Enclosure 8 to TN E-54364

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(Public Version)

1.1 Introduction

The type of fuel to be stored in the NUHOMS® EOS System is light water reactor (LWR) fuel of the PWR and BWR type. The EOS-37PTH DSC is designed to accommodate up to 37 intact PWR FAs with uranium dioxide (UO_2) fuel, zirconium alloy cladding, and with or without control components (CCs). *The EOS-37PTH DSC is also designed to accommodate up to eight damaged FAs or up to four failed fuel canisters (FFCs) with the balance intact FAs.* The EOS-89BTH DSC is designed to accommodate up to 89 intact BWR FAs with uranium dioxide (UO_2) fuel, zirconium alloy cladding, and with or without fuel channels. The physical and radiological characteristics of these payloads are provided in Chapter 2.

The NUHOMS® EOS System consists of the following components as shown in Figure 1-1 through Figure 1-7:

- Two dual-purpose (storage and transportation) DSCs that provide confinement in an inert environment, structural support and criticality control for the FAs; the EOS-37PTH DSC and the EOS-89BTH DSC. The DSC shells are welded stainless or duplex steel pressure vessels that includes thick shield plugs at either end to maintain occupational exposures as low as reasonably achievable (ALARA).
- *Six* EOS-37PTH DSC basket designs. *Basket Types 1 through 3 correlate with the respective HLZCs 1 through 3 (Figures 1A through 1C of the Technical Specifications [1-7]). Basket Type 4 incorporates a plate configuration that offsets the aluminum plates to allow for damaged/failed fuel storage in the EOS-37PTH DSC. The Type 4 basket has two options. The Type 4H basket is fabricated from a coated steel plate for higher emissivity and higher conductivity poison plate, while the Type 4L basket has a low emissivity coated steel plate and a low conductivity poison plate. These requirements are further detailed in the material and design limits discussed in Section 4.2 and Section 10.1. The Type 5 basket is similar to the Type 1/2/3 basket in configuration, but also incorporates the low emissivity coated steel plates and low conductivity poison plate. The maximum heat loads and the allowable HLZCs for Basket Types 4 and 5 are listed in Table 1-2. Each of these basket types also allows for two levels of boron loading in the poison plates (A and B). Each basket type is designated as follows:*

EOS-37PTH Basket Types

Neutron Poison Loading Option	***TYPE 1 (HLZC 1 max. 50 kW)***	***TYPE 2 (HLZC 2 max. 41.8 kW)***	***TYPE 3 (HLZC 3 (max. 36.35 kW)***	***TYPE 4 (Damaged/ Failed Fuel)***	***TYPE 5 (Low K, Low ϵ)***
A (Low B-10)	*A1*	*A2*	*A3*	*A4L*	*A5*
B (High B-10)	*B1*	*B2*	*B3*	*B4L*	*B5*

All Boxed and/or Shaded Changes are in response to Revised RAI 7-1

Proprietary and Security Related Information
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4.7.1.3 Quantity of Initial Helium Backfill Gas in the DSC Cavity

The free volume in the EOS-37PTH DSC cavity is assumed to be filled with 3.5 psig (18.2 psia) of helium. Based on the evaluations performed for the loading operations in Section 4.5.11, a bounding (lowest) average temperature of 303 °F (424 K) is determined for the EOS-37PTH DSC cavity gas for the backfilling operation. This temperature is used to determine the quantity of helium backfill gas in the DSC cavity in accordance with ideal gas law ($PV = nRT$). The bounding quantity of helium in the EOS-37PTH DSC cavity due to the initial backfill is summarized in Table 4-45.

4.7.1.4 Quantity of Initial Fill Gas in Fuel Rods

Based on the plenum volume, initial fuel rod fill pressure and initial temperature of fill gas in the fuel rod plenum noted earlier, the quantity of helium fill gas within the fuel rods is computed using the ideal gas law ($PV = nRT$). The bounding quantity of helium within the fuel rods for the bounding FAs in the EOS-37PTH DSC are summarized in Table 4-45 for normal, off-normal, and accident conditions based on 1%, 10%, and 100% rod rupture percentage, respectively.

4.7.1.5 Quantity of Irradiation Gases in Fuel Rods

For the EOS-37PTH DSC, the quantities of irradiation gases in the fuel rods for the bounding FAs for short and medium DSC configurations are *54.3* and *59.4* moles, respectively, as shown in Section 6.2.7. The irradiation gases are from both the FAs and control components based on a maximum burnup of 62 GWd/MTU. Considering 30% of the irradiation gases are released into the plenum, the total quantities of irradiation gases released per DSC are summarized in Table 4-45 for normal, off-normal, and accident conditions based on 1%, 10%, and 100% rod rupture percentage, respectively.

