the TC.

17. EOS-TC125 or TC135, or TC108 with filled neutron shield tank attached: If applicable, drain water from the DSC while filling the space inside the DSC with helium, as necessary to meet the plant lifting crane capacity limits. Helium must be used to fill the space above the water inside the DSC.

    a. EOS-TC108 without neutron shield tank: Water shall not be drained from the DSC below the top of the fuel, since the water is relied upon for shielding when the neutron shield tank is removed.

CAUTION: Do not remove water from the DSC cavity in a way that would leave the fuel uncovered if the water is drained from the neutron shield as well, or the neutron shield tank is removed.

18. Lift the TC from the fuel pool. As the TC is raised from the pool, continue to spray the TC with clean water.

19. Move the TC with loaded DSC to the designated area.

20. If applicable to keep the occupational exposure ALARA, replace the water removed from the DSC in Step 17 with spent fuel pool water or another source meeting the proper boron concentration, as specified in Section 3.2.1 of the Technical Specifications [9-5], if specified.

    a. EOS-TC108: Carefully remove the protective cover around the outer shell of the TC108 in a way that precludes cross-contamination of the outer shell of the TC. Temporary shielding may be installed, as necessary, to minimize personnel exposure.

21. Disengage the lifting yoke from the trunnions and position it clear of the TC.

    a. EOS-TC108: Using good ALARA practices, install the neutron shield tanks to the TC108.

22. Using good ALARA practices, if the neutron shield is drained, fill the neutron shield tanks of the TC with clean water.

9.1.3 DSC Drying and Backfilling

CAUTION: During performance of steps listed in Section 9.1.3, monitor the TC/DSC annulus water level and replenish if necessary.

1. Place scaffolding around the TC so that any point on the surface of the TC is easily accessible to personnel. Temporary shielding may be installed as necessary to minimize personnel exposure.

2. Decontaminate the exposed surfaces of the DSC shell perimeter and remove the inflatable TC/DSC annulus seal.

3. Connect the TC drain line to the TC, open the TC cavity drain port and allow water from the TC/DSC annulus to drain out until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with Section 3.3.1 of the Technical Specification [9-5] limits.

CAUTION: Radiation dose rates are expected to be high at the drain and vent port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

4. Prior to the start of welding operations, drain a minimum of 60 gallons of water from the DSC back into the fuel pool or other suitable location using the vacuum drying system (VDS) or an optional liquid pump. Alternatively, all the water from the DSC may be drained if precautions are taken to keep the occupational exposure ALARA. Helium must be used to fill the space above the water inside the DSC.

5. Install the automatic welding machine onto the inner top cover plate (ITCP) and place the inner top cover plate with the automatic welding machine onto the DSC. Verify proper fit-up of the inner top cover plate with the DSC shell.

6. Check radiation levels along surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.

CAUTION: Insert tubing of sufficient length and adequate temperature resistance through the vent port, to where it terminates just below the DSC shield plug. Connect the flexible tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner cover plate in compliance with Section 5.4 of the Technical Specifications [9-5]. Optionally, other methods may be used for continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate.

7. Attach the helium purge system to the drain port to allow purging of the atmosphere inside the DSC cavity to remove hydrogen gas.

8. Cover the TC/DSC annulus to prevent debris and weld splatter from entering the annulus.

   Note: Provision must be made to monitor the TC/DSC annulus water level and replenish if necessary.

9. Ready the automatic welding machine and tack weld the inner top cover plate to the DSC shell. Install the inner top cover plate weldment and remove the automatic welding machine.

CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity using the flexible tube arrangement or other alternate methods described in Step 6 during the inner top cover plate cutting and welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% (60.0% of flammability limit of 4.0%) [9-1 and 9-2]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with approximately 2-3 psig helium to reduce the hydrogen concentration safely below the 2.4% limit.

10. Perform root and final dye penetrant weld examination of the inner top cover plate weld in accordance with Section 4.4.4 of the Technical Specifications [9-5].

11. Install temporary shielding to minimize personnel exposure throughout the subsequent draining and vacuum drying, and welding operations, as required.

12. Remove the hydrogen monitoring system and attach a helium supply through the vent port on the ITCP.

13. Attach the VDS or the optional liquid pump through the drain port and drain water from the DSC while filling the space inside the DSC with helium.

14. Attach the VDS, if necessary, to vacuum dry the canister through the drain port and attach the VDS to a helium supply.

15. Close off the vent port to allow vacuum drying of the DSC.

16. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool.

    Note: Proceed cautiously when evacuating the DSC to avoid freezing consequences.

17. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the DSC cavity. The cavity pressure should be reduced in steps of approximately 100 mm Hg, 50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg or less, as specified in Section 3.1.1 of the Technical Specifications [9-5].

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with Section 3.1.1 of the Technical Specification [9-5] requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.

CAUTION: Radiation dose rates are expected to be high at the port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

18. Open the valve and allow the helium to flow into the DSC cavity.

19. Pressurize the DSC with helium to more than 18 psig, but do not exceed 23 psig and hold for 10 minutes.

20. Helium leak test the ITCP weld for a leak rate of 1×10^{-4} atm cm^3 /sec. This test is optional.

21. If a leak is found, repair the weld, repressurize the DSC, and repeat the helium leak test.

22. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

23. Seal weld the prefabricated plug over the vent port and perform root and final dye penetrant weld examinations in accordance with Section 4.4.4 of the Technical Specifications [9-5].

24. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced to 3 mm Hg or less.

25. Open the valve allow helium to flow into the DSC cavity to pressurize the DSC to 2.5 ± 1 psig in accordance with Section 3.1.2 of the Technical Specification [9-5] limits.

26. Close the valves on the helium source.

27. Decontaminate as necessary.

9.1.4 DSC Sealing Operations

CAUTION: During the performance of steps listed in Section 9.1.4, monitor the TC/DSC annulus water level and replenish, as necessary, to maintain cooling.

1. Disconnect the VDS from the DSC. Seal weld the prefabricated cover plate over the drain port, inject helium into blind space just prior to completing welding, and perform root and final dye penetrant weld examinations in accordance with Section 4.4.4 of the Technical Specification [9-5] requirements.

2. Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate (OTCP) and place the OTCP with the automatic welding system onto the DSC. Verify proper fit up of the OTCP with the DSC shell.

3. Tack weld the OTCP to the DSC shell. Place the OTCP weld root pass.

4. Helium leak test the inner top cover plate and vent/drain port plug/plate welds using the leak test port in the OTCP in accordance with Section 4.4.4 of the Technical Specification [9-5] limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [9-3]. Alternatively, this can be done with a test head prior to installing and welding the root pass for the OTCP.

5. If a leak is found, remove the OTCP root pass, the drain plugs and repair the ITCP welds. Repeat procedure steps from Section 9.1.3 Step 17.

6. Perform dye penetrant examination of the root pass weld. Weld out the OTCP to the DSC shell. Perform root and multilayer dye penetrant examination in accordance with Section 4.4.4 of the Technical Specifications [9-5].

7. Seal weld the prefabricated plug over the outer cover plate test port and perform root and final dye penetrant weld examinations.

8. Remove the automatic welding machine from the DSC.

9. Open the TC drain port valve and drain the water from the TC/DSC annulus.

   Note: The time limit for Transfer Operations, if any, starts from initiation of this step.

CAUTION: Monitor the applicable time limits of Section 3.1.3 of the Technical Specifications [9-5] until the completion of DSC transfer Step 16 of Section 9.1.6.

10. Rig the TC top cover plate and lower the cover plate onto the TC.

    Note: To meet weight limits the aluminum TC top cover plate may be used during the lift to the trailer.

11. Bolt the TC cover plate into place, tightening the bolts to the required torque in a star pattern.

12. Check the exterior of the TC to verify that it meets the limits specified in Section 3.3.1 of the Technical Specifications [9-5] for surface contamination.

9.1.5 <u>TC Downending and Transfer to ISFSI</u>

Note: Alternate procedure for downending of transfer cask: Some plants have limited floor hatch openings above the cask/trailer/skid, or other conditions which limit crane travel in the direction that would be needed in order to downend the TC with the trailer/skid in a stationary position. For these situations, alternate procedures are to be developed on a plant-specific basis, with detailed steps for downending.

1. Re-attach the TC lifting yoke to the crane hook, as necessary. Ready the transfer trailer and TC support skid for service.

CAUTION: Perform a functional test of the air circulation system, including the blowers, generators, and power cords, etc. per Section 3.1.3 of the Technical Specifications [9-5].

2. Move the scaffolding away from the TC as necessary. Engage the lifting yoke and lift the TC over the TC support skid on the transfer trailer.

3. The transfer trailer should be positioned so that TC support skid is accessible to the crane with the trailer supported on the vertical jacks.

4. Position the TC lower trunnions onto the transfer trailer trunnions.

5. Move the crane forward while simultaneously lowering the TC until the TC upper trunnions are just above the support skid upper trunnion pillow blocks.

6. Inspect the positioning of the TC to ensure that the TC and trunnion pillow blocks are properly aligned.

7. Lower the TC onto the skid until the weight of the TC is distributed to the trunnion pillow blocks.

8. Inspect the trunnions to ensure that they are properly seated onto the skid.

9. If used, remove the aluminum TC top cover plate and replace with the steel TC top cover plate.

10. Remove the bottom ram access cover plate from the TC and install the air flow adaptor, if a time limit for transfer is specified in Section 3.1.3 of the Technical Specifications [9-5].

9.1.6 DSC Transfer to the EOS-HSM

1. Prior to transporting the TC to the ISFSI, verify that the wind deflectors are in place, if needed per Section 5.5 of the Technical Specifications [9-5], remove the HSM door, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

CAUTION: The insides of empty modules have the potential for high does rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

3. Verify specified lubrication of the DSC support structure rails.

4. Transport the TC from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

5. Once at the ISFSI, position the transfer trailer to within a few feet of the HSM.

6. Check the position of the trailer to ensure the centerline of the HSM and TC approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

7. Unbolt and remove the TC top cover plate.

8. Back the trailer to within a few inches of the HSM, set the trailer brakes and disengage the tractor, if applicable. Extend the transfer trailer vertical jacks.

9. Use the skid positioning system to bring the TC into approximate vertical and horizontal alignment with the HSM. Using alignment equipment and the alignment marks on the TC and the HSM, adjust the position of the TC until it is properly aligned with the HSM.

10. Using the skid positioning system, fully insert the TC into the HSM access opening docking collar.

11. Secure the TC/skid to the front wall embedments of the HSM using the TC restraints.

12. After the TC is docked with the HSM, verify the alignment of the TC using the alignment equipment.

13. Position the ram behind the TC in approximate horizontal alignment with the TC and level the ram. Remove either the bottom ram access cover plate or the airflow adaptor. Extend the ram through the bottom TC opening into the DSC grapple ring.

14. Operate the ram grapple and engage the grapple arms with the DSC grapple ring.

15. Recheck all alignment marks and ready all systems for DSC transfer.

16. Activate the ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

    Note: The time limit for transfer operations, if any, starts with the initiation of the TC/DSC annulus water draining described in Step 9 of Section 9.1.4 and ends when the DSC is fully inserted into the HSM.

CAUTION: Verify that the applicable time limits for transfer operations of Section 3.1.3 of the Technical Specifications [9-5] are met.

17. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

18. Retract and disengage the ram system from the TC and move it clear of the TC. Remove the TC restraints from the HSM.

19. Using the skid positioning system, disengage the TC from the HSM access opening.

20. Install the DSC axial restraint through the HSM door opening.

21. The trailer can be moved, as necessary, to install the HSM door. Install the HSM door and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Section 5.1.2 of the Technical Specifications [9-5].

22. Replace the TC top cover plate. Secure the skid to the trailer, retract the vertical jacks, and disconnect the skid positioning system.

23. Move the trailer and TC to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

24. Close and lock the ISFSI access gate and activate the ISFSI security measures.

9.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform a daily visual surveillance of the EOS-HSM air inlets and outlets (bird screens) to verify that no debris is obstructing the HSM vents in accordance with Section 5.1.3(a) of the Technical Specification [9-5] requirements, or, perform a temperature measurement for each EOS-HSM in accordance with Section 5.1.3(b) of the Technical Specification [9-5] requirements.

9.2 Procedures for Unloading the DSC

The following section outlines the procedures for retrieving the DSC from the HSM and for removing the FAs from the DSC.

9.2.1 DSC Retrieval from the HSM

1. Ready the TC, transfer trailer, and support skid for service. Fill the TC liquid neutron shield and remove the top cover plate from the TC. Transport the trailer to the HSM.

   Note: Verify that a TC spacer of appropriate height is placed inside the TC to provide the correct airflow and interface at the top of the TC during cutting and unloading operations for DSCs that are shorter than the TC cavity length.

2. Remove the HSM door and the DSC axial restraint. Position the transfer trailer to within a few feet of the HSM.

3. Check the position of the trailer to ensure the centerline of the HSM and TC approximately coincide. If the trailer is not properly oriented, reposition the trailer as necessary.

CAUTION: High dose rates are expected in the HSM cavity after removal of the HSM door. Proper ALARA practices should be followed.

4. Back the TC to within a few inches of the HSM, set the trailer brakes and disengage the tractor, if applicable. Extend the transfer trailer vertical jacks.

5. Use the skid positioning system to bring the TC into approximate vertical and horizontal alignment with the HSM. Using alignment equipment and the alignment marks on the TC and the HSM, adjust the position of the TC until it is properly aligned with the HSM.

6. Using the skid positioning system, fully insert the TC into the HSM access opening docking collar.

7. Secure the TC/skid to the front wall embedments of the HSM using the TC restraints.

8. After the TC is docked with the HSM, verify the alignment of the TC using the alignment equipment.

9. Position the ram behind the TC in approximate horizontal alignment with the TC and level the ram. Remove the bottom ram access cover plate or the air flow adaptor, if installed. Extend the ram through the TC into the HSM until it is inserted in the DSC grapple ring.

10. Operate the ram grapple and engage the grapple arms with the DSC grapple ring.

11. Recheck all alignment marks and ready all systems for DSC transfer.

CAUTION: The time limits for the unloading of the DSC should be determined using the heat loads at the time of the unloading operation and the methodology presented in Sections 4.5 and 4.6 before pulling the DSC out of the HSM.

12. Activate the ram to pull the DSC into the transfer TC.

13. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

14. Retract and disengage the ram system from the TC and move it clear of the TC. Remove the TC restraints from the HSM.

15. Using the skid positioning system, disengage the TC from the HSM access opening.

16. Bolt the TC cover plate into place, tightening the bolts to the required torque in a star pattern.

17. Retract the vertical jacks and disconnect the skid positioning system.

18. Ready the trailer for transfer.

19. Replace the HSM door and DSC axial restraint on the HSM.

9.2.2 Removal of Fuel from the DSC

If it is necessary to remove fuel from the DSC, it can be removed in a dry transfer facility or the initial fuel loading sequence can be reversed and the plant's spent fuel pool utilized.

Procedures for wet unloading of the DSC are presented here. Dry unloading procedures are essentially identical up to the removal of the DSC vent plug and drain port cover.

CAUTION: Monitor the applicable time limits determined for the unloading operation in Step 11, Section 9.2.1 above, until the TC/DSC Annulus is filled with water in Step 12 of Section 9.2.2. If the time limits for unloading cannot be met, initiate forced cooling.

1. Transfer the loaded TC from the ISFSI to inside the plant's fuel or reactor building along the designated transfer route.

2. Position and ready the trailer for access by the crane. The trailer is supported on the vertical jacks.

3. If required to meet crane weight limits, replace steel TC cover plate with the aluminum TC cover plate.

4. Attach the TC lifting yoke to the crane hook. Then engage the TC lifting yoke with the trunnions of the TC on the transfer trailer.

5. Verify that the yoke lifting arms are properly aligned and engaged onto the TC trunnions.

6. Lift the TC approximately one inch off the saddle supports. Verify that the yoke lifting arms are properly positioned on the trunnions.

7. Move the crane in a horizontal motion while simultaneously raising the crane hook vertically and lift the TC off the trailer. Move the TC to the TC designated area.

8. Lower the TC into the TC staging area in the vertical position.

9. Clean the TC, if needed, to remove any dirt that may have accumulated on the TC during the DSC loading and transfer operations.

10. Place scaffolding around the TC.

11. Unbolt the TC cover plate and remove it.

12. Install temporary shielding to reduce personnel exposure as required. Fill the TC/DSC annulus with clean water. Place an inflatable seal into the upper TC liner recess and seal the TC\DSC annulus by pressurizing the seal with compressed air.

13. Locate the drain and vent port using the indications on the OTCP. Place a portable drill press on the top of the DSC. Align the drill over the drain port.

14. Cut or drill a hole through the OTCP to expose the drain port on the ITCP. Remove the drain port cover plate with an annular hole cutter. Repeat for the vent plug.

CAUTION: Radiation dose rates are expected to be high at the vent and drain locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

15. Obtain a sample of the DSC atmosphere. Confirm acceptable hydrogen concentration and check for presence of fission gas indicative of degraded fuel cladding.

16. If degraded fuel is suspected, additional measures appropriate for the specific conditions are to be planned, reviewed, and implemented to minimize exposures to workers and radiological releases to the environment.

17. Verify that the boron content of the fill water conforms to Section 3.2.1 of the Technical Specifications [9-5], if applicable. Fill the DSC with water from the fuel pool or equivalent source through the drain port with the vent port open. The vented cavity gas may include steam, water, and radioactive material, and should be routed accordingly. Monitor the DSC vent pressure and regulate the water fill rate to ensure that the pressure does not exceed 15 psig.

18. Per Section 5.4 of the Technical Specifications [9-5], provide for continuous hydrogen monitoring of the DSC cavity atmosphere during all subsequent cutting operations to ensure that hydrogen concentration does not exceed 2.4%. Purge with helium as necessary to maintain the hydrogen concentration below this limit before resuming cutting operations.

CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% (60.0% of flammability limit of 4.0%) [9-1 and 9-2]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with approximately 2-3 psig helium to reduce the hydrogen concentration safely below the 2.4% limit.

19. Provide suitable protection for the TC during cutting operations.

20. Using a suitable method, such as mechanical cutting, remove the weld of the OTCP to the DSC shell.

21. Remove the OTCP.

22. Remove the weld of the inner top cover/shield plug to the shell in the same manner as the OTCP. Do not remove the inner top cover/shield plug at this time unless the removal is being done remotely in a dry transfer system.

23. Remove any remaining excess material on the inside shell surface by grinding.

24. Clean the TC surface of dirt and any debris that may be on the TC surface as a result of the weld removal operation.

    a. EOS-TC125 or TC135: The neutron shield may be drained if it is necessary to meet the crane capacity when the loaded TC is placed into the pool.

    b. EOS-TC108: The neutron shield tank may be drained or removed from the TC108 and staged in an appropriate location.

25. Position the TC lifting yoke and engage the TC lifting trunnions, install eyebolts or other lifting attachment(s) into the shield plug, and connect the rigging cables to the eyebolts/lifting attachment(s).

26. Move the scaffolding away from the TC as necessary.

27. Lift the TC just far enough to allow the weight of the TC to be distributed onto the yoke lifting arms. Verify that the lifting arms are properly positioned on the trunnions.

28. Optionally, secure a sheet of suitable material to the bottom of the TC to minimize the potential for ground-in contamination. This may also be done prior to initial placement of the TC in the designated area.

29. Prior to the TC being lifted into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the TC/DSC volume, as necessary.

30. Position the TC over the TC loading area in the spent fuel pool.

31. Lower the TC into the pool. As the transfer TC is being lowered, the exterior surface of the TC should be sprayed with clean water.

32. Lower the TC into the fuel pool leaving the top surface of the TC above the surface of the pool water. Verify correct connections of the annulus seal and annulus/neutron shield tank, if used.

33. Disengage the lifting yoke from the TC and lift the shield plug from the DSC.

34. Remove the fuel from the DSC.

    Note: Special attention should be given to unloading the FAs (especially for boiling water reactor (BWR fuel) to wait until any loose particles have settled and slowly move the FAs to minimize fuel crud dispersion in the spent fuel pool. The dry TC reflood process, during unloading of BWR fuel, has the potential to disperse crud into the pool and become airborne, creating airborne exposure and personnel contamination hazards.

9.3 References

9-1 U.S. Nuclear Regulatory Commission, Office of the Nuclear Material Safety and Safeguards, "Safety Evaluation of VECTRA Technologies' Response to Nuclear Regulatory Commission Bulletin 96-04 For the NUHOMS®-24P and NUHOMS®–7P.

9-2 U.S. Nuclear Regulatory Commission Bulletin 96-04, "Chemical, Galvanic or Other Reactions in Spent Fuel Storage and Transportation Casks," July 5, 1996.

9-3 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 2006.

9-4 U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61 "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Container," February 1989.