4.7.1.6 Total Amount of Gases with the EOS-37PTH DSC Cavity

The total amount of gases within the DSC cavity for normal, off-normal, and accident conditions is the sum of the initial helium backfill gas in the DSC cavity noted in Section 4.7.1.3, initial fill gas in the fuel rods released into the DSC cavity from Section 4.7.1.4, and irradiation gases released into the DSC cavity from Section 4.7.1.5.

The total amount of gases within the EOS-37PTH DSC cavity for normal, off-normal, and accident operations are summarized in Table 4-45.

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Table 4-45
Maximum Internal Pressures in the EOS-37PTH DSC

Operating Conditions		Free Volume in DSC Cavity (in^3)	Helium Backfill Amount (mol)	Plenum Volume(1) (in^3)	Fuel Rod Fill Gas Amount(2) (mol)	Fuel Rod Fission Gases Amount(2) (mol)	Total Gas Amount (mol)	Average Temperature of Helium in DSC (K)	Calculated Pressure (psig)	Pressure Used for Structural Evaluation (psig)
Symbols		V_{total}	$n_{He\_backfill}$	$f \times V_{plenum}$	$f \times n_{He\_fuel\_rod}$	$f \times n_{fission\_gas}$	n_{total}	T_{He_DSC}	P_{DSC}	
Short	Normal	329,937	192.42	67	1.44	*6.03*	*199.9*	565	10.5	20
	Off-normal	330,543	192.42	673	14.44	*60.27*	*267.1*	565	*18.9*	20
	Accident	336,599	192.42	6,729	144.42	*602.73*	*939.6*	653	*119.5*	130
Medium	Normal	352,613	205.65	76	1.51	*6.59*	*213.8*	561	10.3	20
	Off-normal	353,298	205.65	761	15.10	*65.93*	*286.7*	561	18.8	20
	Accident	360,147	205.65	7,611	151.04	*659.34*	*1,016.0*	649	*119.9*	130

Notes:

1. Plenum volumes released for normal, off-normal, and accident conditions are calculated based on the assuming rupture of 1%, 10%, and 100% of the fuel rods, respectively.
2. Quantities of initial fill and irradiation gases for normal, off-normal, and accident conditions are calculated based on the assuming rupture of 1%, 10%, and 100% of the fuel rods, respectively.

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Table 4-46
Maximum Internal Pressures in the EOS-89BTH DSC

Operating Conditions	Free Volume in DSC Cavity (in^3)	Helium Backfill Amount (mol)	Plenum Volume[1] (in^3)	Fuel Rod Fill Gas Amount[2] (mol)	Fuel Rod Fission Gases Amount[2] (mol)	Total Gas Amount (mol)	Average Temperature of Helium in DSC (K)	Calculated Pressure (psig)	Used for Structural Evaluation Pressure (psig)
Symbols	V_{total}	$n_{He\_backfill}$	$f \times V_{plenum}$	$f \times n_{He\_fuel\_rod}$	$f \times n_{fission\_gas}$	n_{total}	T_{He_DSC}	P_{DSC}	
Normal	367505	215.6	190.1	1.4	*4.6*	*221.6*	572	*10.7*	20
Off-normal	369216	215.6	1901.0	14.1	*46.5*	*276.1*	572	*16.8*	20
Accident	386326	215.6	19010	140.5	*464.6*	*820.6*	671	*90.2*	130

Notes:

(1) Plenum volumes released for normal, off-normal, and accident conditions are calculated based on the assuming rupture of 1%, 10%, and 100% of the fuel rods, respectively.

(2) Quantities of initial fill and irradiation gases for normal, off-normal, and accident conditions are calculated based on the assuming rupture of 1%, 10%, and 100% of the fuel rods, respectively.

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For the bounding normal condition with HLZCs 4 and 5, the average helium gas temperature within the DSC cavity is 557 K and 547 K, respectively. Both of these temperatures are lower compared to the design basis value of 565 K (see Table 4-45 of Section 4.7) used to evaluate the internal pressure. Therefore, there is no impact on the internal pressure for normal and off-normal conditions.

For the bounding accident condition, the average helium temperature is 657 K and higher compared to the design basis value of 653 K (see Table 4-45 of Section 4.7) used to evaluate the internal pressure. Based on the methodology in Section 4.7.1, the maximum accident internal pressure is re-computed to be *120.6* psig and remains below the maximum internal pressure limit of 130 psig for accident conditions. Therefore, the design criteria for internal pressure are satisfied.

Based on this discussion, all the temperature criteria along with the internal pressure criteria are satisfied for transfer of the EOS-37PTH DSC with HLZCs 4, 5 or 6 in an EOS-TC125/TC135 with intact FAs.

4.9.6.1.5 Evaluation for Damaged FAs in HLZC 6

HLZC 6 can accommodate a combination of intact FAs along with damaged FAs. It can be loaded with up to eight damaged FAs as noted in Section 4.9.6.1.2. This section presents the thermal evaluation of the EOS-37PTH DSC with Basket Type 4L for HLZC 6 with intact and damaged FAs during storage and transfer conditions.

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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.

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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.

The bounding HLZCs are used for dose rate analysis. For each zone within a DSC, higher heat loads result in stronger source terms and larger dose rates if the minimum cooling time is the same. When the EOS-89BTH DSC is used with the EOS-TC125, the minimum cooling time is three years in all zones. The EOS-89BTH DSC HLZC 1 and 2 are identical except for Zone 2. The Zone 2 heat load is larger for HLZC 1 compared to HLZC 2. Therefore, HLZC 1 bounds HLZC 2. Also, every zone of HLZC 2 is hotter than the corresponding zone of HLZC 3. Therefore, HLZC 2 bounds HLZC 3. Because HLZC 1 > HLZC 2 > HLZC 3, EOS-89BTH DSC HLZC 1 is bounding for the EOS-TC125 analysis.

The EOS-89BTH DSC HLZC 1 is not allowed in the EOS-TC108. As discussed in the previous paragraph, HLZC 2 has larger heat loads in each zone compared to HLZC 3. When HLZC 2 or 3 is used with the EOS-TC108, the minimum cooling time in zone 3 is 9.7 years and 9.0 years, respectively. While the EOS-89BTH DSC HLZC 2 zone 3 has a slightly longer minimum cooling time than HLZC 3 zone 3, the minimum cooling time difference (0.7 years) is small compared to the large difference in decay heat (0.1 kW/FA). Therefore, EOS-89BTH DSC HLZC 2 is bounding for EOS-TC108 analysis.

Likewise, for the EOS-TC108, the EOS-37PTH DSC HLZC 2 has a larger heat load in each zone compared to HLZC 3. Also, the minimum cooling time is lower in HLZC 2 Zone 3 (five to eight years) compared to HLZC 3 (nine years). Therefore, EOS-37PTH DSC HLZC 2 bounds HLZC 3 for EOS-TC108 analysis.

For PWR fuel in the EOS-TC125/135, the bounding HLZC cannot readily be determined, although the nine HLZCs may be reduced to three candidates based on head load considerations. HLZC 4 has the largest total heat load in the peripheral zone, HLZC 1 has a large heat load in an inner zone, and HLZC 5 has the largest heat load per fuel assembly. Therefore, each of these HLZCs is examined explicitly.

Based on MCNP calculations, HLZC 4 bounds HLZC 1. However, source terms are developed for both HLZC 4 and HLZC 5 because it cannot be determined which HLZC is bounding without performing detailed calculations. Source terms for HLZC 4 are derived for 1.0 kW/FA in Zone 1 and 1.625 kW/FA in Zones 2 and 3 for a total DSC heat load of 52.0 kW. This bounds the maximum DSC heat load of 50.0 kW. HLZC 5 source terms are developed for 0.7 kW/FA in Zone 1, 0.5 kW/FA in Zone 2, 2.4 kW/FA in Zone 3, and 0.85 kW/FA in Zone 4.

Note that up to eight damaged PWR fuel assemblies or up to four FFCs are authorized for HLZC 6 and HLZC 8. Source terms are also developed for a damaged/failed fuel HLZC that bounds both HLZC 6 and 8. These source terms are derived for 1.0 kW/FA in Zone 1, 1.5 kW/FA in Zone 2, 1.5 kW/FA for intact fuel in Zone 3, and 0.85 kW/FA for failed fuel in Zone 3. The ORIGEN-ARP methodology for developing damaged/failed fuel source terms is the same as used for developing intact fuel source terms. Reconfiguration of damaged/failed fuel source terms is addressed in Section 6.3.2.