9-5 Proposed CoC 1042 Appendix A, NUHOMS® EOS System Generic Technical Specifications, Amendment 0.

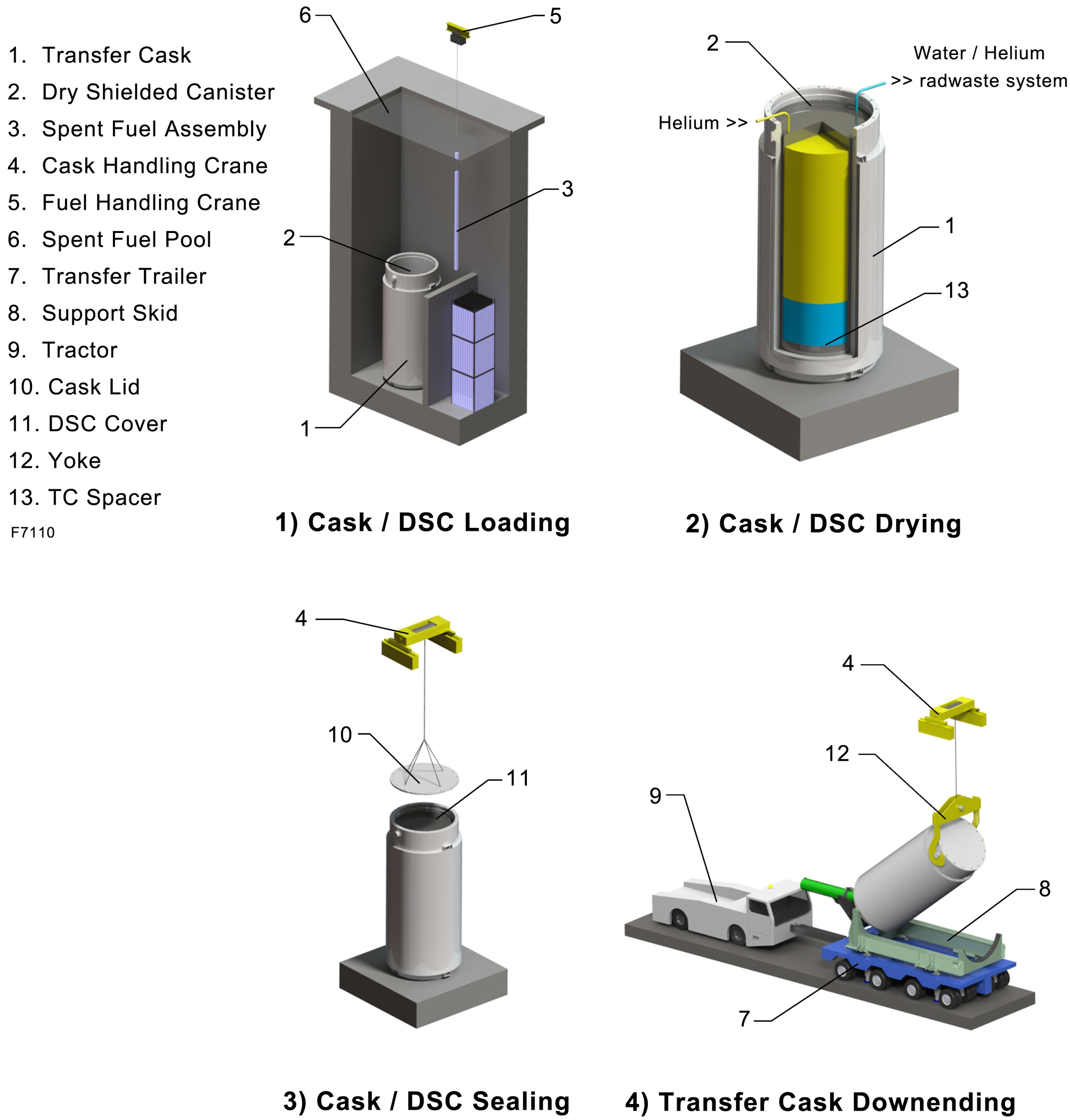

Figure 9-1
NUHOMS® EOS System Loading Operations
2 Pages

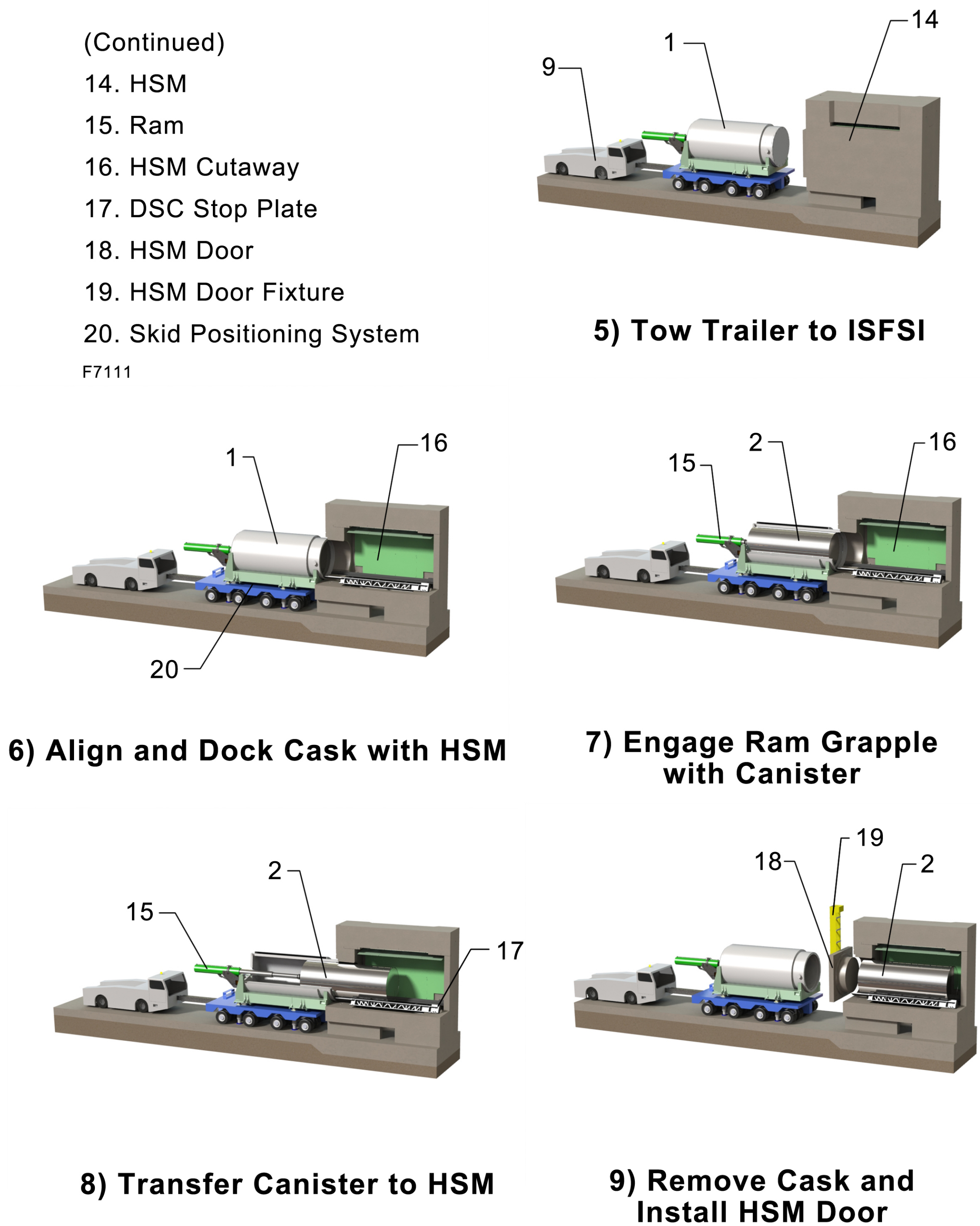

Figure 9-1
NUHOMS® EOS System Loading Operations
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CHAPTER 10
ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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10. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

This chapter specifies the acceptance testing and maintenance program for important-to-safety (ITS) components of the NUHOMS® EOS System, which is the dry shielded canister (DSC), horizontal storage module (HSM), and transfer cask (TC). The testing and maintenance of the TC lift yoke are governed by 10 CFR Part 50 heavy load regulations, and are not covered here.

10.1 Acceptance Tests

10.1.1 Structural and Pressure Tests

10.1.1.1 DSC

The DSC confinement boundary is fabricated, inspected, and tested in accordance with American Society of Mechanical Engineers (ASME) Code Subsection NB [10-1] with alternatives specified in Section 4.4.4 of the Technical Specifications [10-32]. The shell and inner bottom cover plate assembly is pneumatically tested during fabrication in accordance with ASME Article NB-6300. The inner top cover plate and its weld to the shell are pneumatically tested in the field, in accordance with the above-cited ASME Code alternatives. Per ASME Article NB-6300, the pneumatic test pressure shall be 1.1 times the design pressure, which results in a test pressure of 16.5 psig. For conservatism, a test pressure of 18.0 psig is selected to bound potential conditions for transportation under 10 CFR Part 71.

DSC	Normal Pressure (psig)	Design Pressure (psig)	Test Pressure (psig)
EOS37PTH	10.5	15(1)	18.0
EOS89BTH	10.8	15(1)	18.0

(1) A pressure of 20 psig is used for structural evaluations under normal and off-normal conditions as noted in Section 3.9.1.2.7.3.

Mechanical properties of materials are tested in accordance with the American Society for Testing and Materials (ASTM), ASME, AMS, or other material specification called out on the drawings in Chapter 1. Weld procedures and welders are qualified in accordance with ASME Code Section IX. Additional material and welding requirements of ASME Code Articles NB-2000 and NB-4000 apply to the confinement boundary unless an ASME Code alternative governs.

Acceptance testing for the high-strength low-alloy steel used in the DSC baskets is specified in 10.1.7. There is no welding on the baskets.

10.1.1.2 HSM

Concrete mix design, placement, and testing are performed in accordance with ACI-318 [10-2]. The minimum 28-day compressive strength is 5000 psi if controls are placed on the aggregate type or coefficient of thermal expansion as described in Section 8.2.1.3. If the alternative described in that Section is used, the minimum is 7000 psi. In accordance with American Concrete Institute (ACI) 349 Appendix E, paragraph E.4.3 [10-3], compressive testing of the concrete mix design for the base, roof, and doors is conducted after heating the test cylinders at 500 °F prior to testing. See Sections 4.4.4 and 5.3 of the Technical Specifications.

The reinforcing steel, ITS fasteners, and steel for the door and the DSC support structure are tested for mechanical properties in accordance with the governing specifications called out on the drawings in Chapter 1.

Weld procedures and welders for the DSC support structure are qualified in accordance with ASME Code Section IX or American Welding Society (AWS) D1.1 [10-4].

10.1.1.3 Transfer Cask

The TC structural assembly welds are performed in accordance with ASME Code Section IX, and examined by magnetic particle examination (MT) to the acceptance standards of ASME Code Section III, Subarticle NF-5340. The upper trunnions are load tested at fabrication to three times the design load, and inspected afterwards in accordance with ANSI N14.6 [10-5].

The liquid neutron shield shell is hydrostatically tested to 1.25 times the pressure relief valve setting shown on the drawings in Chapter 1.

Transfer Cask	Test Load, Each Upper Trunnion (ton)	Neutron Shield Test Pressure (psig)
108 ton	162	35
125 ton	187.5	31.25
135 ton	202.5	31.25

Mechanical properties of materials are tested in accordance with the ASTM, ASME, or other material specification called out on the drawings in Chapter 1. Weld procedures and welders are qualified in accordance with ASME Code Section IX.

10.1.2 Leak Tests

Confinement welds in the DSC shell and bottom are leak tested during fabrication to the ANSI N14.5 [10-6] leaktight acceptance criterion 1×10^{-7} ref cm^3/s. Personnel performing the leak test are qualified in accordance with American Society for Nondestructive Testing (ASNT) SNT-TC-1A [10-7]. The DSC inner top cover plate, port covers, and their welds are leak-tested in the field to the same acceptance criterion after the fuel assemblies are loaded. Leak testing procedures follow ASME Code Section V, Article 10, Appendix IX, ASTM E1603 [10-8], or equivalent standard that provides the sensitivity required by ANSI N14.5.

10.1.3 Visual Inspection and Non-Destructive Examinations

Visual inspections are performed at the fabricator's facility to ensure that the DSC, the HSM, and the TC conform to the drawings and specifications. The visual inspections include weld, dimensional, surface finish, and cleanliness inspections. Visual inspections specified by codes applicable to a component are performed in accordance with the requirements and acceptance criteria of those codes. Requirements specific to each component follow.

10.1.3.1 DSC

Non-destructive examination (NDE) requirements for welds are specified on the drawings provided in Chapter 1. The confinement welds on the DSC are inspected in accordance with ASME Boiler and Pressure Vessel Code Subsection NB, including alternatives to ASME Code cited in Section 4.4.4 of the Technical Specifications.

Non-destructive examination personnel are qualified in accordance with SNT-TC-1A.

10.1.3.2 HSM

Reinforcing steel placement is inspected in accordance with ACI 117 [10-9] and cured concrete is visually inspected in accordance with ACI 311.4R [10-10].

Weld inspections and inspector qualifications conform to AWS D1.1.

10.1.3.3 Transfer Cask

The load-bearing welds for the EOS-TC structural body are inspected by MT as specified on the drawings in Chapter 1, with acceptance criteria of ASME Subarticle NF-5340. Non-destructive examination personnel are qualified in accordance with SNT-TC-1A.

10.1.4 Shielding Tests

10.1.4.1 DSC

The shielding performance of the top and bottom shield plugs and cover plates of the DSC is verified by their material certifications and dimensional inspections. No further testing is required.

10.1.4.2 HSM

The HSM concrete is tested in accordance with ASTM C138 [10-11] to verify a minimum density of 140 lb/ft^3.

10.1.4.3 Transfer Cask

The TC lead shielding is installed as lead sheet or interlocking lead bricks, rather than being cast in place. The neutron shield material in the lid and bottom end is also installed as solid sheets of borated high density polyethylene. Shielding performance is verified by material certification and dimensional inspection.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary.

10.1.5 Neutron Absorber Tests

The neutron absorber used for criticality control in the DSC baskets may consist of one of the following materials:

- Boron carbide/aluminum metal matrix composite (MMC)
- BORAL®

The safety analyses do not rely upon the tensile strength of these materials. The neutron absorber's design function relies only on the B-10 areal density (mass of B-10 per unit area) and the thermal conductivity in the plane of the sheet. The minimum specified B-10 areal density of these materials is specified on the DSC basket drawings in Chapter 1. The criticality calculations in Chapter 7 use 90% of the minimum for MMC and 75% for BORAL.

10.1.5.1 MMC Specification and Acceptance Testing

Metal matrix composites are defined for the purpose of the NUHOMS® EOS System as a near ≥ 97% of density composite of fine boron carbide particles in an aluminum alloy matrix, with or without an aluminum skin. The d50[1] size of the boron carbide feed material is 80 microns with no more than 10% over 100 microns. The maximum boron carbide content in the MMC is 50 volume %.

Metal matrix composites may be supplied by any manufacturer subject to qualification testing and control of key manufacturing processes in accordance with ASTM C1671 [10-12], as described in Sections 10.1.5.3 and 10.1.5.4.

[1] Median particle size by volume, or equivalently, mass.

B10 Areal Density Acceptance Testing of MMC

B-10 areal density is verified by neutron attenuation testing per ASTM E2971[10-13] or by chemical and spectrometric analysis per ASTM D3171 [10-14] and ASTM C791 [10-15], benchmarked against neutron attenuation testing. The areal density testing is performed on coupons taken adjacent to the finished panels. The sampling is either systematic or random, with a sampling rate sufficient to yield at least twenty-five coupons per DSC. For each lot of material, the lower tolerance limit at 95% probability and 95% confidence is determined in accordance with ASTM C1671. This lower tolerance limit must equal or exceed the minimum B10 areal density specified on the drawings in Chapter 1.

Dimensional Inspections of MMC

All MMC finished panels are inspected for length, width and thickness.

Thermal Conductivity Acceptance Testing of MMC

Acceptance testing for thermal conductivity is performed at room temperature according to ASTM E1225 [10-16], ASTM E1461 [10-17], or equivalent method, on coupons taken adjacent to the finished panels. The minimum acceptable thermal conductivity is [

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Thermal conductivity acceptance testing is conducted until sufficient data are collected for a single supplier, manufacturing method, and material composition to validate the consistent performance of that material.

Visual Inspections of MMC

Finished panels are visually inspected to verify that there are no blisters, cracks, or surface peeling. Clad MMCs are also inspected to verify that there is no exposed core on the face of the sheet or solid aluminum at the edge of the sheet. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

10.1.5.2 BORAL Specification and Acceptance Testing

BORAL consists of a core of aluminum and boron carbide powders between two outer layers of aluminum. It is distinguished from MMCs by the interconnected porosity of the core, which is exposed at the edges of the sheet, the larger size of the boron carbide powder, and the higher boron carbide content. At least 80% by weight of the boron carbide particles in BORAL are smaller than 200 microns. The boron carbide content is limited to 67% of the core by weight.

Because there is long experience with BORAL, and because only 75% of the minimum specified areal density is used in criticality calculations, BORAL is not subject to qualification testing.

B-10 Areal Density Acceptance Testing of BORAL

B-10 areal density will be verified by chemical analysis and by certification of the B-10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final panels produced from that ingot.

Dimensional Inspections of BORAL

All BORAL finished panels are inspected for length, width and thickness.

Thermal Conductivity Testing of BORAL

Acceptance testing for thermal conductivity parallel to the plane of the sheet is performed at room temperature according to ASTM E1225, or equivalent method, on coupons taken adjacent to the finished panels. The minimum acceptable thermal conductivity is [

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Thermal conductivity acceptance testing is conducted until sufficient data are collected to validate the consistent performance of BORAL.

Visual Inspection of BORAL

Visual inspection verifies that there are no cracks, blisters, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

10.1.5.3 Qualification Testing and Key Process Control for MMCs

Qualification testing and key process controls follow the guidance of ASTM C1671 and ISG-23 [10-18].

Applicability and Scope

Each MMC of a given manufacturer, production method, or composition is subject to qualification to verify that the product will perform its design functions in the DSC internal environments. Key process controls are identified so that the production material is equivalent to or better than the qualification test material.

Service Environment Durability

Accelerated radiation damage testing is not required. Such testing has already been performed on MMCs, and the results confirm what would be expected of MMCs due to their composition. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2, nor are they susceptible to gamma radiation damage.

Thermal damage and corrosion (hydrogen generation) tests are performed unless such tests on materials of the same chemical composition have already been performed and found acceptable.

Mechanical Integrity Testing

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage system.

At least three samples, one each from approximately the two ends and middle of the qualification material run are subject to:

- Room temperature tensile testing (ASTM B557 [10-19]) demonstrating that the material has the following tensile properties:
  - Minimum yield strength, 0.2% offset: 1.5 ksi
  - Minimum ultimate strength: 5 ksi
  - Minimum elongation in 2 inches: 0.5%

As an alternative to the elongation requirement, ductility can be demonstrated by bend testing per ASTM E290 [10-20]. The radius of the pin or mandrel is no greater than three times the material thickness, and the material is bent at least 90 degrees without complete fracture.

- Testing to verify more than 97% of theoretical density (ASTM B311 [10-29], or microscopic examination verifying less than 3% by volume isolated pores).
- Testing to verify maximum interconnected porosity 0.5 volume % (ASTM B963 [10-30] or microscopic examination).

Clad MMCs are subject to additional mechanical integrity testing to demonstrate that they are not subject to water intrusion and subsequent formation of steam blisters during vacuum drying [10-21].

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Test for Uniform B-10 Distribution

Uniformity of the boron distribution is verified by measurement of B-10 areal density by neutron attenuation per ASTM E2971, or the boron carbide weight fraction per ASTM D3171, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean.

Qualification Report

Qualification report is prepared by or is subject to approval by the Certificate Holder.

10.1.5.4 Manufacturing Key Process Controls for MMCs

Applicability and Scope

Key processing changes are subject to qualification prior to use of the material produced by the revised process. The Certificate Holder determines whether a complete or partial re-qualification program per Section 10.1.5.3 is required, depending on the characteristics of the material that could be affected by the process change.

Definition of Key Process Changes

Key process changes are those that could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce thermal durability, or reduce the mechanical strength or ductility of the MMC.

The following are examples of such changes.

Potential change effect	Example
Reduction of the yield or ultimate strength or the elongation	Increase in nominal boron carbide content over that previously qualified
Adverse effect on the uniformity of boron carbide distribution at the microscopic scale	Increase in the boron carbide particle size
Adverse effect on the uniformity of boron carbide distribution at the macroscopic level	Change in the blending process

Reduced density of the final product	Change in the method of billet production or thermo-mechanical processing to plate
Adverse reaction between the boron carbide and the matrix alloy under normal and off-normal service temperatures	Change in the matrix alloy
Lower corrosion resistance or higher rate of hydrogen generation	Change in the matrix alloy

Identification and Control of Key Process Changes

The manufacturer provides the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer prepare a written list of key process changes that cannot be made without prior approval of the Certificate Holder.

10.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit.

10.1.7 Low Alloy High Strength Steel for Basket Structure

CAUTION

Portions of Section 10.1.7 are incorporated by reference into the NUHOMS® EOS System Generic Technical Specifications 4.2.3 and shall not be deleted or altered in any way without approval from the NRC. The text of these portions is shown in bold type to distinguish it from other sections.

The structural steel for the basket shall be a low alloy high strength steel meeting the following requirements:

a) All requirements of ASTM A829, except acceptable chemical compositions are not limited to those in ASTM A829 Table 1.
b) Weld repair is not permitted.
c) The material with its specified heat treatment is qualified prior to first use by testing multiple lots with the following acceptance criteria:

   i. Fracture toughness value [ ] with 95% confidence
   ii. Yield and tensile strengths ≥ the values in Table 8-10 with 95% confidence

d) [

]

e) Acceptance tensile testing is performed at room temperature per ASTM A370 twice per heat and heat treatment lot.

f) Acceptance test criteria for yield and ultimate strength for ASTM A829 Gr 4130 are:

Minimum yield strength (ksi)	**103.6**
Minimum ultimate strength (ksi)	**123.1**

The acceptance criteria for testing at room temperature for other HSLA steels shall be identified using a similar method described in [10-31].

g) [

] The dynamic tear testing is based on Regulatory Guide 7.11, Table 4 [10-28].

[

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10.1.8 Cask Identification

Each DSC, HSM, and TC is marked with a model number, serial number, and empty weight per 10 CFR 72.236(k).

10.2 Maintenance Program

There are no maintenance requirements for the DSC for the initial licensed period of storage. This section addresses maintenance for the HSM and the TC.

10.2.1 Inspection

10.2.1.1 Transfer Cask Inspections

The following inspections are performed prior to each loading campaign using the TC inspection of the cask exterior for cracks, dents, gouges, tears, or damaged bearing surfaces.

- Visual inspection of all threaded parts and bolts for burrs, chafing, distortion or other damage.
- Operational check of all quick-connect fittings.
- Visual inspection of the interior surface of the cask for any indications of excessive wear to bearing surfaces.
- Visual inspection of neutron shield jacket.

The following inspections and tests are performed on an annual basis:

- Leak testing of the cask cavity quick-connect fittings and ram penetration seal.
- Liquid penetrant testing (PT) examination of the trunnions welds.
- Visual inspection of all painted surfaces for damage to paint and for corrosion.

10.2.1.2 HSM Inspections

The HSM is inspected for blockage of the inlet vents by either direct visual inspection, or by temperature monitoring as specified in accordance with Section 5.1.3 of the Technical Specifications.

10.2.2 Tests

10.2.2.1 Transfer Cask Tests

There are no tests required for the transfer cask after completion of fabrication during the initial license period.

10.2.2.2 HSM Tests

There are no HSM tests during the initial license period.

10.3 Repair, Replacement, and Maintenance

10.3.1 Transfer Cask Repair, Replacement, and Maintenance

Typical repair, replacement, and maintenance activities include:

- Paint damage is repaired by surface preparation and application of the original paint system.
- Raised burrs from wear are removed by light abrasives, followed by paint repair as necessary.
- Wear damage that violates a design minimum dimension is corrected by weld repair, followed by paint repair as necessary.
- Bolts, screw thread inserts, quick connect fittings, and the ram access port o-ring are replaced as needed.
- TC 108 removable neutron shield latches, hinges and hinges are replaced as needed.
- Bolt threads and removable neutron shield hinges are lubricated as needed.
- Rust on unpainted wear surfaces at the upper trunnions and lower sockets can be limited by temporary coatings when not in use.
- The neutron shield is filled with potable water prior to each loading campaign, and drained afterwards.
    - [

    ]
- Abrasive pads such as Scotch-Brite™ should not be used to decontaminate painted surfaces. Abrasive pads degrade the paint's ease of decontamination.

10.3.2 HSM Repair, Replacement, and Maintenance

Cracking or other damage noted by visual inspection is generally entered into the licensee's corrective action program and repair or maintenance actions are developed with the help of AREVA Inc. Examples of typical exterior surface repairs are:

- Cracks below acceptable width – verify soundness of concrete by rebound hammer, and apply concrete sealant.
- Cracks above acceptable width – Fill crack with epoxy or cement –based grout, then apply concrete sealant.
- Damage greater than cracking (spalling, corner breaks) – repair using 5000 psi grout and bonding agent.

10.3.3 Maintenance of Thermal Monitoring System

In lieu of visual inspection for vent blockage, the licensee has the option to monitor the temperature by thermocouples or resistor temperature detectors inserted into wells embedded in the HSM roof. The licensee is responsible for maintenance and calibration of this temperature monitoring instrumentation and data collection.

10.4 References

10-1 American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," 2010 Edition through 2011 Addenda

10-2 ACI 318-08, "Building Code Requirements for Structural Concrete and Commentary," American Concrete Institute, Detroit, MI.

10-3 ACI 349-06, "Code Requirements for Nuclear Safety Related Structures," American Concrete Institute, Detroit, MI.

10-4 American Welding Society, AWS D1.1/D1.1M, "Structural Welding Code – Steel."

10-5 ANSI N14.6, "Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More," American National Standards Institute, 1993.

10-6 ANSI N14.5, "Leakage Tests on Packages for Shipment of Radioactive Materials," American National Standards Institute, 1997.

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CHAPTER 11
RADIATION PROTECTION

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11. RADIATION PROTECTION

This chapter describes the design features of the NUHOMS® EOS System that maintain radiation exposure to site personnel as low as reasonably achievable (ALARA), as well as minimize exposure to the public. An occupational dose assessment for operation of the NUHOMS® EOS System is provided. Radiation exposures to offsite individuals are also computed for both normal and accident conditions of an independent spent fuel storage installation (ISFSI). This chapter provides an example of how to demonstrate compliance with the relevant radiological requirements of 10 CFR Part 20 [11-1], 10 CFR Part 72 [11-2], and 40 CFR Part 190 [11-3]. Each user must perform site-specific calculations to account for the actual layout of the EOS horizontal storage modules (EOS-HSMs) and fuel source.

11.1 Radiation Protection Design Features

The NUHOMS® EOS System has design features that ensure a high degree of integrity for the confinement of radioactive materials and reduction of direct radiation exposures during storage. These features are described in Section 11.4.2.

11.2 Occupational Dose Assessment

This section provides estimates of occupational dose for typical EOS transfer cask (EOS-TC) and ISFSI loading operations. Assumed annual occupancy times, including the anticipated maximum total hours per year for any individual, and total person-hours per year for all personnel for each radiation area during normal operation and anticipated operational occurrences, will be evaluated by the licensee in a 10 CFR 72.212 evaluation to address the site-specific ISFSI layout, inspection, and maintenance requirements. In addition, the estimated annual collective doses associated with loading operations will be addressed by the licensee in a 10 CFR 72.212 evaluation.

11.2.1 EOS-DSC Loading, Transfer, and Storage Operations

The near-field dose rates are computed for the EOS-TC108 and EOS-TC125/135 in Chapter 6. Dose rates are computed at the surface (R8), 30 cm (R9), 100 cm (R10), and 300 cm (R11) at various axial locations. The dose rate locations (DRLs) are illustrated in Figure 11-1. EOS-TC dose rates are computed for three generalized operational configurations: loading/decontamination, welding/drying, and downending/transfer.

The average distance from the EOS-TC for a given operation takes into account that the operator may be in contact with the EOS-TC, but this duration will be limited. For draining activities and vacuum drying, the attachment of fittings will take place closer to the EOS-TC than the operation of the pumps. For decontamination activities, although operators could be near the EOS-TC for some activities, other activities of the operation could be performed from farther away. For this reason, the average distances used for these operations range from 30 cm to 300 cm from the EOS-TC.

The operator's hands may be in a high dose rate location momentarily, for example, when connecting fittings at the ports. This does not translate into a whole-body exposure and, therefore, these localized streaming effects are not considered in this case. For operations near the top end of the EOS-DSC, most of the work will take place around the perimeter and a smaller portion will take place directly over the shielded inner top cover.

Based on the considerations outlined above, the region around the EOS-TC is divided into 10 general DRLs appropriate for estimation of worker whole-body exposure (DRL1 through DRL10). These DRLs and the average dose rate at each location are summarized in Table 11-1. For example, location "DRL1/Decon." is the average dose rate for the loading/decontamination configuration in axial segment A1 through A18 at radial location R11 (300 cm from the EOS-TC). The EOS-HSM dose rate reported in Table 11-1 is the average dose rate on the front surface of an EOS-HSM that has an adjacent EOS-HSM on each side. This dose rate is also obtained from Chapter 6.

The estimated occupational exposures to ISFSI personnel during loading, transfer, and storage operations using the EOS-TC108 (time and number of workers may vary depending on individual ISFSI practices) are provided in Table 11-2 and Table 11-3 for the EOS-37PTH DSC and EOS-89BTH DSC, respectively. Similar operations for the EOS-TC125/135 are provided in Table 11-4 and Table 11-5. The task times, number of personnel required, and total doses are listed in these tables. The total exposure results are as follows:

- EOS-TC108 with EOS-37PTH DSC: 3361 person-mrem (~3.4 person-rem)
- EOS-TC108 with EOS-89BTH DSC: 3270 person-mrem (~3.3 person-rem)
- EOS-TC125/135 with EOS-37PTH DSC: 2081 person-mrem (~2.1 person-rem)
- EOS-TC125/135 with EOS-89BTH DSC: 1680 person-mrem (~1.7 person-rem)

The EOS-TC108 results in larger exposures than the EOS-TC125/135 because it is a lighter cask and provides less shielding. Because the EOS-TC108 results in a total computed exposure of ~3.4 person-rem, the exposure due to an EOS-TC108 crane hang-up off-normal event is also considered. Assuming four workers for one hour and location DRL1/Decon, the additional dose due to this event is approximately 0.9 person-rem (872 person-mrem for the EOS-37PTH DSC and 776 person-mrem for the EOS-89BTH DSC).

The EOS-TC125 does not allow an aluminum top cover plate, although the EOS-TC135 may utilize an optional aluminum top cover plate that is exchanged for a steel top cover plate after downending. This option is applicable only to the EOS-TC135 with the EOS-37PTH DSC, since the EOS-89BTH DSC is not an allowed content in the EOS-TC135. The exposure calculations in Table 11-4 are based on a steel top cover plate. If the aluminum top cover plate option is used for the EOS-TC135, the total exposure will increase by approximately 4%, or ~84 person-mrem.

The exposures provided above are bounding estimates. Measured exposures from typical NUHOMS® System loading campaigns have been 600 mrem or lower per canister for normal operations, and exposures for the NUHOMS® EOS System are expected to be similar.

Regulatory Guide 8.34 [11-4] is to be used to define the onsite occupational dose and monitoring requirements.

11.2.2 EOS-DSC Retrieval Operations

Occupational exposures to ISFSI personnel during EOS-DSC retrieval are similar to those exposures calculated for EOS-DSC insertion. Dose rates for retrieval operations will be lower than those for insertion operations due to radioactive decay of the spent fuel inside the EOS-HSM. Therefore, the dose rates for EOS-DSC retrieval are bounded by the dose rates calculated for insertion.

11.2.3 Fuel Unloading Operations

The process of unloading the EOS-DSC is similar to that used for loading the EOS-DSC. The identical ALARA procedures utilized for loading should also be applied to unloading. Occupational exposures to plant personnel are bounded by the exposures calculated for EOS-DSC loading.

11.2.4 Maintenance Operations

The dose rate for surveillance activities is shown in Table 11-8 and Table 11-9 for doses rates 6.1 m from the front of an EOS-HSM. The 6.1-meter dose rate is a conservative estimate for surveillance activities. The EOS-HSM surface dose rates provided in Chapter 6 can be used for temperature sensor maintenance activities, including calibration and repair.

The general licensee will evaluate the additional dose to personnel from ISFSI operations, based on the particular storage configuration and site personnel requirements.

11.2.5 Doses during ISFSI Array Expansion

ISFSI expansion should be planned to eliminate the need for entry into a module adjacent to a loaded module. Similarly, during array expansion, when the shield wall is removed, personnel access to the area should be controlled. Optionally, two empty EOS-HSM base modules can be used in lieu of an end shield wall, though the side inlet vent of the outer module must be blocked. The resulting dose will be less than that calculated in Chapter 6 for the side dose rate of an array with an installed shield wall.

11.3 Offsite Dose Calculations

Calculated dose rates in the immediate vicinity of the NUHOMS® EOS System are presented in Chapter 6, which provides a detailed description of source term configuration, analysis models, and bounding dose rates. Offsite dose rates and annual exposures are presented in this section. Neutron and gamma-ray offsite dose rates are computed, including skyshine, in the vicinity of the two generic ISFSI layouts containing design-basis contents.

11.3.1 Normal Conditions (10 CFR 72.104)

Offsite dose rates from the NUHOMS® EOS System are a result of direct radiation from the ISFSI. The operation of loading an EOS-HSM occurs over a very short time period and contributes negligibly to the offsite dose rates. Therefore, normal condition offsite dose rate calculations are computed only for a loaded ISFSI. No off-normal conditions have been identified that affect offsite dose rates.

Two generic ISFSI designs are considered, a 2x10 back-to-back array and a two 1x10 front-to-front array. In the 2x10 back-to-back array, the rear and corner shield walls are absent because the rears of the EOS-HSMs are in contact. In the two 1x10 front-to-front arrays, the EOS-HSMs are aligned with the rear shield walls facing outward and the front of the EOS-HSMs facing inward, separated by approximately 35 ft. This configuration has the advantage of minimizing the dose rate near the ISFSI because the inlet vents are directed inward in an area that would not normally be occupied.

It is demonstrated in Chapter 6 that EOS-HSM dose rates are larger for the EOS-89BTH DSC compared to the EOS-37PTH DSC. Therefore, offsite dose rates are computed only for the bounding EOS-DSC. This evaluation provides results for distances ranging from 6.1 to 600 m from each face of the two configurations.

The Monte Carlo computer code MCNP5 [11-5] is used to calculate the dose rates at the specified locations around the arrays of EOS-HSMs. The results of this calculation provide an example of how to demonstrate compliance with the relevant radiological requirements of 10 CFR 20, 10 CFR 72, and 40 CFR 190 for a specific site. Each user must perform site-specific calculations to account for the actual layout of the EOS-HSMs and fuel source.

The total annual exposure for each ISFSI layout as a function of distance from each face is given in Table 11-6 and plotted in Figure 11-2. The total annual exposure estimates are based on 100% occupancy for 365 days. At large distances, the annual exposure from the 2x10 back-to-back array is similar to the two 1x10 front-to-front array configuration. Per 10 CFR 72.104, the annual whole-body dose to an individual at the site boundary is limited to 25 mrem. Based on the data shown in Table 11-6, the offsite dose rate drops below 25 mrem at a distance of approximately 370 m from the ISFSI. Therefore, 370 m is the minimum distance with design basis fuel to the site boundary for a 20-cask array with the NUHOMS® EOS System, however a shorter distance can be demonstrated in a site-specific calculation. These calculations are performed without dose reduction hardware in the inlet and outlet vents, which would reduce the off-site dose rates by 30-60%.

The methodology, inputs, and assumptions for the Monte Carlo N-Particle (MCNP) analyses are summarized in the following paragraphs.

- The 20 EOS-HSMs in the 2x10 back-to-back array are modeled as a box enveloping the 2x10 array of EOS-HSMs, including the 3-foot shield walls on the two ends of the array. Source particles are started on the surfaces of the box. A sketch of this geometry is shown in Figure 11-3. The interiors of the EOS-HSMs and shield walls are modeled as air. Most particles that enter the interiors of the EOS-HSMs and shield walls will, therefore, pass through unhindered.
- The 20 EOS-HSMs in the two 1x10 front-to-front arrays are modeled as two boxes that envelop each 1x10 array of EOS-HSMs, including the 3-foot shield walls on the two ends and back of each array. Source particles are started on the surfaces of one of the 1x10 arrays, which is modeled as air. The opposite 1x10 array is modeled as solid concrete. A sketch of this geometry is shown in Figure 11-4. The dose field is then created for a source in both arrays by accounting for model symmetry, as indicated in Figure 11-4.
- The ISFSI approach slab is modeled as concrete. Because the ground composition has, at best, only a secondary impact on the dose rates at the detectors, any differences between this assumed layout and the actual layout would not have a significant effect on the site dose rates.
- The "universe" is a sphere surrounding the ISFSI. To account for skyshine, the radius of this sphere (r=500,000 cm) is more than 10 mean free paths for neutrons and 50 mean free paths for gammas in air, thus ensuring that the model is of a sufficient size to include all interactions, including skyshine, affecting the dose rate at the detectors.
- The 2x10 and two 1x10 surface sources are input to reproduce the average dose rate and spectrum on the surface of the EOS-HSM, as computed in Chapter 6. The surface average fluxes on the front, roof, side, and rear of the EOS-HSM are explicitly computed, and are provided in Tables 6-56 through 6-58. The primary and secondary gamma fluxes are simply summed in the gamma input file. These surface spectra are directly input to MCNP for each face.

- Source particles on the ISFSI array surface are specified with a cosine distribution. For a cosine distribution, the outward particle current is equal to half of the flux. The MCNP source description requires the number of source particles per second emitted on each face (particle current). Because the current is half of the flux for a cosine distribution, and the flux at each face is known, the input current for each face (particles/s) is computed as A*F/2, where A is the area of the face (cm^2) and F is the total flux on each face (particles/cm^2-s). The surface source calculations are summarized in Table 11-7.
- ANSI/ANS 6.1.1-1977 flux-to-dose rate factors are utilized [11-6]. These factors are provided in Table 6-51.
- For the 2x10 back-to-back array with end shield walls, the "box" dimensions are 1361 cm wide, 3129 cm long, and 564 cm high. For the two 1x10 front-to-front arrays with end and back shield walls, the "box" dimensions for each array are 772 cm wide, 3129 cm long, and 564 cm high. The two 1x10 arrays are 1067 cm (35 ft) apart.
- Dose rates are calculated for distances of 6.1 m (20 ft) to 600 m from the edges of the two ISFSI configurations. Point detectors are placed at the following locations, as measured from each face of the "box": 6.095 m (20 ft), 10 m, 20 m, 30 m, 40 m, 50 m, 60 m, 70 m, 80 m, 90 m, 100 m, 200 m, 300 m, 400 m, 500 m, and 600 m. Each point detector is placed 91 cm (~3 ft) above the ground.

The MCNP results for the 2x10 back-to-back and two 1x10 front-to-front configurations are summarized in Table 11-8 and Table 11-9, respectively. At near distances, the 2x10 configuration results in larger front dose rates than the outward rear of the two 1x10 configuration. For example, the 6.1 m front dose rate is 9.51 mrem/hr for the 2x10 array compared to 3.66 mrem/hr for the two 1x10 array. However, at near distances, the two 1x10 configuration results in nominally larger side dose rates than the 2x10 array.

At large distances, the dose rates are approximately the same, regardless of configuration or direction from the ISFSI, as the dose rate at large distances is dominated by skyshine from the radiation streaming from the roof outlet vents. Also, note that the neutron dose rate is negligible compared to the gamma dose rate at all dose rate locations.

The Monte Carlo convergence is excellent due to the low uncertainty (<5%) for most dose rate locations. The Monte Carlo uncertainty is large (~15%) only for the detector 600 m from the front of the 2x10 array. However, the dose rate at this location is vanishingly small, and the result is acceptable. The annual exposures reported in Table 11-6 are simply the computed dose rates multiplied by 8760 hours (1 year).

The preceding analyses and results are intended to provide high estimates of dose rates for generic ISFSI layouts. The written evaluations performed by a general licensee for the actual ISFSI must consider the type and number of storage units, layout, characteristics of the irradiated fuel to be stored, site characteristics (e.g., berms, distance to the controlled area boundary, etc.), and reactor operations at the site in order to demonstrate compliance with 10 CFR 72.104.

11.3.2 Accident Conditions (10 CFR 72.106)

Per 10 CFR 72.106, the exposure to an individual at the site boundary due to an accident is limited to 5 rem. In an accident, the EOS-HSM outlet vent covers and wind deflectors (if required) may be lost. Only the dose rates on the roof are affected, since the front, rear, and side dose rates remain the same. As the dose rates at large distances are mostly due to skyshine from the roof, the dose rates at the site boundary are directly affected. The average EOS-HSM roof dose rates and surface fluxes in an accident are computed in Chapter 6, Tables 6-56 through 6-58. Table 11-10 shows the bounding dose rate as a function of distance from a 2x10 back-to-back array of EOS-HSMs for the accident configuration described above. These dose rates are calculated assuming that the outlet vent covers and wind deflectors (if required) for the entire array are lost.

MCNP inputs for a 2x10 ISFSI accident configuration are prepared using the same method as described for the normal condition models. At a distance of 200 m and 370 m from the ISFSI, the accident dose rate is approximately 1.1 mrem/hr and 0.1 mrem/hr, respectively. It is assumed that the recovery time for this accident is five days (120 hours). Therefore, the total exposure to an individual at a distance of 200 m and 370 m is 132 mrem and 12 mrem respectively. This is significantly less than the 10 CFR 72.106 limit of 5 rem.

The EOS-TC may also be damaged in an accident during transfer operations, which would result in an offsite dose. For accident conditions, it is assumed that the neutron shield, including the steel or aluminum shell, is absent. The EOS-TC accident calculations are documented in Section 6.4.3 and the results presented in Table 6-54. The maximum dose rate is 2.15 mrem/hr at a distance of 100 m from the EOS-TC. If an 8-hour recovery time is assumed, the dose to an individual at the site boundary is 17 mrem, which is significantly below the 10 CFR 72.106 limit of 5 rem. This dose is also conservatively large because it is calculated as a distance of 100 m from the EOS-TC.

11.4 Ensuring that Occupational Radiation Exposures Are ALARA

11.4.1 Policy Considerations

The licensee's radiation safety and ALARA policies should be applied to the ISFSI. The ALARA program should follow the general guidelines of Regulatory Guides 1.8 [11-7], 8.8 [11-8], 8.10 [11-9], and 10 CFR 20. ISFSI personnel should be trained in the proper operation, inspection, repair and maintenance of the NUHOMS® EOS System, and updated on ALARA practices and dose reduction techniques. Implementation of ISFSI procedures should be reviewed by the licensee to ensure ALARA exposure.

11.4.2 Design Considerations

The thick inner cover of the EOS-DSC is designed to minimize exposure during draining, drying, and closure operations. The vent and drain ports are designed for maximum water flow rate and vacuum conductance to minimize the time (and thereby the exposure) associated with draining and vacuum drying. The design of the cover welds minimizes exposure during closure operations. The welds are designed to be easily performed by remote welding equipment. Because the cover welds are not used to lift the canister, they are relatively small, reducing the time needed to complete them. Because they are austenitic welds, no pre-heating is required. These welds are tested to be leak-tight as described in Chapter 8. Therefore, exposure associated with a leaking EOS-DSC is eliminated.

Lead, steel, water, and borated polyethylene in the EOS-TC provide required gamma and neutron shielding during transfer activities. The exterior of the EOS-TC is decontaminated prior to transfer to the ISFSI, thereby minimizing exposure of personnel to surface contamination.

The NUHOMS® EOS-HSM storage modules include no active components that require periodic maintenance, thereby minimizing potential personnel dose due to maintenance activities.

The shielding design features of the storage modules storage minimize occupational exposure for any activities on or near the ISFSI. These features are:

- The EOS-DSCs are loaded and sealed prior to transfer to the ISFSI. Seals are stainless steel welds with at least two layers.
- The fuel will not be unloaded, nor will the EOS-DSCs be opened at the ISFSI unless the ISFSI is specifically licensed for these purposes.
- The fuel is stored in a dry, inert environment inside the EOS-DSCs, so that no radioactive liquid is available for leakage.
- The EOS-DSCs are sealed with a helium atmosphere to prevent oxidation of the fuel. The leaktight design features are described in Chapter 8.

- The EOS-DSCs are heavily shielded on both ends to reduce external dose rates. The shielding design features are discussed in Chapter 6.
- No radioactive material will be discharged during storage since the EOS-DSC is designed, fabricated, and tested to be leaktight.
- The EOS-DSC outside surface is contamination free due to the use of clean water sealed in the annulus between the EOS-TC and EOS-DSC during loading operations.
- EOS-HSMs provide thick concrete shielding, while placement of modules immediately adjacent to one another enhances the effectiveness of this shielding.

Regulatory Position 2 of Regulatory Guide 8.8, is incorporated into the design considerations, as described below:

- Regulatory Position 2a, on access control, is met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access.
- Regulatory Position 2b, on radiation shielding, is met by the heavy shielding of the NUHOMS® EOS System, which minimizes personnel exposures.
- Regulatory Position 2c, on process instrumentation and controls, is met by designing the instrumentation for a long service life and locating readouts in a low dose rate location. The use of temperature sensors for temperature measurements located in embedded thermowells provides reliable, easily maintainable instrumentation for this monitoring function.
- Regulatory Position 2d, on control of airborne contaminants, does not apply during transfer or storage because neither gaseous releases nor significant surface contamination are expected.
- Regulatory Position 2e, on crud control, is not applicable to the ISFSI because there are no systems at the ISFSI that could transport crud. The leaktight EOS-DSC design ensures that used fuel crud will not be released or transferred from the EOS-DSC. Draining back to the used fuel pool provides control over any crud that could be entrained in the outflow from the EOS-DSC draining operations.
- Regulatory Position 2f, on decontamination, is met because the EOS-TC is decontaminated prior to transfer to the ISFSI. The EOS-TC accessible surfaces are designed to facilitate decontamination.
- Regulatory Position 2g, on radiation monitoring, does not apply. There is no need for airborne radioactivity monitoring because the EOS-DSCs are sealed by leaktight welds. Area radiation monitors are not required because the ISFSI will not be occupied on a regular basis.
- Regulatory Position 2h, on resin and sludge treatment systems, is not applicable to the ISFSI because there are no radioactive systems containing resins or sludge associated with the ISFSI.

- Regulatory Position 2i concerning other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the ISFSI.

11.4.3 Operational Considerations

The operations description in Chapter 9 makes provision for measures that can minimize doses during operations, including:

- using temporary shielding,
- wetting equipment with clean water prior to pool immersion to improve ease of decontamination,
- preventing contamination of the EOS-DSC exterior by the use of clean water in a sealed EOS-TC/DSC annulus,
- using items such as remote equipment for welding and long-handled tools for decontamination, and
- controlling gases and liquids removed from the EOS-DSC during EOS-DSC vacuum drying and during fuel unloading.

The areas of highest operational dose are the front of a loaded EOS-HSM at the air inlet vent, at the EOS-TC side or EOS-DSC top with a partially or completely drained EOS-DSC (cover welding, transfer operations), and at the EOS-TC/DSC annulus. Operating procedures, temporary shielding, and personnel training are put into practice to minimize personnel exposure in these areas.

The EOS-DSCs contain no radioactive liquids and radioactive gases will be contained inside the fuel rods. The EOS-DSC is designed and welded to be leaktight.

The EOS-HSM and EOS-DSCs are designed to be essentially maintenance free. It is a passive system with no moving parts. The only anticipated maintenance procedures are the visual inspection of the bird screens on the EOS-HSM ventilation inlet and outlet openings, and periodic maintenance of the temperature sensors. Maintenance operations on the EOS-TC, transfer equipment and other auxiliary equipment are normally performed, in a low dose environment, during periods when fuel movement is not occurring.

The ISFSI contains no systems that process liquids or gases, or contain, collect, store, or transport radioactive liquids or solids, other than payloads identified in Chapter 2. Therefore, the ISFSI meets ALARA requirements, since there are no systems to be maintained other than the transfer and auxiliary equipment.

11.5 References

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11-2 Title 10, Code of Federal Regulations Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related greater than Class C Waste."

11-3 Title 40, Code of Federal Regulations, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

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Table 11-1
Occupational Dose Rates

			Dose Rate (mrem/hr)			
			EOS-TC108		EOS-TC125/135	
Dose Rate Location	Averaged Segments	Config.	EOS-37PTH DSC	EOS-89BTH DSC	EOS-37PTH DSC	EOS-89BTH DSC
DRL1	A1-18, R11	Decon.	218	194	82	62
DRL2	A3-16, R10	Decon.	-	-	258	181
		Transfer	726	747	209	239
DRL3	A17, R9	Decon.	-	-	223	98
		Welding	206	198	120	113
		Transfer	208	199	-	-
DRL4	A3-11, R9	Decon.	1191	1050	-	-
DRL5	A1-18, R10	Transfer	559	586	167	191
DRL6	A17-18, R9	Transfer	118	100	54	43
DRL7	A17-18, R10	Transfer	199	189	89	91
DRL8	A2, R9	Transfer	91	165	54	65
DRL9	A19, R0	Transfer	75	137	73	114
DRL10	A1, R10	Transfer	82	121	34	38
EOS-HSM	Front face surface average	-	22	22	22	22

Table 11-2
Occupational Exposure, EOS-TC108 with EOS-37PTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
1	Place an empty EOS-DSC into an EOS-TC and prepare the EOS-TC for placement into the spent fuel pool.	N/A	N/A	6	4	0	0	0%
2	Move the EOS-TC containing an EOS-DSC without fuel into the spent fuel pool.	N/A	N/A	6	1.5	0	0	0%
3	Remove a loaded EOS-TC from the fuel pool and place in the decontamination area.	Decon.	DRL1	2	0.25	218	109	3%
4	Install neutron shield. Fill neutron shield with water.	Decon.	DRL4	3	0.33	1191	1179	35%
5	Prepare and weld inner top cover plate.	Welding	DRL3	2	0.75	206	309	9%
6	Vacuum dry and backfill with helium.	Welding	DRL3	2	0.5	206	206	6%
7	Weld outer top cover plate and port covers, perform non-destructive examination.	Welding	DRL3	2	0.5	206	206	6%
8	Drain annulus. Install EOS-TC aluminum top cover. Ready the support skid and transfer trailer.	Transfer	DRL5	1	0.5	559	280	8%
9	Place the EOS-TC onto the skid and trailer. Secure the EOS-TC to the skid.	Transfer	DRL2	2	0.33	726	479	14%
10	Remove aluminum top cover and replace with steel top cover.	Transfer	DRL3	2	0.33	208	137	4%
11	Ready the skid and trailer. Transfer the EOS-TC to ISFSI. Position the EOS-TC in close proximity with the EOS-HSM.	N/A	N/A	6	1.83	0	0	0%
12	Remove the EOS-TC top cover.	Transfer	HSM+DRL6	2	0.67	140	188	6%

Table 11-2
Occupational Exposure, EOS-TC108 with EOS-37PTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
13	Align and dock the EOS-TC with the EOS-HSM.	Transfer	HSM+DRL7	2	0.25	221	111	3%
14	Position and align ram with EOS-TC.	Transfer	HSM+DRL8	2	0.5	113	113	3%
15	Remove ram access cover plate.	Transfer	DRL9	1	0.08	75	6	0%
16	Transfer the EOS-DSC from the EOS-TC to the EOS-HSM.	N/A	N/A	3	0.5	0	0	0%
17	Lift the ram back onto the trailer and un-dock the EOS-TC from the EOS-HSM.	Transfer	HSM+DRL10	2	0.08	104	17	0%
18	Install EOS-HSM access door.	Transfer	HSM	2	0.5	22	22	1%
						Total	3361(1)	

Note:

(1) A crane hang-up off-normal event adds 872 person-mrem (DRL1/decon * 4 workers * 1 hour).

Table 11-3
Occupational Exposure, EOS-TC108 with EOS-89BTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
1	Place an empty EOS-DSC into an EOS-TC and prepare the EOS-TC for placement into the spent fuel pool.	N/A	N/A	6	4.00	0	0	0%
2	Move the EOS-TC containing an EOS-DSC without fuel into the spent fuel pool.	N/A	N/A	6	1.50	0	0	0%
3	Remove a loaded EOS-TC from the fuel pool and place in the decontamination area.	Decon.	DRL1	2	0.25	194	97	3%
4	Install neutron shield. Fill neutron shield with water.	Decon.	DRL4	3	0.33	1050	1040	32%
5	Prepare and weld inner top cover plate.	Welding	DRL3	2	0.75	198	297	9%
6	Vacuum dry and backfill with helium.	Welding	DRL3	2	0.50	198	198	6%
7	Weld outer top cover plate and port covers, perform non-destructive examination.	Welding	DRL3	2	0.50	198	198	6%
8	Drain annulus. Install EOS-TC aluminum top cover. Ready the support skid and transfer trailer.	Transfer	DRL5	1	0.50	586	293	9%
9	Place the EOS-TC onto the skid and trailer. Secure the EOS-TC to the skid.	Transfer	DRL2	2	0.33	747	498	15%
10	Remove aluminum top cover and replace with steel top cover.	Transfer	DRL3	2	0.33	199	133	4%
11	Ready the skid and trailer. Transfer the EOS-TC to ISFSI. Position the EOS-TC in close proximity with the EOS-HSM.	N/A	N/A	6	1.83	0	0	0%
12	Remove the EOS-TC top cover.	Transfer	HSM+DRL6	2	0.67	122	163	5%

Table 11-3
Occupational Exposure, EOS-TC108 with EOS-89BTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
13	Align and dock the EOS-TC with the EOS-HSM.	Transfer	HSM+DRL7	2	0.25	211	106	3%
14	Position and align ram with EOS-TC.	Transfer	HSM+DRL8	2	0.50	187	187	6%
15	Remove ram access cover plate.	Transfer	DRL9	1	0.08	137	11	0%
16	Transfer the EOS-DSC from the EOS-TC to the EOS-HSM.	N/A	N/A	3	0.50	0	0	0%
17	Lift the ram back onto the trailer and un-dock the EOS-TC from the EOS-HSM.	Transfer	HSM+DRL10	2	0.08	143	24	1%
18	Install EOS-HSM access door.	Transfer	HSM	2	0.50	22	22	1%
						Total	3270(1)	

Note:

(1) A crane hang-up off-normal event adds 776 person-mrem (DRL1/decon * 4 workers * 1 hour).

Table 11-4
Occupational Exposure, EOS-TC125/135 with EOS-37PTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
1	Drain neutron shield if necessary. Place an empty EOS-DSC into an EOS-TC and prepare the EOS-TC for placement into the spent fuel pool.	N/A	N/A	6	4	0	0	0%
2	Move the EOS-TC containing an EOS-DSC without fuel into the spent fuel pool.	N/A	N/A	6	1.5	0	0	0%
3	Remove a loaded EOS-TC from the fuel pool and place in the decontamination area. Refill neutron shield tank if necessary.	Decon.	DRL1	2	0.25	82	41	2%
4	Decontaminate the EOS-TC and prepare welds.	Decon.	DRL2	2	1.75	258	904	43%
		Decon.	DRL3	2	0.5	223	223	11%
5	Weld inner top cover plate.	Welding	DRL3	2	0.75	120	181	9%
6	Vacuum dry and backfill with helium.	Welding	DRL3	2	0.5	120	120	6%
7	Weld outer top cover plate and port covers, perform non-destructive examination.	Welding	DRL3	2	0.5	120	120	6%
8	Drain annulus. Install EOS-TC top cover. Ready the support skid and transfer trailer.	Transfer	DRL5	1	0.5	167	83	4%
9	Place the EOS-TC onto the skid and trailer. Secure the EOS-TC to the skid.	Transfer	DRL2	2	0.33	209	138	7%
10	Ready the skid and trailer for service. Transfer the EOS-TC to ISFSI. Position the EOS-TC in close proximity with the EOS-HSM.	N/A	N/A	6	1.83	0	0	0%
11	Remove the EOS-TC top cover.	Transfer	HSM+DRL6	2	0.67	76	102	5%

Table 11-4
Occupational Exposure, EOS-TC125/135 with EOS-37PTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
12	Align and dock the EOS-TC with the EOS-HSM.	Transfer	HSM+DRL7	2	0.25	111	56	3%
13	Position and align ram with EOS-TC.	Transfer	HSM+DRL8	2	0.5	76	76	4%
14	Remove ram access cover plate.	Transfer	DRL9	1	0.08	73	6	0%
15	Transfer the EOS-DSC from the EOS-TC to the EOS-HSM.	N/A	N/A	3	0.5	0	0	0%
16	Lift the ram back onto the trailer and un-dock the EOS-TC from the EOS-HSM.	Transfer	HSM+DRL10	2	0.08	56	9	0%
17	Install EOS-HSM access door.	Transfer	HSM	2	0.5	22	22	1%
						Total	2081(1)	

Note:

(1) Use of aluminum cask lid increases total occupational dose by approximately 4 % (~84 person-mrem).

Table 11-5
Occupational Exposure, EOS-TC125/135 with EOS-89BTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
1	Drain neutron shield if necessary. Place an empty EOS-DSC into an EOS-TC and prepare the EOS-TC for placement into the spent fuel pool.	N/A	N/A	6	4.00	0	0	0%
2	Move the EOS-TC containing an EOS-DSC without fuel into the spent fuel pool.	N/A	N/A	6	1.50	0	0	0%
3	Remove a loaded EOS-TC from the fuel pool and place in the decontamination area. Refill neutron shield tank if necessary.	Decon.	DRL1	2	0.25	62	31	2%
4	Decontaminate the EOS-TC and prepare welds.	Decon.	DRL2	2	1.75	181	634	38%
		Decon.	DRL3	2	0.50	98	98	6%
5	Weld inner top cover plate.	Welding	DRL3	2	0.75	113	170	10%
6	Vacuum dry and backfill with helium.	Welding	DRL3	2	0.50	113	113	7%
7	Weld outer top cover plate and port covers, perform non-destructive examination.	Welding	DRL3	2	0.50	113	113	7%
8	Drain annulus. Install EOS-TC top cover. Ready the support skid and transfer trailer.	Transfer	DRL5	1	0.50	191	96	6%
9	Place the EOS-TC onto the skid and trailer. Secure the EOS-TC to the skid.	Transfer	DRL2	2	0.33	239	159	10%
10	Ready the skid and trailer for service. Transfer the EOS-TC to ISFSI. Position the EOS-TC in close proximity with the EOS-HSM.	N/A	N/A	6	1.83	0	0	0%
11	Remove the EOS-TC top cover.	Transfer	HSM+DRL6	2	0.67	65	87	5%

Table 11-5
Occupational Exposure, EOS-TC125/135 with EOS-89BTH DSC
(2 Pages)

No.	Operation	Configuration	Dose Rate Location	No. of People	Duration (hr)	Dose Rate (mrem/hr)	Dose (person-mrem)	% of Total Dose
12	Align and dock the EOS-TC with the EOS-HSM.	Transfer	HSM+DRL7	2	0.25	113	57	3%
13	Position and align ram with EOS-TC.	Transfer	HSM+DRL8	2	0.50	87	87	5%
14	Remove ram access cover plate.	Transfer	DRL9	1	0.08	114	10	1%
15	Transfer the EOS-DSC from the EOS-TC to the EOS-HSM.	N/A	N/A	3	0.50	0	0	0%
16	Lift the ram back onto the trailer and un-dock the EOS-TC from the EOS-HSM.	Transfer	HSM+DRL10	2	0.08	60	10	1%
17	Install EOS-HSM access door.	Transfer	HSM	2	0.50	22	22	1%
						Total	1680	

Table 11-6
Total Annual Exposure from ISFSI

Distance (m)	2x10		Two 1x10	
	Front Total Dose (mrem)	Side Total Dose (mrem)	Back Total Dose (mrem)	Side Total Dose (mrem)
6.1	8.33E+04	2.83E+04	3.20E+04	4.22E+04
10	5.51E+04	2.14E+04	2.51E+04	2.87E+04
20	2.62E+04	1.30E+04	1.56E+04	1.56E+04
30	1.55E+04	8.93E+03	1.08E+04	1.04E+04
40	1.03E+04	6.45E+03	7.80E+03	7.41E+03
50	7.26E+03	4.87E+03	5.85E+03	5.54E+03
60	5.40E+03	3.74E+03	4.47E+03	4.22E+03
70	4.07E+03	2.92E+03	3.57E+03	3.31E+03
80	3.14E+03	2.31E+03	2.79E+03	2.63E+03
90	2.49E+03	1.85E+03	2.21E+03	2.10E+03
100	1.99E+03	1.51E+03	1.81E+03	1.72E+03
200	3.03E+02	2.43E+02	2.93E+02	2.80E+02
300	6.41E+01	5.02E+01	6.02E+01	5.94E+01
400	1.43E+01	1.30E+01	1.53E+01	1.43E+01
500	4.13E+00	3.42E+00	4.42E+00	4.08E+00
600	1.37E+00	1.04E+00	1.21E+00	1.18E+00

Table 11-7
ISFSI Surface Sources

2x10 Back-to-Back Array			
Source	**Area (cm^2)**	**Neutron Source (n/s)**	**Gamma Source (γ/s)**
Roof	4.260E+06	4.017E+08	8.802E+11
Front 1	1.765E+06	3.054E+07	5.073E+10
Front 2	1.765E+06	3.054E+07	5.073E+10
Side 1	7.677E+05	1.547E+06	1.621E+09
Side 2	7.677E+05	1.547E+06	1.621E+09
Total	9.325E+06	4.658E+08	9.849E+11
Two 1x10 Front-to-Front Arrays			
Source	**Area (cm^2)**	**Neutron Source (n/s)**	**Gamma Source (γ/s)**
Roof	2.416E+06	2.278E+08	4.992E+11
Front	1.765E+06	3.054E+07	5.073E+10
Back	1.765E+06	2.262E+06	6.164E+09
Side 1	4.354E+05	8.773E+05	9.193E+08
Side 2	4.354E+05	8.773E+05	9.193E+08
Total	6.816E+06	2.624E+08	5.580E+11

Table 11-8
2x10 Back-to-Back Dose Rates
(2 Pages)

In Front of ISFSI				
Distance (m)	**Gamma Dose Rate (mrem/hr)**	**Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**σ**
6.1	9.35E+00	1.55E-01	9.51E+00	0.1%
10	6.18E+00	1.10E-01	6.29E+00	0.1%
20	2.93E+00	5.53E-02	2.99E+00	0.1%
30	1.74E+00	3.26E-02	1.77E+00	0.2%
40	1.15E+00	2.14E-02	1.17E+00	0.2%
50	8.14E-01	1.48E-02	8.29E-01	0.3%
60	6.06E-01	1.06E-02	6.17E-01	0.4%
70	4.57E-01	7.97E-03	4.65E-01	0.3%
80	3.53E-01	6.08E-03	3.59E-01	0.4%
90	2.80E-01	4.79E-03	2.84E-01	0.5%
100	2.23E-01	3.80E-03	2.27E-01	0.6%
200	3.40E-02	6.32E-04	3.46E-02	1.3%
300	7.15E-03	1.73E-04	7.32E-03	4.5%
400	1.57E-03	5.89E-05	1.63E-03	1.7%
500	4.54E-04	1.79E-05	4.72E-04	3.3%
600	1.49E-04	7.89E-06	1.57E-04	14.9%

Table 11-8
2x10 Back-to-Back Dose Rates
(2 Pages)

At Side of ISFSI				
Distance (m)	**Gamma Dose Rate (mrem/hr)**	**Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**σ**
6.1	3.16E+00	7.72E-02	3.23E+00	0.1%
10	2.38E+00	5.98E-02	2.44E+00	0.2%
20	1.45E+00	3.45E-02	1.49E+00	0.2%
30	9.97E-01	2.23E-02	1.02E+00	0.2%
40	7.21E-01	1.52E-02	7.36E-01	0.3%
50	5.45E-01	1.10E-02	5.56E-01	0.3%
60	4.19E-01	8.04E-03	4.27E-01	1.0%
70	3.28E-01	6.16E-03	3.34E-01	0.4%
80	2.59E-01	4.84E-03	2.64E-01	0.4%
90	2.08E-01	3.81E-03	2.11E-01	0.4%
100	1.70E-01	2.99E-03	1.73E-01	0.6%
200	2.71E-02	6.14E-04	2.77E-02	1.1%
300	5.59E-03	1.35E-04	5.73E-03	1.6%
400	1.42E-03	6.43E-05	1.48E-03	3.5%
500	3.70E-04	2.00E-05	3.90E-04	3.8%
600	1.11E-04	7.94E-06	1.19E-04	4.1%

Table 11-9
Two 1x10 Front-to-Front Dose Rates
(2 Pages)

In Back of ISFSI				
Distance (m)	**Gamma Dose Rate (mrem/hr)**	**Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**σ**
6.1	3.58E+00	7.86E-02	3.66E+00	0.1%
10	2.80E+00	6.46E-02	2.86E+00	0.1%
20	1.75E+00	3.99E-02	1.79E+00	0.2%
30	1.20E+00	2.60E-02	1.23E+00	0.3%
40	8.73E-01	1.79E-02	8.91E-01	0.2%
50	6.55E-01	1.29E-02	6.68E-01	0.4%
60	5.00E-01	9.51E-03	5.10E-01	0.3%
70	4.01E-01	7.23E-03	4.08E-01	1.5%
80	3.12E-01	5.63E-03	3.18E-01	0.4%
90	2.48E-01	4.37E-03	2.52E-01	0.4%
100	2.04E-01	3.47E-03	2.07E-01	0.5%
200	3.29E-02	6.08E-04	3.35E-02	1.0%
300	6.72E-03	1.57E-04	6.87E-03	1.4%
400	1.69E-03	5.74E-05	1.74E-03	3.0%
500	4.78E-04	2.63E-05	5.04E-04	6.4%
600	1.30E-04	8.19E-06	1.38E-04	4.1%

Table 11-9
Two 1x10 Front-to-Front Dose Rates
(2 Pages)

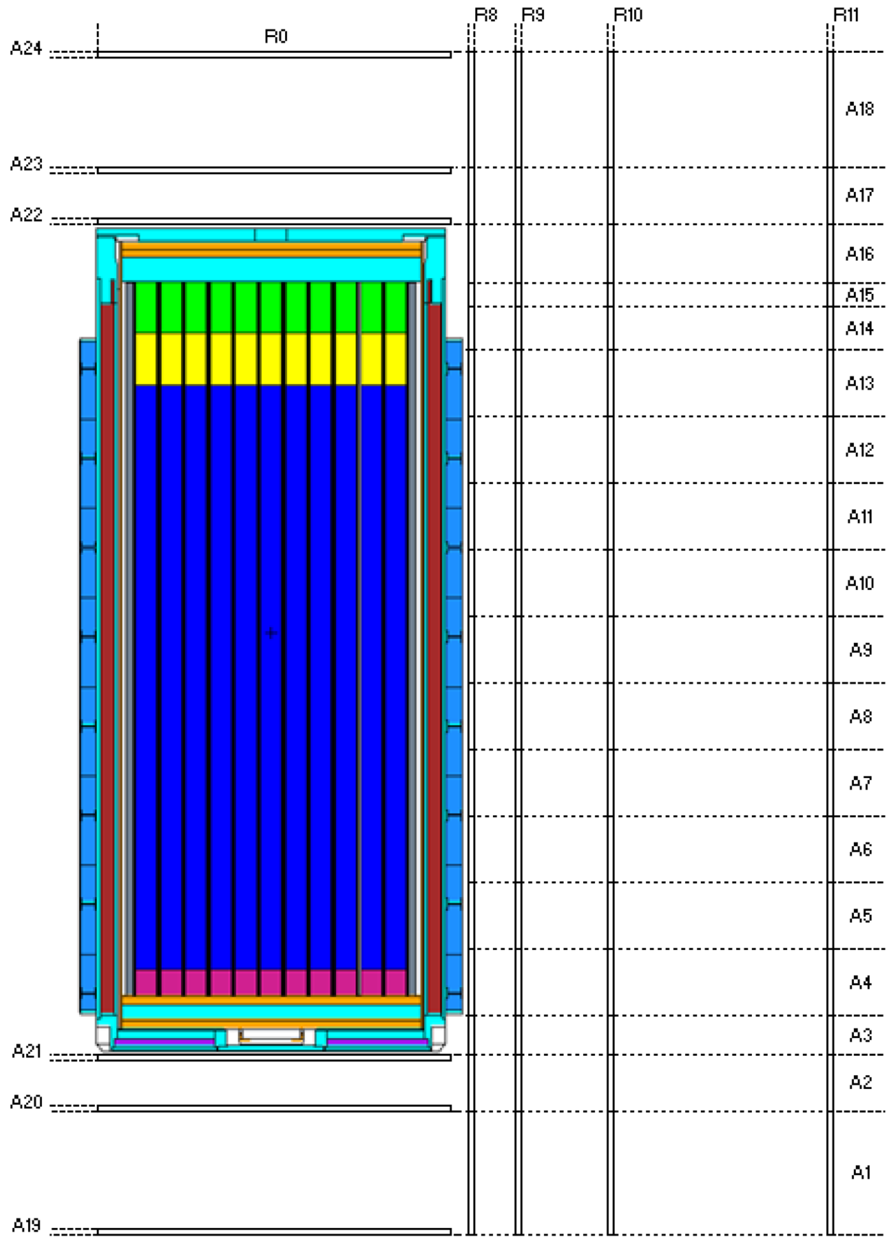
At Side of ISFSI				
Distance (m)	**Gamma Dose Rate (mrem/hr)**	**Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**σ**
6.1	4.73E+00	9.14E-02	4.82E+00	0.2%
10	3.21E+00	6.85E-02	3.27E+00	0.2%
20	1.74E+00	3.82E-02	1.78E+00	0.3%
30	1.16E+00	2.43E-02	1.19E+00	0.3%
40	8.29E-01	1.66E-02	8.46E-01	0.4%
50	6.20E-01	1.23E-02	6.32E-01	0.5%
60	4.72E-01	8.94E-03	4.81E-01	0.4%
70	3.71E-01	6.93E-03	3.78E-01	0.5%
80	2.95E-01	5.13E-03	3.01E-01	0.8%
90	2.35E-01	4.01E-03	2.39E-01	0.7%
100	1.92E-01	3.49E-03	1.96E-01	0.8%
200	3.13E-02	6.73E-04	3.20E-02	1.5%
300	6.61E-03	1.66E-04	6.78E-03	2.9%
400	1.58E-03	4.63E-05	1.63E-03	2.8%
500	4.47E-04	1.93E-05	4.66E-04	7.0%
600	1.28E-04	7.39E-06	1.35E-04	6.7%

Table 11-10
2x10 Back-to-Back Accident Dose Rates
(2 Pages)

Accident Front Dose Rate				
Distance (m)	**Gamma Dose Rate (mrem/hr)**	**Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**σ**
6.1	1.05E+02	4.59E-01	1.05E+02	0.2%
10	8.37E+01	3.72E-01	8.41E+01	0.1%
20	5.29E+01	2.22E-01	5.31E+01	0.2%
30	3.65E+01	1.48E-01	3.66E+01	0.2%
40	2.64E+01	1.05E-01	2.65E+01	0.3%
50	1.98E+01	7.81E-02	1.99E+01	0.4%
60	1.52E+01	5.96E-02	1.53E+01	0.3%
70	1.20E+01	4.61E-02	1.21E+01	0.4%
80	9.58E+00	3.82E-02	9.62E+00	0.8%
90	7.65E+00	3.01E-02	7.68E+00	0.4%
100	6.26E+00	2.44E-02	6.29E+00	0.6%
200	1.04E+00	4.95E-03	1.05E+00	0.8%
300	2.25E-01	1.34E-03	2.27E-01	1.5%
400	6.33E-02	5.91E-04	6.39E-02	6.4%
500	1.71E-02	1.82E-04	1.72E-02	3.3%
600	4.88E-03	9.30E-05	4.97E-03	3.6%

Table 11-10
2x10 Back-to-Back Accident Dose Rates
(2 Pages)

Accident Side Dose Rate				
Distance (m)	**Gamma Dose Rate (mrem/hr)**	**Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**σ**
6.1	7.91E+01	3.11E-01	7.94E+01	0.2%
10	6.37E+01	2.64E-01	6.39E+01	0.2%
20	4.12E+01	1.67E-01	4.14E+01	0.2%
30	2.89E+01	1.15E-01	2.91E+01	0.2%
40	2.15E+01	8.25E-02	2.16E+01	0.4%
50	1.62E+01	6.30E-02	1.63E+01	0.3%
60	1.26E+01	4.88E-02	1.27E+01	0.4%
70	1.00E+01	3.88E-02	1.00E+01	0.4%
80	7.98E+00	3.12E-02	8.01E+00	0.4%
90	6.52E+00	2.57E-02	6.54E+00	0.5%
100	5.35E+00	2.20E-02	5.37E+00	0.7%
200	9.27E-01	4.32E-03	9.31E-01	1.0%
300	2.12E-01	1.30E-03	2.13E-01	2.5%
400	5.40E-02	4.13E-04	5.44E-02	2.5%
500	1.43E-02	1.82E-04	1.45E-02	3.3%
600	4.60E-03	7.45E-05	4.68E-03	4.9%



EOS-89BTH DSC in EOS-TC125/135 depicted. Other configurations are similar.

Figure 11-1
EOS-TC General Dose Rate Tally Locations

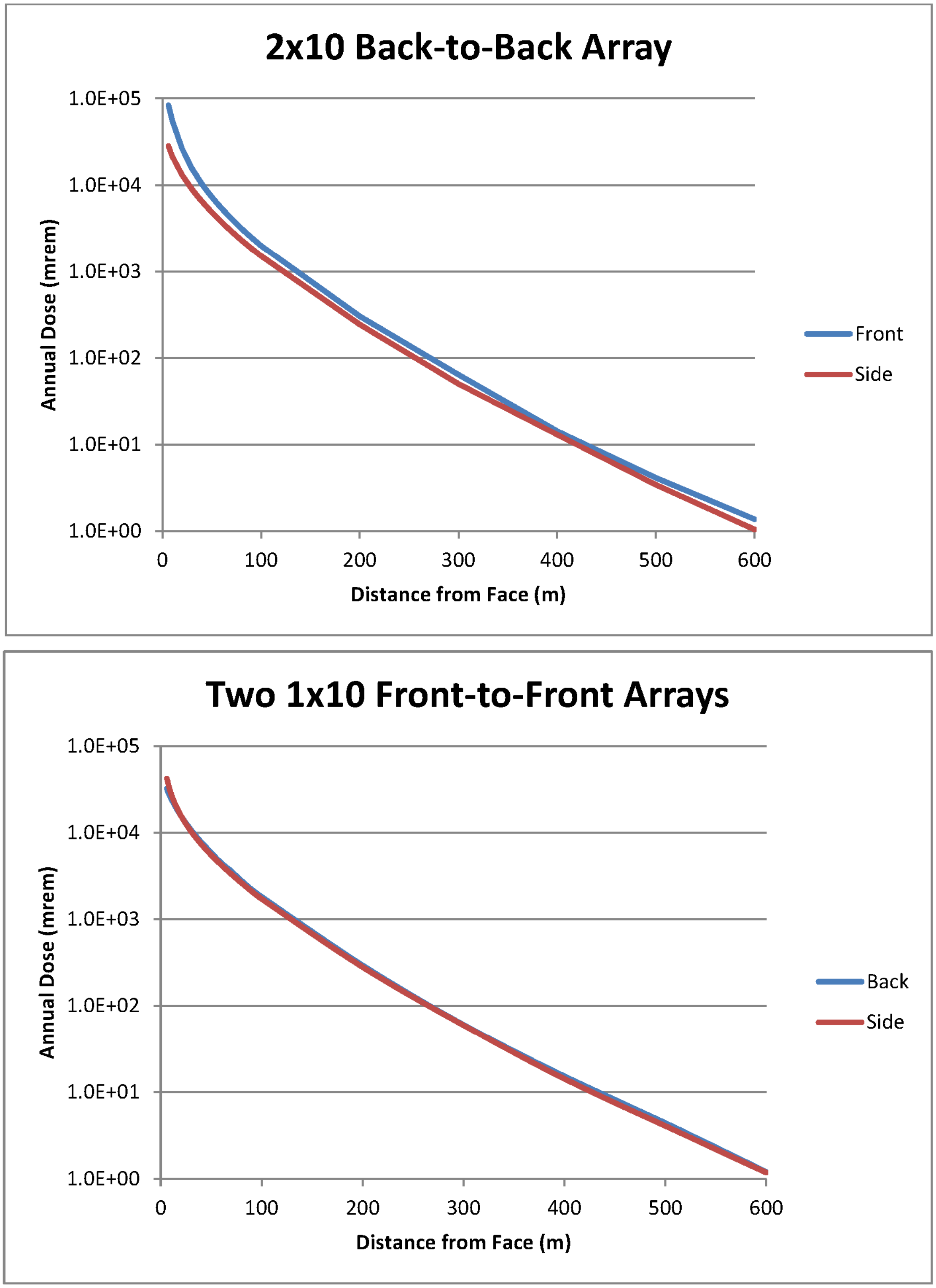


Figure 11-2
Total Annual Exposure from the ISFSI

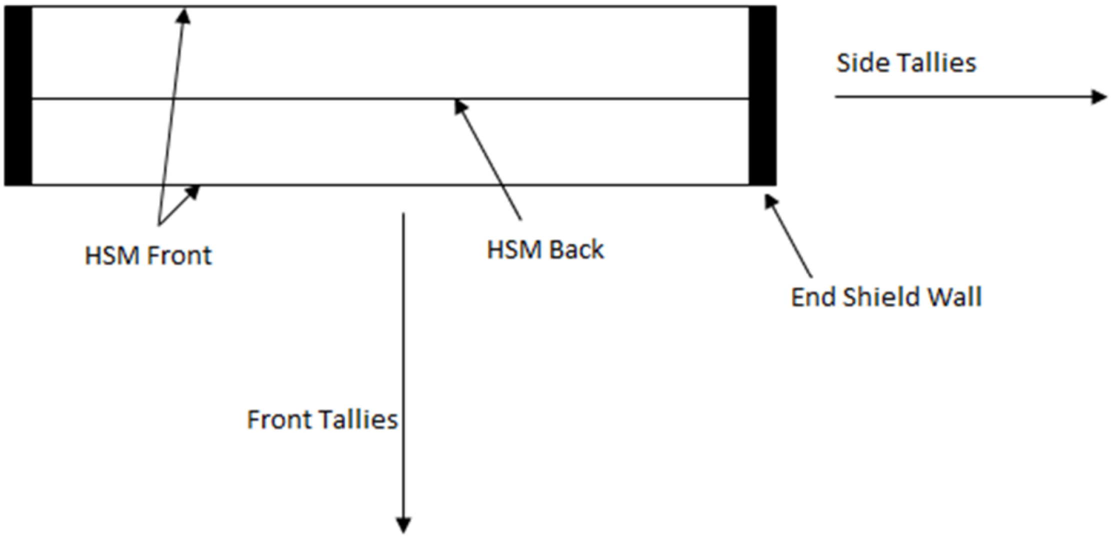


Figure 11-3
2x10 ISFSI MCNP Geometry

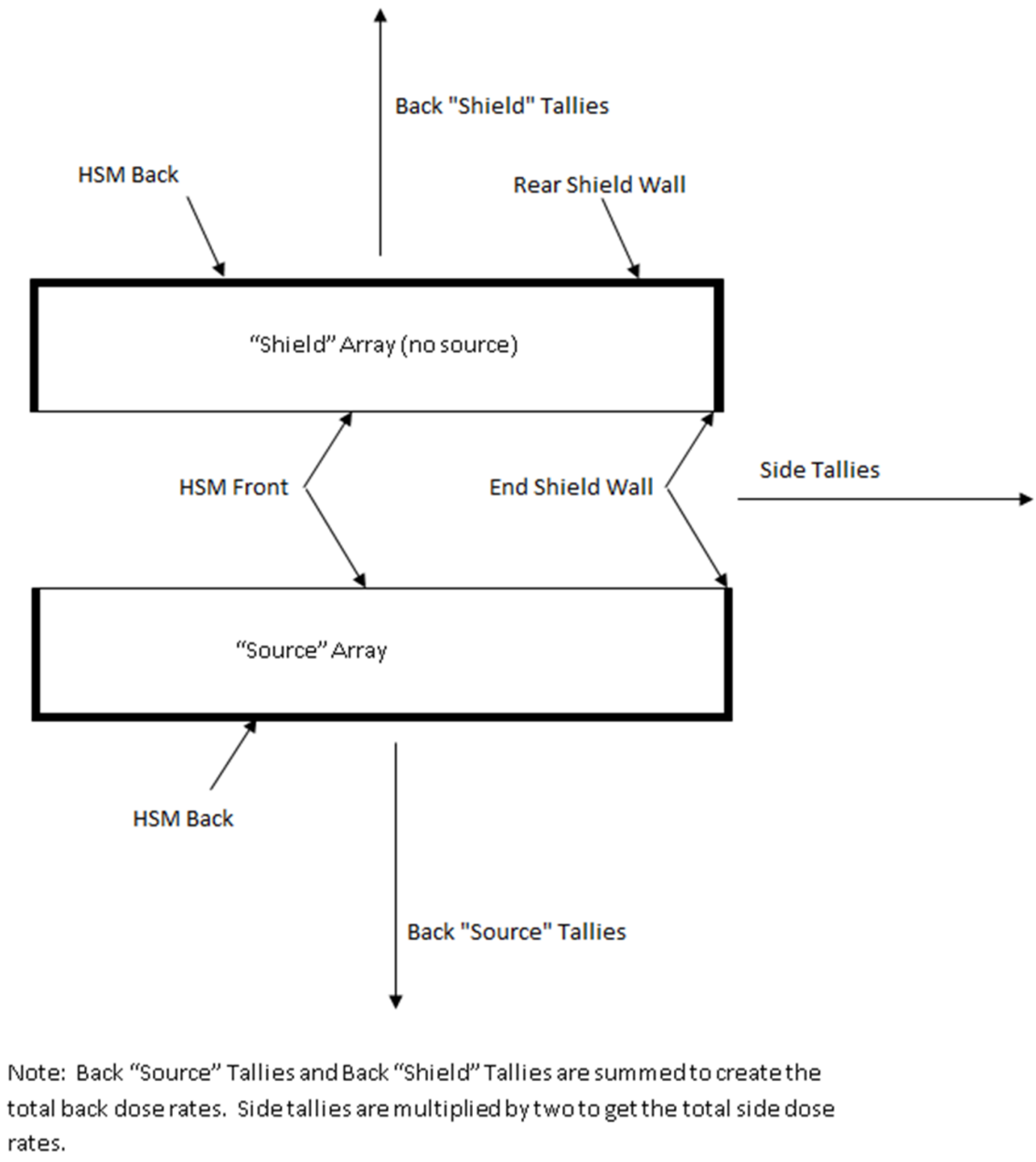


Figure 11-4
Two 1x10 ISFSI MCNP Geometry

CHAPTER 12
ACCIDENT ANALYSIS

Table of Contents

12. ACCIDENT ANALYSIS

12.1 Introduction

This Chapter describes the postulated off-normal and accident events that might occur during transfer/storage of the EOS-37PTH or EOS-89BTH dry shielded canister (DSC) to the independent spent fuel storage installation (ISFSI). This chapter also addresses the potential causes of these events, their detection and consequences, and the corrective course of action to be taken by ISFSI personnel. Off-normal and accident analyses demonstrate that the functional integrity of the system is maintained by:

- Maintaining sub-criticality within margins defined in Chapter 7.
- Maintaining confinement boundary integrity
- Ensuring fuel retrievability and
- Maintaining doses within 10 CFR 72.104 [12-1] limits for off-normal and 10 CFR 72.106 limits for accident conditions.

The accident dose calculations sections report the expected doses resulting from the postulated event in terms of whole body doses only. The leak tight DSC design and the maintenance of confinement boundary integrity under all credible off-normal and accident scenarios ensures no radiation leakage from the EOS-37PTH DSC or EOS-89BTH DSC, thereby limiting dose consequences to direct and scattered radiation doses without any associated inhalation or ingestion doses.

12.2 Off-Normal Events

Off-normal events are design events of the second type (Design Event II) as defined in ANSI/ANS 57.9 [12-2]. Design Event II conditions consist of a set of events that do not occur regularly, but can be expected to occur with a moderate frequency, or about once during a calendar year of ISFSI operation.

For the NUHOMS® EOS System, off-normal events could occur during fuel loading, trailer movement, EOS-37PTH DSC or EOS-89BTH DSC transfer and other operational events. The two off-normal events, which bound the range of off-normal conditions, are:

- A “jammed” DSC during loading or unloading from the EOS-HSM
- The extreme ambient temperatures of -40 °F (winter) and +117 °F (summer)

These two events envelop the range of expected off-normal structural loads and temperatures acting on the NUHOMS® EOS System.

12.2.1 Off-Normal Transfer Load

Although unlikely, the postulated off-normal handling event assumes that the leading edge of the DSC becomes jammed against some element of the support structure during transfer between the EOS transfer cask (TC) and the EOS horizontal storage module (HSM).

Cause of the Event

It is postulated that if the EOS-TC is not accurately aligned with respect to the EOS-HSM and as a result the DSC binds or jams during transfer operations.

The interiors of the EOS-TC and the EOS-HSM are inspected prior to transfer operations to ensure there are no obstacles. Also, the DSC has beveled lead-ins on each end, designed to avoid binding or sticking on small (less than 0.25-inch) obstacles. The EOS-TC and the DSC support structure inside the EOS-HSM are also designed with lead-ins to minimize binding or obstruction during DSC transfer. The postulated off-normal handling load event assumes that the leading edge of the DSC becomes jammed against some element of the DSC support structure because of an unlikely gross misalignment of the EOS-TC.

The interfacing dimensions of the top end of the EOS-TC and the EOS-HSM access opening sleeve are specified so that docking the EOS-TC with the EOS-HSM is not possible should gross misalignments between the EOS-TC and EOS-HSM exist.

Detection of the Event

The normal load to push/pull the DSC in and out of the EOS-TC/EOS-HSM is less than 32 kips (135 kips x 0.2/Cos 30). This movement is performed at a very low speed. System operating procedures and technical specification limits defining the safeguards to be provided ensure that the system design margins are not compromised. If the DSC were to jam or bind during transfer, the pressure for the ram increases. The off-normal load set for the "jammed DSC" for both insertion and retrieval are 135 kips and 80 kips, respectively. This load is administratively controlled to ensure that during the transfer operation this load is not exceeded.

During the transfer operation, the force exerted on the DSC by the ram is that required to first overcome the static frictional resisting force between the EOS-TC rails and the DSC. Once the DSC begins to slide, the resisting force is a function of the sliding friction coefficient between the DSC and the EOS-TC rails and/or between the DSC and the EOS-HSM support rails. If motion is prevented, the pressure increases, thereby increasing the force on the DSC until the ram system pressure limit is reached. This limit is controlled so that adequate force is available to overcome variations in surface finish, etc., but is sufficiently low to ensure that component damage does not occur.

Analysis of Effects and Consequences

The DSC and the EOS-HSM are designed and analyzed for off-normal transfer loads of 135 kips for insertion and 80 kips for retrieval during insertion and retrieval (unloading) operations. These analyses are discussed in Appendix 3.9.1 for DSC and 3.9.4 for EOS-HSM. For either loading or unloading of the DSC under off-normal conditions, the stresses on the shell assembly components are demonstrated to be within the ASME allowable stress limits. Therefore, permanent deformation of the DSC shell components does not occur. The internal basket assembly components are unaffected by these loads based on clearances provided between the basket and DSC internal cavity.

There is no breach of the confinement pressure boundary and, therefore, no potential for release of radioactive material exists.

Corrective Actions

The required corrective action is to reverse the direction of the force being applied to the DSC by the ram, and return the DSC to its previous position. Since no permanent deformation of the DSC occurs, the sliding of the DSC back to its previous position is unimpeded. The EOS-TC alignment is then rechecked, and the EOS-TC repositioned as necessary before attempts at transfer are renewed.

12.2.2 Extreme Temperature

The NUHOMS® EOS System is designed for use at ambient temperatures of -40 °F (winter) and 117 °F (summer). Even though these extreme temperatures are likely to occur for a short period of time, it is conservatively assumed that these temperatures occur for a sufficient duration to produce steady state temperature distributions in each of the affected NUHOMS® EOS System components. Each licensee should verify that this range of ambient temperatures envelopes the design basis ambient temperatures for the ISFSI site. The NUHOMS® EOS System components affected by the postulated extreme ambient temperatures are the EOS-TC and DSC during their transfer from the plant's fuel/reactor building to the ISFSI site, and the EOS-HSM during storage of a DSC.

Cause of the Event

Off-normal ambient temperatures are natural phenomena.

Detection of Event

Off-normal ambient temperature conditions are confirmed by the licensee to be bounding for their site.

Analysis of Effects and Consequences

Thermal analysis of the NUHOMS® EOS System for extreme ambient conditions is presented in Chapter 4. The effects of extreme ambient temperatures on the NUHOMS® EOS System are analyzed as follows:

Components	**SAR Sections**
EOS-37PTH DSC and EOS-89BTH DSC Shell	Appendix 3.9.1
EOS-37PTH Basket and EOS-89BTH Basket	Appendix 3.9.2
EOS-HSM	Appendix 3.9.4
EOS-TC	Appendix 3.9.5

Corrective Actions

None

12.3 Postulated Accident

The design basis accident events specified by ANSI/ANS 57.9-1984 [12-2] and other postulated accidents that may affect the normal safe operation of the NUHOMS® EOS System are addressed in this section.

The following sections provide descriptions of the analyses performed for each accident condition. The analyses demonstrate that the requirements of 10 CFR 72.122 [12-1] are met and that adequate safety margins exist for the NUHOMS® EOS System design. The resulting accident condition stresses in the NUHOMS® EOS System components are evaluated and compared with the applicable code limits set forth in Chapter 2.

Radiological calculations are performed to confirm that on-site and off-site dose rates are within acceptable limits.

The postulated accident conditions addressed in this section include:

- EOS-TC drop
- Earthquake
- Tornado wind pressure and tornado-generated missiles
- Flood
- Blockage of EOS-HSM air inlet and outlet openings
- Lightning
- Fire/Explosion

12.3.1 EOS-TC Drop

Cause of Accident

As described in Chapter 9, handling operations involving hoisting and movement of EOS-TC loaded with the EOS-37PTH DSC or EOS-89BTH DSC is typically performed inside the plant's fuel handling building. These include utilizing the crane for placement of the empty DSC into the EOS-TC cavity, lifting the EOS-TC/DSC into and out of the plant's spent fuel pool, and placement of the EOS-TC/DSC onto the transfer skid/trailer. An analysis of the plant's lifting devices used for these operations, including the crane and lifting yoke, is needed to address a postulated drop accident for the ESO-TC and its contents. The postulated drop accident scenarios addressed in the plant's 10 CFR Part 50 [12-3] licensing basis are plant-specific and should be addressed by the licensee.

Once the EOS-TC is loaded onto the transfer skid/trailer and secured, it is pulled to the EOS-HSM site by a tractor vehicle. A predetermined route is chosen to minimize the potential hazards that could occur during transfer. This movement is performed at very low speeds. System operating procedures and technical specification limits defining the safeguards to be provided ensure that the system design margins are not compromised. As a result, it is highly unlikely that any plausible incidents leading to a EOS-TC drop accident could occur. Similarly, at the ISFSI site, the transfer skid/trailer is backed-up to, and aligned with, the EOS-HSM using hydraulic positioning equipment. The EOS-TC is then docked with, and secured to, the EOS-HSM access opening. The loaded DSC is transferred to or from the EOS-HSM using a hydraulic ram system. The bolts that secure the transfer skid to the transfer trailer remain in place at all times when the transfer trailer is in motion. The EOS-TC is secured to the transfer skid. As a result, there is no reasonable way during these operations for a loaded EOS-TC drop accident to occur.

Lifts of the EOS-TC loaded with the dry storage canister are made within the existing heavy loads requirements and procedures of the licensed nuclear power plant. The EOS-TC design meets requirements of NUREG-0612 [12-4] and American National Standards Institute (ANSI) N14.6 [12-5].

The EOS-TC is transferred to the ISFSI in a horizontal configuration. Therefore, the only drop accident evaluated during storage or transfer operations is a side drop or a corner drop.

The EOS-TC and DSC are evaluated for a postulated side and corner drops to demonstrate structural integrity during transfer and plant handling.

Accident Analysis

Chapter 3 evaluates the structural integrity of the EOS-TC loaded with an EOS-37PTH DSC or EOS-89BTH DSC under two postulated accident drop scenarios during transfer using LS-DYNA.

Of the three EOS-TCs, TC108, TC125, and TC135, the TC108 is the bounding EOS-TC for analytical purposes, due to the higher accelerations expected from the lighter weight construction. The scenarios are a 65-inch drop onto the side of the loaded EOS-TC and another onto the corner of the EOS-TC. The second drop on the corner assumes a 30-degree angle from the horizontal side drop orientation. The drop height is based on the 65-inch height of the transfer trailer on which the EOS-TC is transferred.

Components	SAR Sections
EOS-TC loaded with EOS-37PTH DSC (Side and corner drop)	Appendices 3.9.1 - 3.9.3 and 3.9.5
EOS-TC loaded with EOS-89BTH DSC (Side and corner drop)	Appendices 3.9.1 - 3.9.3 and 3.9.5
EOS-37PTH DSC, PWR Fuel Cladding (Side and corner drop)	Appendix 3.9.6
EOS-89BTH DSC, BWR Fuel Cladding (Side and corner drop)	Appendix 3.9.6

All stresses are within allowable limits in both drop scenarios for the TC108, EOS-37PTH DSC and EOS-89BTH DSC. The largest strain in the basket is 0.8 % and 0.6 % for the EOS-TC108 loaded with EOS-37PTH DSC and EOS-89BTH DSC, respectively. The maximum stresses and strains in the fuel cladding for the side and corner drops remain below the applicable yield strength, therefore there is no fuel deformations.

The strain is limited in effect given the mode of deformation as the basket plates maintain their general shape. This deformation is limited and the position of the fuel assemblies is maintained from their initial positions relative to each other. The deformations of the basket plates are approximately uniform in the direction of impact and the fuel does not change configuration. Therefore, these deformations do not have an effect on criticality control.

Accident Dose Calculation

Based on analysis results presented in Appendix 3.9.3, Sections 3.9.3.3 and 3.9.3.4, the accidental EOS-TC drop scenarios do not breach the EOS-37PTH or the EOS-89BTH DSC confinement boundaries. The function of EOS-TC lead shielding is not compromised by these drops. The EOS-TC neutron shield, however, may be damaged in an accidental drop.

Dose rates are computed at 100 m from the EOS-TC with the neutron shield removed, which is the minimum allowed distance to the site boundary. As presented in Chapter 6, Table 6-54, the maximum dose rate at 100 m from an EOS-TC during a loss of neutron shield and lead slump in an accidental drop accident is 2.15 mrem/hr. If an 8-hour recovery time is assumed, the dose to an individual at the site boundary is 2.15*8 = 17 mrem, which is significantly below the 10 CFR 72.106 dose limit of 5 rem.

Corrective Actions

The DSC is inspected for damage, and the DSC opened and the fuel removed for inspection, as necessary. Removal of the EOS-TC top cover plate may require cutting of the bolts in the event of a corner drop onto the top end. These operations take place in the plant fuel building decontamination area and spent fuel pool after recovery of the EOS-TC.

Following recovery of the EOS-TC and unloading of the DSC, the EOS-TC is inspected, repaired and tested as appropriate prior to reuse.

For recovery of the EOS-TC and contents, it may be necessary to develop a special sling/lifting apparatus to move the EOS-TC from the drop site to the fuel pool. This may require several weeks of planning to ensure all steps are correctly organized. During this time, lead blankets may be added to the EOS-TC to minimize onsite exposure to site operations personnel. The EOS-TC can be roped off to ensure the safety of the site personnel.

12.3.2 Earthquake

Cause of Accident

The explicitly evaluated seismic response spectra for the NUHOMS® EOS System consist of the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.60 (Reg. Guide 1.60) [12-6] response spectra, anchored to a maximum ground acceleration of 0*.45g* horizontal and 0*.30g* for the vertical peak accelerations. The results of the frequency analysis of the EOS-HSM structure (which includes a simplified model of the DSC) yield a lowest frequency of 18.7 Hz in the transverse direction and 32.7 Hz in the longitudinal direction. The lowest vertical frequency is 60.3 Hz. Thus, based on the Reg. Guide 1.60 response spectra amplifications, *and conservatively using ZPA values of 0.50g horizontal and 0.333g vertical,* the corresponding seismic accelerations used for the *structural* design of the EOS-HSM are 0.936g and 0.628g in the transverse and longitudinal directions, respectively, and 0.333g in the vertical direction. The corresponding accelerations applicable to the DSC are 1.229g and 0.694g in the transverse and longitudinal directions, respectively, and 0.333g in the vertical direction. *Stability analyses are based on accelerations of 0.45g horizontal and 0.30g vertical.*

Accident Analysis

The seismic analyses of the components that are important to safety are analyzed as follows:

Components	**SAR Sections**
EOS-37PTH DSC and EOS-89BTH DSC Shell	Appendix 3.9.1
EOS-37PTH Basket and EOS-89BTH Basket	Appendix 3.9.2
EOS-HSM	Appendices 3.9.4 & 3.9.7
EOS-TC	Appendices 3.9.5 & 3.9.7

Note that the seismic loads are bounded by the transfer and drop loads for the basket and EOS-TC components.

The results of these analyses show that seismic stresses are well below the applicable stress limits.

Accident Dose Calculations

All the components that are important-to-safety are designed and analyzed to withstand the design basis earthquake accident. Hence, no radiation is released and there is no associated dose increase due to this event.

Corrective Actions

After a seismic event, all components are inspected for damage. Any debris is removed. An evaluation is performed to verify that the system components are still within the licensed design basis.

12.3.3 Tornado Wind and Tornado Missiles Effect on EOS-HSM

Cause of Accident

In accordance with ANSI-57.9 [12-2] and 10 CFR 72.122 [12-1], the NUHOMS® EOS System is designed for tornado effects, including tornado wind loads. In addition, the NUHOMS® EOS System is designed to withstand tornado missile effects. The NUHOMS® EOS System is designed to be located anywhere within the United States. Therefore, the most severe tornado wind and missile loadings specified by NUREG-0800 [12-7] and NRC Reg. Guide 1.76 [12-8] are selected as a design basis for this postulated accident. The determination of the tornado wind pressures and tornado missile loads acting on the NUHOMS® EOS System are detailed in Chapter 2, Section 2.3.1.

Accident Analysis

Stability and stress analyses are performed to determine the response of the EOS-HSM to tornado wind pressure loads. The stability analyses are performed using closed-form calculation methods to determine the sliding and overturning response of the EOS-HSM array. A single EOS-HSM with both the end and the rear shield walls is conservatively selected for the analyses. The stress analyses are performed using the ANSYS [12-9] finite element model of a single EOS-HSM to determine design forces and moments. These conservative generic analyses envelop the effects of wind pressures on the EOS-HSM array. These analyses are described in Appendix 3.9.7, Section 3.9.7.1. Thus, the requirements of 10 CFR 72.122 are met.

In addition, the EOS-HSM is evaluated for tornado missiles. The adequacy of the EOS-HSM to resist tornado missile loads is also addressed in Appendix 3.9.7.

Accident Dose Calculation

As shown in the above evaluations, the tornado wind and tornado missiles do not breach the EOS-HSM such that the DSC confinement boundary is compromized. Localized scabbing of the end shield wall of an EOS-HSM array may be possible.

The EOS-HSM outlet vent covers and wind deflectors (if required) may be lost due to a tornado or tornado missile event. Only the dose rates on the roof are affected, since the front, rear, and side dose rates remain the same. Information in Chapters 6 and 11 is used to determine that the EOS-HSM accident increases the average dose rate on the roof of the module to ~7400 mrem/hr.

The evaluation for the impact on public exposure, a 2 x 10 back to back array of EOS-HSMs and a distance to the site boundary of 370 m is used. As documented in Chapter 11, Section 11.3.2, for a 2x10 ISFSI configuration, the accident dose rate is approximately 1.1 mrem/hour and 0.1 mrem/hr at a distance of 200 m and 370 m respectively from the ISFSI. It is assumed that the recovery time for this accident is five days (120 hours). Therefore, the total exposure to an individual at a distance of 200 m and 370 m is 132 mrem and 12 mrem respectively. This is significantly less than the 10 CFR 72.106 limit of 5 rem.

Corrective Action

After excessive high winds or a tornado, the EOS-HSM is inspected for damage. Any debris is removed. Any damage resulting from impact with a missile is evaluated to verify that the system is still within the licensed design basis.

The need for temporary shielding is evaluated and EOS-HSM repairs are performed to return the EOS-HSM to pre-accident design conditions.

12.3.4 Tornado Wind and Tornado Missiles Effect on EOS-TC

Cause of Accident

The EOS-TC is evaluated for the tornado wind speed and missile whose design parameters are specified in Chapter 2, Section 2.3.2. The maximum design basis tornado (DBT) tornado wind speed of 230 mph in accordance with Reg. Guide 1.76, Revision 1 [12-10] was considered. The 4000 pound automobile, as well as a 287-pound schedule 40 pipe, and a 1-inch solid steel missiles are considered. The other types of missiles are enveloped by the schedule 40 pipe missile.

This analysis is performed for the EOS-TC, secured in the horizontal position on the support skid. The following criteria are used to evaluate the adequacy of the EOS-TC for the loads described above.

- Stability analysis
- Penetration resistance
- Impact stress analysis

Accident Analysis

The EOS-TC is evaluated for tornado wind and missile effects in Appendix 3.9.5 and 3.9.7.

The factor of safety on tip is *1.30* from the bounding DBT *wind plus missile load combination* on the EOS-TC while sitting on the trailer ready for transfer. The primary membrane intensity and combined membrane plus bending stresses due to DBT and missile impact are calculated, which are below the allowable stresses. The maximum missile penetration depth is found to be 0.526 inch, which is less than the thickness of EOS-TC outer shell of 1.0 inch and top cover plate thickness of 3.25 inches.

Accident Dose Calculation

Based on the above analyses, the DSC confinement boundary is not breached as a result of the missile impacts. Accordingly, no DSC damage or release of radioactivity is postulated.

The missile impact scenario may result in the loss of EOS-TC neutron shielding and local deformation/damage of the gamma shielding. The effect of loss of the neutron shielding due to a missile impact is bounded by that resulting from a EOS-TC drop scenario evaluated above in Section 12.3.1. The change in radiation dose due to local deformation/damage of the gamma shielding is negligible.

Corrective Action

After excessive high winds or a tornado, the EOS-TC is inspected for damage. These operations take place in the plant fuel building decontamination area and spent fuel pool after recovery of the EOS-TC. Following recovery of the EOS-TC and unloading of the DSC, the EOS-TC is inspected, repaired and tested as appropriate prior to reuse.

For recovery of the EOS-TC and contents, it may be necessary to develop a special sling/lifting apparatus to move the EOS-TC from the site to the fuel pool. This may require several weeks of planning to ensure all steps are correctly organized. During this time, lead blankets may be added to the EOS-TC to minimize on-site exposure to site operations personnel. The EOS-TC can be roped-off to ensure the safety of the site personnel.

12.3.5 Flood

Cause of Accident

Flooding conditions simulating a range of flood types, such as tsunami and seiches as specified in 10 CFR 72.122 (b) are considered. In addition, floods resulting from other sources, such as high water from a river or a broken dam, are postulated as the cause of the accident.

Accident Analysis

The EOS-HSM is evaluated for flooding in Appendix 3.9.4. Based on the evaluation presented in that section, the EOS-HSM can withstand the design basis flood.

Accident Dose Calculation

The radiation dose due to flooding of the EOS-HSM is negligible. Flooding does not breach the DSC confinement boundary. Therefore, radioactive material inside the DSC remains sealed in the DSC and, therefore, does not contaminate the encroaching flood water.

Corrective Actions

Because of the location and geometry of the EOS-HSM vents, it is unlikely that any significant amount of silt would enter an EOS-HSM should flooding occur. Any silt deposits are removed using a pump suction hose or fire hose inserted through the inlet vent to suck the silt out, or to produce a high velocity water flow to flush the silt through the EOS-HSM inlet vents.

12.3.6 Blockage of EOS-HSM Air Inlet and Outlet Openings or Loss/Damage of Wind Deflectors

This accident conservatively postulates the complete blockage of the ventilation air inlet and outlet openings of the EOS-HSM and complete loss of wind deflectors.

Cause of Accident

Since the EOS-HSMs are located outdoors, there is a remote probability that the ventilation air inlet and outlet openings could become blocked by debris from such unlikely events as floods and tornados. The wind deflectors could also become lost or damaged by extreme winds, tornadoes, or similar accidents. These wind deflectors are needed only for high heat load DSCs. The NUHOMS® EOS System design features, such as the perimeter security fence and the redundant protected location of the air inlet and outlet openings, reduce the probability of occurrence of such an accident. Nevertheless, for this conservative generic analysis, such an accident is postulated to occur and is analyzed.

Accident Analysis

The thermal evaluation of this event is presented in Chapter 4, Sections 4.4.5 and 4.4.11 for the EOS-37PTH DSC and EOS-89BTH DSC, respectively stored inside an EOS-HSM. The condition caused by a lost or damaged wind deflector is bounded by the blocked vent accident condition. The analysis performed for the EOS-37PTH DSC bounds the values for the EOS-89BTH DSC. Therefore, the temperatures determined for Load case #5 in Section 4.4.5, are used in the EOS-HSM structural evaluation of this event. The EOS-HSM structural analysis, presented in Appendix 3.9.4, demonstrates that the EOS-HSM component stresses remain below allowable values.

Accident Dose Calculation

There are no offsite dose consequences as a result of this accident.

Corrective Actions

Debris removal is all that is required to recover from a postulated blockage of the HSM ventilation air inlets and outlets. Cooling begins immediately following removal of the debris from the inlets and outlets. The amount and nature of debris can vary, but even in the most extreme case, manual means, or readily available equipment can be used to remove debris. The damaged or lost wind deflectors that are needed for a high heat load system must be repaired or replaced.

12.3.7 Lightning

Cause of Accident

The likelihood of lightning striking the EOS-HSM, and causing an off-normal condition is not considered to be a credible event. Lightning protection system requirements are site-specific and depend on the frequency of occurrences of lightning storms in the proposed ISFSI location, and the degree of protection offered by other grounded structures in the proximity of the EOS-HSMs. The addition of simple lightning protection equipment, required by plant criteria, to EOS-HSM structures (i.e., grounded handrails, ladders, etc.) is considered a miscellaneous attachment.

Accident Analysis

Should lightning strike in the vicinity of the EOS-HSM, the normal storage operations of the EOS-HSM are not affected. The current discharged by the lightning follows the low impedance path offered by the surrounding structures. Therefore, the EOS-HSM is not damaged by the heat or mechanical forces generated by current passing through the higher impedance concrete. Since the EOS-HSM requires no equipment for its continued operation, the resulting current surge from the lightning does not affect the normal operation of the EOS-HSM.

Corrective Actions

Since no off-normal condition develops as the result of lightning striking in the vicinity of the EOS-HSM, no corrective action is necessary. Also, there are no radiological consequences.

12.3.8 Fire/Explosion

Cause of Accident

Combustible materials are not normally stored at an ISFSI. Therefore, a credible fire is very small and of short duration caused potentially by fire or explosion from a vehicle or portable crane.

Direct engulfment of the EOS-HSM is highly unlikely. Any fire within the ISFSI boundary while the DSC is in the EOS-HSM is bounded by the fire during EOS-TC movement. The EOS-HSM concrete acts as a significant insulating fire wall to protect the DSC from the high temperatures of the fire.

Accident Analysis

The evaluation of the hypothetical fire event is presented in Chapter 4, Section 4.5.5. The thermal evaluation of the fire event is bounded by the loss of neutron shield and loss of air circulation accident. The maximum temperatures for the bounding loss of neutron shield and loss of air circulation steady-state accident condition (Load Case # 5) are presented in Chapter 4, Table 4-28, which demonstrates that the maximum component temperatures are below the allowable limits. Temperatures in this table are used for structural evaluation of the EOS-TC. The results of this EOS-TC structural analysis is presented in Appendix 3.9.5.

Accident Dose Calculation

The DSC confinement boundary is not breached as a result of the postulated fire/explosion scenario. Accordingly, no DSC damage or release of radioactivity is postulated. Because no radioactivity is released, no resultant dose increase is associated with this event.

The fire scenario may result in the loss of EOS-TC neutron shielding should the fire occur while the DSC is in the EOS-TC.

The effect of loss of the neutron shielding due to a fire is bounded by that resulting from a EOS-TC drop scenario. See Section 12.3.1 for evaluation of dose consequences of a EOS-TC drop.

Corrective Actions

Evaluation of EOS-TC neutron shield damage as a result of a fire is to be performed to assess the need for temporary shielding (if fire occurs during transfer operations) and repairs to restore the EOS-TC to pre-fire design conditions.

12.4 References

12-1 Title 10 Code of Federal Regulation Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

12-2 ANSI/ANS-57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," American National Standards Institute, American Nuclear Society.

12-3 Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities."

12-4 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1980.

12-5 ANSI N14.6-1993, "American National Standard for Special Lifting Device for Shipping Containers Weighing 10,000 lbs. or More for Nuclear Materials," American National Standards Institute.

12-6 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," U.S. Atomic Energy Commission, Revision 1, December 1973.

12-7 NUREG-0800, Standard Review Plan, Section 3.5.1.4, "Missiles Generated by Natural Phenomenon," Revision 2, U.S. Nuclear Regulatory Commission, July 1981.

12-8 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," U.S. Atomic Energy Commission, Revision 0, April 1974.

12-9 ANSYS Computer Code and User's Manual, Release 14.0.

12-10 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," U.S. Atomic Energy Commission, Revision 1, March 2007.

APPENDIX 13A

TECHNICAL SPECIFICATION BASES

TECHNICAL SPECIFICATION BASES

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B 2.0 SAFETY LIMITS (SLs)

BASES

BACKGROUND	The DRY SHIELDED CANISTER (DSC) design for the EOS System (EOS-37PTH and EOS-89BTH) requires certain limits on spent fuel parameters, including fuel type, maximum allowable enrichment prior to irradiation, maximum burnup, and minimum acceptable cooling time prior to storage in the DSC. Other important limitations are the radiological source terms from Control Components (CCs). These limitations are included in the thermal, structural, radiological, and criticality evaluations performed for these DSC designs.
APPLICABLE SAFETY ANALYSES	Various analyses have been performed that use these fuel parameters as assumptions. These assumptions are included in the thermal, criticality, structural, shielding and confinement analyses. Technical Specification Tables 1 through 8, and Figures 1 and 2 provide the key fuel parameters that require confirmation prior to DSC loading.
FUNCTIONAL AND OPERATING LIMITS VIOLATIONS	If Functional and Operating Limits are violated, the limitations on the fuel assemblies in the DSC have not been met. Actions must be taken to place the affected fuel assemblies in a safe condition. This safe condition may be established by returning the affected fuel assemblies to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to remain in the DSC if that is determined to be a safe condition. Notification of the violation of a Functional and Operating Limit to the NRC is required within 24 hours. Written reporting of the violation must be accomplished within 60 days. This notification and written report are independent of any reports and notification that may be required by 10 CFR 72.75.
REFERENCES	1. SAR Chapters 2, 4, 6 and 12.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1, 3.0.2, 3.0.4 and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the canister is in the specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, the canister may have to be placed in the spent fuel pool and unloaded. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS). The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires if the equipment remains removed from service or bypassed.

When a change in specified Condition is required to comply with Required Actions, the equipment may enter a specified Condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

This specification is not applicable to the NUHOMS® EOS System. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in specified Conditions in the applicability when an LCO is not met. It precludes placing the NUHOMS® EOS System in a specified Condition stated in that applicability (e.g., Applicability desired to be entered) when the following exist:

a. Conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the equipment being required to exit the Applicability desired, to be entered to comply with the Required Actions.

BASES

LCO 3.0.4 (continued)

Compliance with Required Actions that permit continued operation of the equipment for an unlimited period of time in specified Condition provides an acceptable level of safety for continued operation. Therefore, in such cases, entry into a specified Condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified Condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified Conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified Conditions in the Applicability that are related to the unloading of a canister.

Exceptions to LCO 3.0.4 are stated in the individual Specifications.

Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Surveillances do not have to be performed on the associated equipment out of service (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing specified Conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated out of service equipment.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or not in service in compliance with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The equipment being returned to service meets the LCO; or

b. Other equipment meets the applicable LCOs.

BASES

LCO 3.0.5 (continued)

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed required testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

LCO 3.0.6 This specification is not applicable to the NUHOMS® EOS System. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 3.0.7 This specification is not applicable to the NUHOMS® EOS System. The placeholder is retained for consistency with the power reactor technical specifications.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1, 3.2 and 3.3 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified Conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the equipment is in a specified Condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment within its LCO. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post-maintenance testing may not be possible in the current specified Conditions in the Applicability due to the necessary equipment parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function.

This will allow operation to proceed to a specified Condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR includes a Note in the Frequency stating, "SR 3.0.2 is not applicable."

BASES

SR 3.0.2 (continued)

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance. The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

BASES

SR 3.0.3 (continued)

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified Conditions in the Applicability imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility that is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered not in service or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is not in service, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance. Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified Condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to an appropriate status before entering an associated specified Condition in the Applicability. However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified Condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that Surveillances do not have to be performed on such equipment. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the

BASES

SR 3.0.4 (continued)

specified Frequency does not result in an SR 3.0.4 restriction to changing specified Conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified Condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified Conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in specified Conditions in the Applicability that are related to the unloading of an EOS-HSM or DSC.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and Conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite Condition(s) specified in a Surveillance procedure require entry into the specified Condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific Conditions needed are met. Alternatively, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SR annotation is found in Technical Specifications Section 1.4, operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

B.3.1 DSC FUEL INTEGRITY

B.3.1.1 Fuel Integrity during Drying

BASES

BACKGROUND	A DSC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield plug is then placed on the DSC. Subsequent operations involve moving the DSC to the decontamination area and removing water from the DSC (using helium as a cover gas for assisting in the drainage of bulk water). After welding and non-destructive examination of the DSC inner top cover plate, vacuum drying of the DSC is performed, and the DSC is backfilled with helium. During normal storage conditions, the fuel assemblies are stored in the DSC with an inert helium atmosphere, which results in lower fuel cladding temperatures and provides an inert atmosphere during storage conditions. DSC vacuum drying is utilized to remove residual moisture from the cavity after the DSC has been drained of water. Any water which was not drained from the DSC evaporates from fuel or basket surfaces due to the vacuum. This vacuum drying operation is aided by the temperature increase due to the heat generation of the fuel.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in a DSC is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are the fuel pellet matrix, the fuel cladding tubes in which the fuel pellets are contained, and the DSC in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This protective environment is accomplished by removing water from the DSC (using helium for assisting in the drainage of bulk water) and backfilling the DSC with helium. The removal of water is necessary to prevent phase change–related pressure increase upon heatup. The analysis in Chapter 4 demonstrates that if helium is used as a cover gas for bulk water removal operations, the conductivity of helium during vacuum drying operations provides assurance that the cladding temperature remains below the cladding temperature limit. The DSC/TC annulus contains water during the vacuum drying process. This Safety Analysis Report (SAR) evaluates and documents that the DSC confinement boundary is not compromised due to any normal, off-normal or accident condition postulated (SAR Chapter 3 and 12 structural analyses) and the fuel cladding temperature remains below allowable values (SAR Chapter 4).

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The potential exists for oxidation of fuel pellets if they are exposed to air for a sufficient duration at a high temperature. Use of helium for blowdown or draindown operations will help prevent oxidation of fuel pellets due to air by replacing air with helium, which is an inert gas.

LCO

A stable vacuum pressure of ≤3 torr further ensures that all liquid water has evaporated in the DSC cavity, and that the resulting inventory of oxidizing gases in the DSC is below 0.25 volume %.

Technical Specification 3.1.1 requires the use of helium during the bulkwater removal process. Therefore, water from the DSC cavity is replaced by helium during the bulkwater removal process. Fuel cladding temperatures are low during this short duration process due to the presence of liquid water and helium.

Therefore use of helium during bulkwater removal, vacuum drying and long-term storage operations assures that the fuel assemblies will have limited (or no) exposure to the oxidizing environment.

APPLICABILITY

This is applicable to all DSCs during LOADING OPERATIONS but before TRANSFER OPERATIONS.

ACTIONS

The actions specified require restoring the vacuum drying system to an operable status or ensuring the integrity of the DSC shell/ITCP weld or the establishment of a helium pressure of at least 0.5 atmosphere within the DSC or flooding the DSC to submerge the fuel assemblies, within 30 days. The specified value of helium atmosphere allows the transfer of decay heat from the DSC while allowing implementation of corrective actions to return the DSC to an analyzed condition. The 15 psig limit in the Actions section is conservatively below the maximum analyzed blowdown pressure. The basis for 30 days is as follows: LCO 3.1.1 requires the use of helium for all water removal from the DSC before vacuum drying. Therefore, vacuum drying operations are carried out with water replaced by helium. The SAR thermal analysis demonstrates that if helium is used as a cover gas for water removal, the conductivity of helium during vacuum drying operations assures that cladding temperatures remain below the cladding temperature limit. The DSC/TC annulus also contains water during the vacuum drying process. Because the cladding

BASES

ACTIONS (continued)

temperatures are below the cladding temperature limits, the criterion of 30 days is used as a reasonable time period for identifying and repairing vacuum drying system or seal welds.

SURVEILLANCE REQUIREMENTS	Ensure that vacuum pressure remains sufficiently low for a sufficient timeframe to provide a minimum oxidizing gas content, and to ensure that the DSC is dry.
REFERENCES	1. SAR Chapters 3, 4, and 12.

B.3.1 DSC FUEL INTEGRITY

B.3.1.2 DSC Helium Backfill Pressure

BASES

BACKGROUND	A DSC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield plug is then placed on the DSC. Subsequent operations involve moving the DSC to the decontamination area and removing water from the DSC using helium to assist in the drainage of bulk water. After the DSC inner top cover is welded, vacuum drying of the DSC is performed and the DSC is backfilled with helium, resulting in lower fuel cladding temperatures. In addition, it provides an inert atmosphere during storage conditions. The inert helium environment protects the fuel from potential oxidizing environments.
APPLICABLE SAFETY ANALYSIS	Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. SAR Section 3.5 evaluates the effect of long-term storage and short term temperature transients on fuel cladding integrity. Credit for the helium backfill pressure is taken to limit the potential for corrosion of the fuel cladding. Leak testing of the DSC welds is performed to ensure that an inert helium atmosphere will be maintained within the DSC, surrounding the fuel, to limit radiological consequences. SAR Chapter 4 evaluates the DSC maximum pressure under normal, off-normal, and accident conditions.
LCO	DSC backfill pressure is maintained within a range of pressure during initial backfill that will ensure maintenance of the helium backfill pressure over time and will not result in excessive DSC pressure in normal, off-normal and accident conditions.
APPLICABILITY	This specification is applicable to all DSCs during LOADING OPERATIONS but before TRANSFER OPERATIONS.

BASES

ACTIONS	The actions required and associated Completion Times are associated with ensuring that the DSC remains in a safe condition and within its design pressure limits and time limits established in the SAR. These limits are imposed to ensure that the DSC confinement integrity is maintained. The thermal analysis in Chapter 4 demonstrates that with water in the DSC/cask annulus and helium atmosphere in the DSC cavity, fuel cladding temperatures are below the cladding material temperature limits. Note that no credit is taken for any convection of helium in the DSC cavity. These time limits are imposed to ensure that there is sufficient time to complete the required actions for identifying and repairing vacuum drying system or seal welds. The fuel cladding will not exceed maximum allowable temperatures during this time.
SURVEILLANCE REQUIREMENTS	The DSC backfill pressure is monitored during the initial DSC loading to ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert helium gas; and (2) the helium backfill pressure level maintains a helium atmosphere that is favorable for the transfer of decay heat and consistent with the SAR thermal analysis.
REFERENCES	1. SAR Chapters 3 and 4.

B.3.1 DSC FUEL INTEGRITY

B.3.1.3 Time Limit for Completion of DSC Transfer

BASES

BACKGROUND	After a DSC has been loaded with fuel assemblies, vacuum dried and sealed, it is ready for transfer to the ISFSI. The design of a loaded TC/DSC system provides sufficient passive heat rejection capacity to ensure that the integrity of the fuel cladding is maintained provided the specified time limits for completion of the transfer are met.
APPLICABLE SAFETY ANALYSIS	Long-term integrity of the fuel cladding depends on storage in an inert atmosphere and maintaining fuel cladding temperature below an acceptable limit. The TC/DSC transient thermal analysis provided in Chapter 4 of the SAR evaluates the fuel cladding temperatures under normal, off-normal, and accident conditions during the transfer of a loaded TC/DSC. The time limits for transfer operations are based on the EOS-37PTH DSC in the TC125 with the maximum allowable heat load of 50 kW. The use of these time limits during the transfer operation is bounding for all DSC/EOS-TC configurations. Longer time limits can be calculated based on the as loaded heat load of the DSC.
LCO	The time to complete the transfer of a loaded TC/DSC is monitored to ensure that the fuel cladding does not exceed the NUREG-1536, Revision 1, limit of 752 °F during transfer.
APPLICABILITY	This specification is applicable to a loaded NUHOMS® EOS-37PTH or EOS-89BTH DSC when transferred in an EOS-TC. Those DSCs with lower heat loads as identified in LCO 3.1.3 have no time limit for completion of DSC transfer. The thermal analysis performed at those lower heat load as documented in SAR Chapter 4 demonstrate that the steady state cladding temperatures during TRANSFER OPERATIONS are below the cladding temperature limit. Those DSCs with higher heat loads identified in LCO 3.1.3 have associated time limits for the DSC transfer based on the associated thermal analyses.
ACTIONS	The actions required and the specified Completion Time of 2 hours are associated with ensuring that the fuel cladding does not exceed 752 °F during transfer.

BASES

SURVEILLANCE REQUIREMENTS	The specified monitoring of the time duration for the completion of the transfer step ensures that the fuel cladding temperatures remain below the regulatory limit of 752 °F during this operation.
REFERENCES	1. SAR Chapter 4.

B.3.2 CASK CRITICALITY CONTROL

B.3.2.1 Soluble Boron Concentration

BASES

BACKGROUND	During loading and unloading of an EOS-37PTH DSC, the DSC cavity is filled with borated water having a minimum boron concentration, which is a function of the DSC basket type, fuel assembly class, and maximum planar average initial enrichment. This specification ensures that a subcritical configuration is maintained in the event of an accidental loading of a DSC with unirradiated fuel.
APPLICABLE SAFETY ANALYSIS	The EOS-37PTH DSC has been designed for unirradiated fuel with a specified maximum planar average initial enrichment while taking credit for the soluble boron concentration in the DSC cavity water and the boron content in the neutron absorber plates. The criticality analysis provided in Chapter 7 of the SAR evaluates the DSC to ensure that a subcritical configuration is maintained.
LCO	The minimum boron concentration limits of the water in the DSC cavity as specified in the LCO ensure that a subcritical configuration is maintained in the event of an accidental loading of a DSC with unirradiated fuel.
APPLICABILITY	This specification is applicable to the EOS-37PTH DSC during LOADING OPERATIONS and UNLOADING OPERATIONS.
ACTIONS	The actions required and the specified Completion Times for the required actions are associated with ensuring that either the dissolved boron concentration is restored above the specified minimum or the fuel is removed from the DSC.
SURVEILLANCE REQUIREMENTS	Performance of two separate independent analysis of the water used to fill the DSC cavity (a) within 4 hours of initiation of loading/unloading operations and (b) subsequent analysis at intervals not exceeding 48 hours until the conclusion of such loading/unloading operations provides assurance that a subcritical DSC configuration is always maintained.

BASES

REFERENCES 1. SAR Chapter 7.

B.3.3 RADIATION PROTECTION

B.3.3.1 DSC and TRANSFER CASK (TC) Surface Contamination

BASES

BACKGROUND	Since the TC with DSC in its interior is placed in the spent fuel pool in order to load the spent fuel assemblies, the exterior of the surface of the TC and the outer top surface of the DSC may become contaminated from radioactive material in the spent fuel pool water. The TC/DSC annulus is filled with clean water and sealed prior to placement in the spent fuel pool; therefore, only the outer top 1-foot surface of the DSC and the inner surface of the TC is susceptible to contamination by the water from the spent fuel pool. After placing the top shield plug onto the DSC, the loaded DSC with TC is lifted out of the pool into the decontamination area. The outer surface of the TC is decontaminated. Following sealing of the inner top cover plate, vacuum drying and backfill with helium, the DSC outer top cover plate is installed and sealed. After the draining of the TC/DSC annulus, the DSC smearable surface contamination on the outer top 1-foot surface of the DSC and the exterior surface of the TC are checked. Contamination on these surfaces is removed to a level that is as low as reasonably achievable (ALARA) and below the LCO limits in order to minimize radioactive contamination to personnel and the environment.
APPLICABLE SAFETY ANALYSYS	This radiation protection measure assures that the surfaces of the TC and the DSC have been decontaminated. This keeps the dose to occupational personnel ALARA.
LCO	The contamination limits on the outer top 1-foot surface of the DSC and the exterior surface of the TC are based on the allowed removable external radioactive contamination specified for spent fuel shipping containers in 49 CFR 173.443 (as referenced in 10 CFR 71.87(i)). Consequently, these contamination levels are considered acceptable for exposure to the general environment. This level will also ensure that the contamination levels of the inner surfaces of the HSM and potential releases of radioactive material to the environment are minimized. In addition, the NUHOMS® EOS storage system provides significant additional protection for the DSC surface than the transportation configuration. The HSM will protect the DSC from direct exposure to the elements and will, therefore, limit potential releases of removable contamination. The probability of any removable contamination being entrapped in the HSM airflow path released outside the HSM is considered extremely small.

BASES

LCO (continued)	The use of an inflatable seal in the upper cask liner recess ensures that the TC/DSC annulus area below the seal remains clean during the subsequent fuel loading steps inside the spent fuel pool. Hence, the only area that needs to be checked for contamination on the DSC is the top 1 foot of the DSC external surface. However, in the unlikely event that contamination is found that exceeds the specified levels, the entire length of the DSC surface will be checked and decontaminated. The number and location of surface swipes used to determine compliance for this LCO for both the exterior surface of the TC and outer top 1-foot surface of the DSC is based on standard industry practice and the licensee's plant-specific radiation protection program.
APPLICABILITY	Measurement and comparison of the removable contamination levels for both the TC and the outer top 1-foot surface of the DSC is performed during LOADING OPERATIONS.
ACTIONS	A note has been added that a separate Condition entry is allowed for each DSC and TC. Separate ACTIONS are provided for the DSC outer top surface and the exterior surface of the TC. If the removable surface contamination is not within the LCO limits, action must be taken to decontaminate either the DSC or TC, as appropriate, to bring the contamination level to within the limits. The Completion Time of 7 days and Prior to TRANSFER OPERATIONS is appropriate given that sufficient time is required to prepare for and perform the decontamination once the limit has been determined to be exceeded.
SURVEILLANCE REQUIREMENTS	The measurement of the removable surface contamination on both the TC and the DSC is performed once, prior to TRANFER OPERATIONS, to verify it is less than the established LCO limits. This Frequency is necessary in order to confirm that the loaded TC can be moved safely to the ISFSI without releasing loose contamination to the environment or causing excessive operational doses to personnel.
REFERENCES	None.

CHAPTER 14
QUALITY ASSURANCE

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14. QUALITY ASSURANCE

AREVA TN Americas' (an operating division of AREVA Inc.) Quality Assurance (QA) program has been established in accordance with the requirements of 10 CFR 72, Subpart G [14-1]. The QA program applies to the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of the NUHOMS® EOS System and components identified as "important-to-safety" and "safety-related." These components and systems are defined in Chapter 2.

14.1 Introduction

The complete description and specific commitments of the QA program are contained in the QA program description manual [14-2]. This manual has been approved by the U.S. Nuclear Regulatory Commission (NRC) for performing 10 CFR Part 72-related activities. Changes to the program will be submitted to the NRC for approval within 30 days of implementation. Changes to the QA program, which decrease or delete previously approved QA commitments will be submitted to the NRC for approval prior to implementation.

The matrix in Table 14-1 shows the 10 CFR 72, Subpart G criteria and the respective sections of the QA program description manual that address the criteria.

Figure 14-1 shows the organizational structure for the NUHOMS® EOS System project.

14.2 “Important-to-Safety and “Safety-Related” NUHOMS® EOS System Components

AREVA TN Americas will apply its QA program to the NUHOMS® EOS System components within its scope of responsibility, which are defined as “important-to-safety” and “ safety-related,” as delineated in Section 2.1. Quality assurance procedures are used to establish the quality category of components, subassemblies, and piece parts according to the importance to safety of each item.

In Table 2-1, each component is identified as “important-to-safety,” “not important-to-safety,” or “safety-related”. During the design process, items that are considered “important-to-safety” are further categorized using a graded quality approach. When the graded quality approach is used, a list will be developed for each “important-to-safety” item, which includes an assigned quality category consistent with the importance to safety of that item. Quality categories are determined based on the following description of categories, and on the guidance provided in NUREG/CR-6407 [14-3]:

- Category A items are critical to safe operation. These items include structures, components, and systems whose failure or malfunction could directly result in a condition adversely affecting public health and safety. This would include conditions such as loss of primary containment with subsequent release of radioactive material, loss of shielding, or an unsafe geometry compromising criticality control.
- Category B items have a major impact on safety. These items include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. An unsafe operation could result only if a primary event occurs in conjunction with a secondary event, or other failure or environmental occurrence.
- Category C items have a minor impact on safety. These items include structures, components, and systems whose failure or malfunction would not significantly reduce the packaging effectiveness and would be unlikely to create a condition adversely affecting public health and safety.

For “safety-related” items, the QA program is applied as described for Category A items. The QA program, as described in Section 14.3, is applied to each “important-to-safety” graded category and is limited as follows.

Category A

A. The design is based on the most stringent industrial codes or standards. Design verification will be accomplished by prototype testing or formal design review.

B. Vendors for items and services for this category may only be selected from the Approved Suppliers List.

C. AREVA TN Americas suppliers and sub-tier suppliers must have a QA program based on applicable criteria in Subpart G to 10 CFR Part 72, or equivalent.

D. Complete traceability of raw materials and the use of certified welders and processes is required.

E. All personnel performing QA-related inspections, tests, and examinations will be qualified and certified in accordance with the requirements of the QA program.

F. Only qualified and certified auditors and lead auditors will perform audits.

G. AREVA TN Americas QA personnel will be required to inspect and/or approve supplier-fabricated components prior to authorizing shipment release.

H. Welding consumables will be procured as a Category A item if the intended use is unknown. If purchased for a specific Category B or C application, the material must be identified, and its use restricted to fabrication of the same level.

Category B

A. The design is based on the most stringent industrial codes and standards. But design verification may be accomplished by use of alternate calculations or computer codes.

B. The procurement of items may be from suppliers on the Approved Suppliers List or QA program requirements for the supplier may be based on the inspection and test requirements of the procured item.

C. Traceability of materials is not required; however, specified welds require completion by qualified, certified welders.

D. Quality Assurance verification activities will be performed by personnel qualified and certified in accordance with the requirements of the QA program.

E. Only lead auditor personnel require certification in accordance with the QA program.

Category C

A. Items may be purchased from a catalog or “off-the-shelf.”

B. When received, the item will be identified and checked for both compliance with the purchase order and for damage.

Items not considered important-to-safety will be controlled in accordance with good industrial practices.

If a utility elects to perform construction, and has an NRC-approved QA program (10 CFR 50) [14-4] that is equivalent to or exceeds AREVA TN Americas’ program, then the utility QA program is considered an acceptable substitute for their scope of responsibility.

14.3 Description of AREVA TN Americas 10 CFR 72, Subpart G QA Program

14.3.1 Project Organization

The NUHOMS® EOS System has been designed by a dedicated AREVA TN Americas project organization.

Quality assurance (QA) duties are performed by the AREVA TN Americas project organization, the Senior Manager, Quality Assurance, and the Quality Assurance staff.

The organization structure for the NUHOMS® EOS System project is presented in Figure 14-1. A description of AREVA TN Americas organizational structure, functional responsibilities, levels of authority, and lines of internal and external (client and supplier) communication may be found in the QA program description manual.

Project QA controls are determined by the Project Manager and approved by QA. All Project Plans, regardless of the indicated applicability of QA requirements, are reviewed by QA to verify that QA controls are commensurate with the specific activity, item complexity, importance to safety and client-imposed contractual requirements.

Project personnel are indoctrinated, trained, and qualified in accordance with the QA program.

14.3.2 QA Program

AREVA TN Americas will apply the QA program to components defined in Table 2-1 as “important-to-safety” and “safety-related,” in accordance with the QA program description manual.

AREVA TN Americas has established and implemented a QA program for the control of quality in the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of storage containers for nuclear products. Training and/or evaluation of personnel qualifications in accordance with written procedures are required for personnel performing activities affecting quality. The QA program verifies that all quality requirements, engineering specifications and specific provisions of any package design approval are met. The characteristics that are critical to safety are emphasized.

The AREVA TN Americas Senior Manager, Quality Assurance regularly evaluates the QA program for adherence to the 18-point criteria in scope, implementation and effectiveness. In addition, the Senior Vice President , AREVA Inc. / AREVA TN Americas requires that the QA program, including the QA program description manual and associated implementing procedures, be implemented and enforced on all applicable projects at AREVA TN Americas.

Annually, a Management Audit of the Quality Assurance organization is conducted by an organization independent of the AREVA TN Americas Quality Assurance organization.

14.3.3 Design Control

"Important-to-safety" and "safety-related" NUHOMS® EOS System design activities will be implemented in accordance with the QA program. Design verification will be performed by a competent individual with the appropriate skill level. However, the skill level of this individual may not be the same as the originator, but must be equivalent.

Errors and deficiencies in the design, including the design process, are documented in the form of Corrective Action Reports.

Industry standards and specifications are used for the selection of suitable materials, parts, equipment and processes for "important-to-safety" and "safety-related" structures, systems, and components, as defined in the various chapters and sections of this safety analysis report.

14.3.4 Procurement Document Control

Procurement documents are prepared in accordance with the QA program, which delineates the actions to be accomplished in the preparation, review, approval, and control of procurement documents. Review and approval of procurement documents by QA are indicated on the procurement documents, prior to release, to confirm the adequacy of quality requirements stated therein. This review determines that quality requirements are correctly stated, inspectable, and controllable; that there are adequate acceptance and rejection criteria; and that the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements. Refer to Section 14.2 for supplier selection requirements.

The procurement documents will identify the documentation required to be submitted for information, review, or approval by AREVA TN Americas or a client of AREVA TN Americas. The time of submittal will also be established. When AREVA TN Americas requires the supplier to maintain specific QA records, the retention times and disposition requirements will be prescribed.

When purchasing commercial calibration and testing services subject to the requirements for commercial- grade dedication, AREVA TN Americas procurement documents include necessary technical and quality requirements based on NRC endorsed industry guidance.

When applicable, AREVA TN Americas procurement documents include the reporting requirements of 10 CFR Part 21 for the Reporting of Defects and Noncompliances.

14.3.5 Procedures, Instructions, and Drawings

As required by the QA program, activities affecting quality are prescribed in approved, written procedures, instructions, or drawings and these procedures, instructions, and drawings will be followed.

14.3.6 Document Control

The issuance, distribution, and receipt of documents that prescribe activities affecting quality are controlled in accordance with the QA program. Controlled documents include, but are not limited to, the AREVA TN Americas' design specifications and criteria documents, drawings, instructions, and test procedures.

The individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto are identified in the "Responsibilities" sections of the AREVA TN Americas' QA program implementing procedures.

14.3.7 Control of Purchased Items and Services

The control of purchased items and services will be implemented in accordance with the QA program.

Surveillance of subcontracted activities is planned and performed in accordance with written procedures to ensure conformance to the purchase order. These procedures provide for instructions that specify the characteristics to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and the individuals responsible for implementing these instructions.

AREVA TN Americas' suppliers will furnish documentation that identifies any procurement requirements that have not been met, together with a description of those nonconformances dispositioned as "use-as-is" or "repair."

Documentation from AREVA TN Americas' suppliers that demonstrates compliance with procurement requirements (such as material test reports, NDE results, performance test results, etc.) is periodically evaluated by audits, independent inspections, or tests, as necessary, to verify its validity.

14.3.8 Identification and Control of Materials, Parts, and Components

Materials, parts, and components will be identified and controlled in accordance with the QA program. Hardware identification requirements are determined during the generation of design drawings and specifications in such a way that the location and method of identification do not affect the form, fit, function, or quality of the item being identified.

14.3.9 Control of Special Processes

The control of special processes, such as nondestructive examination, chemical cleaning, welding, and heat-treating will be performed in accordance with the QA program.

14.3.10 Inspection

Receipt inspections, and in-process and final inspections of AREVA TN Americas-fabricated, -constructed, or -erected items, systems, components, or structures will be performed in accordance with the QA program.

14.3.11 Test Control

Test control will be accomplished in accordance with the QA program.

14.3.12 Control of Measuring and Test Equipment

The QA program defines the requirements for calibration of measuring and test equipment. Calibration is against certified measurement standards that have known relationships to national standards, where such standards exist. Where such standards do not exist, the basis for calibration will be documented.

14.3.13 Handling, Storage and Shipping

Handling, storage, and shipping will be conducted in accordance with the QA program. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by qualified individuals in accordance with predetermined work and inspection instructions.

14.3.14 Inspection and Test Status

The use of inspection and test status tags will be implemented in accordance with the QA program.

14.3.15 Control of Nonconforming Items

The QA program defines the requirements and assigns the responsibilities for the control, identification, segregation, documentation, and close-out of nonconforming items to prevent their inadvertent installation or use in fabrication, construction, or erection.

Nonconformance reports identify the item description and quantity, the disposition of the nonconformance, the inspection requirements, and signature approval of the disposition. They are periodically analyzed to show quality trends and help identify root causes of nonconformances. Significant results are reported to responsible management for review and assessment.

Nonconforming items are segregated from acceptable items and tagged to prevent inadvertent use until properly dispositioned and closed out.

14.3.16 Corrective Action

Corrective action for conditions adverse to quality will be taken in accordance with the QA program. For significant conditions adverse to quality, the cause is determined and action to preclude recurrence is taken and reported to the appropriate levels of management.

14.3.17 Records

The QA program defines the scope of the records program so that sufficient records are maintained to provide documentary evidence of the quality of items and activities affecting quality.

14.3.18 Audits and Surveillances

A comprehensive system of planned and documented audits, including audits of suppliers and site construction activities, verifies compliance with all aspects of the QA program and determines the effectiveness of the program.

Audits are performed by certified lead auditors and are planned, performed, and documented in accordance with the QA program.

The AREVA TN Americas Senior Manager, Quality Assurance or his designee may perform unannounced QA surveillances on activities affecting quality, on an as-needed basis, to further ensure compliance with QA requirements.

14.4 Conditions of Approval Records

As required by 10 CFR 72, Subpart L, AREVA TN Americas will establish and maintain records for each storage component fabricated under a certificate of compliance as required by §72.234(d). The records will be available for inspection as required by §72.234(e). Written procedures and appropriate tests will be established prior to use of the storage components, which will be provided to each NUHOMS® EOS System user as required by §72.234(f).

14.5 Supplemental Information

14.5.1 References

14-1 Title 10 Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

14-2 "AREVA Inc. Quality Assurance Program Description Manual for 10 CFR Part 71, Subpart H and 10 CFR Part 72, Subpart G," current revision.

14-3 NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," U.S. Nuclear Regulatory Commission, February 1996.

14-4 Title 10 Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities."

Table 14-1
Quality Assurance Program Description Manual Sections

10 CFR 72, Subpart G	QA Program Section
.142	1.0 Organization
.144	2.0 Quality Assurance Program
.146	3.0 Design Control
.148	4.0 Procurement Document Control
.150	5.0 Instructions, Procedures and Drawings
.152	6.0 Document Control
.154	7.0 Control of Purchased Material, Equipment and Services
.156	8.0 Identification and Control of Materials, Parts and Components
.158	9.0 Control of Special Processes
.160	10.0 Inspection
.162	11.0 Test Control
.164	12.0 Control of Measuring and Test Equipment
.166	13.0 Handling, Storage and Shipping
.168	14.0 Inspection, Test and Operating Status
.170	15.0 Nonconforming Material, Parts or Components
.172	16.0 Corrective Action
.174	17.0 Quality Assurance Records
.176	18.0 Audits

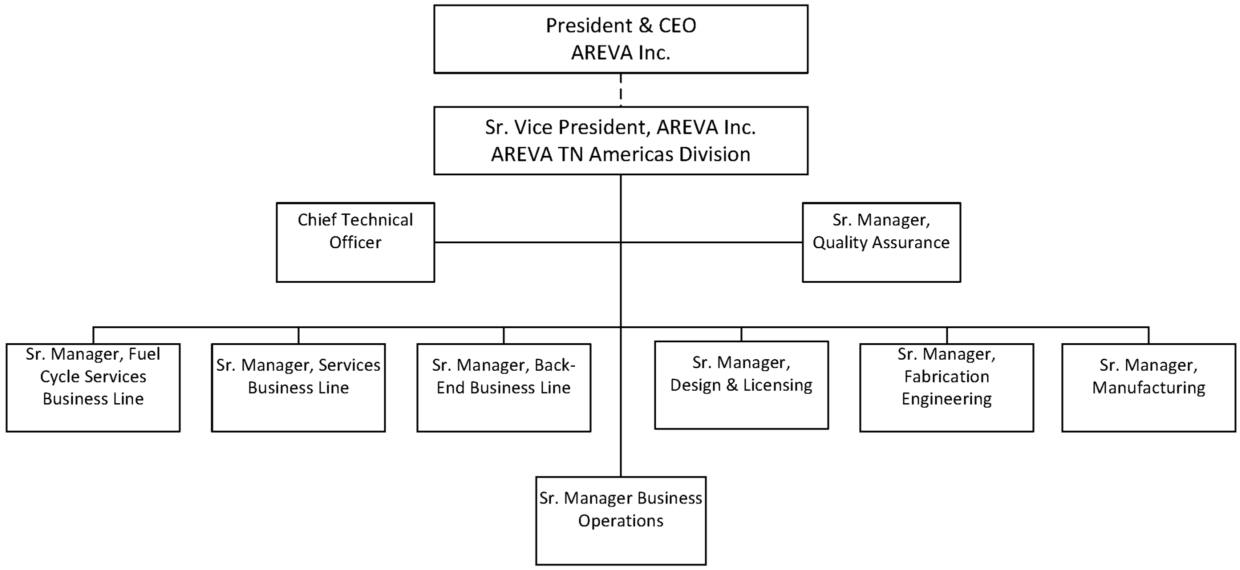

Figure 14-1
AREVA TN Americas Functional Organization for Quality Assurance Program Activities