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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 *in TS Table 3 [6-11]*, 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 *generated during irradiation* 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.

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 *considering* 450 g boron.

Control components also may result in helium gas generation 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, *a boron mass of 450 g is considered for CCs*.

450 g boron is input to ORIGEN-ARP with a burnup of 62 GWd/MTU to estimate the He generation for IFBA or CCs. The resulting helium generation is 8.34 moles. As 450 g boron contains approximately 8.34 moles of B-10, essentially all B-10 is converted to He due to the high burnup.

Irradiation gases are computed for the design basis PWR fuel without IFBA or CCs, and the moles of He due to IFBA and CCs are added. The moles of gas for each isotope of interest are reported in Table 6-40. The design basis PWR fuel assembly without IFBA or CCs contains 42.7 moles of gas. If both IFBA and CCs are included, the total moles of gas for PWR fuel is 42.7 moles + 8.34 moles + 8.34 moles = 59.4 moles.

Moles of gas are also computed for the design basis BWR fuel assembly using the same methodology, although BWR fuel does not contain IFBA or CCs. The design basis BWR fuel contains 17.4 moles of gas, as indicated in Table 6-40.

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 *FAs*. 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 *with CCs and IFBA (59.4* moles) may be scaled by 0.450/0.492 = 0.915. The bounding quantity of gas for the short PWR assemblies is then *59.4* moles*0.915 = 54.*3* 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.

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6.5 Supplemental Information

6.5.1 PWR Fuel Qualification

Chapter 6 presents the shielding analysis for design basis fuel. For the EOS-37PTH DSC, HLZC 4 results in bounding dose rates. HLZC 4 features 1.6 kW fuel in the peripheral region. The peripheral region is illustrated in TS Figure 3. HLZC 5 has a mixture of 2.4 kW and 0.85 kW fuel in the peripheral region, and HLZC 5 dose rates are similar to HLZC 4. HLZC 7 and HLZC 9 have similar peripheral heat loads compared to HLZC 4 and HLZC 5, respectively. HLZC 1, 2, 3, 6, or 8 do not result in bounding dose rates.

To provide additional assurance that TS dose rate limits will be met, a relationship between decay heat, burnup, enrichment, cooling time, and source terms is developed for 2.4 kW and 1.6 kW fuel and provided as fuel qualification tables (FQTs). The methodology to develop these FQTs is the same as used to develop the design basis source terms.

The purpose of the FQTs is solely to provide an additional dose rate constraint. Decay heat for each fuel assembly to be loaded is determined using NRC Regulatory Guide 3.54, ORIGEN-ARP, or other acceptable method.

The FQT developed based on 2.4 kW is a global constraint and is applied to every PWR fuel assembly to be loaded. This FQT is provided as TS Table 7B. The 1.6 kW FQT is applicable only to fuel located in zone 3 of HLZC 4 and HLZC 7 and is provided as TS Table 7C. TS Table 7C does not apply to HLZC 1, 2, 3, 5, 6, 8, or 9, or to the inner basket locations of HLZC 4 or 7.

A range of burnup, enrichment, and cooling time combinations are considered for the inner regions of HLZC 4 and 5, as documented in Table 6-8. The design basis source terms in the inner regions of HLZC 4 and 5 are optimized to maximize dose rates. However, dose rates, both transfer cask and storage, are dominated by thermally hot fuel in the peripheral region because inner locations are heavily self-shielded by peripheral fuel assemblies. For HLZC 4, the peripheral region (zone 3) contributes approximately 80% of the dose rate on the side of the EOS-TC125/135. For the EOS-HSM, the peripheral region (zone 3) contributes approximately 95% of the vent dose rate. Because the inner basket locations do not contribute appreciably to the total dose rate, an FQT constraint on the inner basket locations is not imposed.

The burnup in the FQTs is expressed in units of GWd/FA rather than GWd/MTU. The burnup in GWd/FA is the burnup in GWd/MTU multiplied by the MTU of the fuel assembly. The minimum cooling times are obtained from these tables using linear interpolation.

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As documented in Section 6.2.8, a small percentage (<0.5%) of fuel assemblies are low-enrichment outlier fuel (LEOF). LEOF is rare, occurring at a rate of approximately 1 per 200 fuel assemblies. To determine if a fuel assembly is LEOF, the enrichment is compared to the minimum value specified in TS Table 7A. LEOF would not affect storage dose rates, which are gamma dominated, but could have a small effect (generally < 5%) on transfer cask dose rates. Based on these considerations, up to 4 LEOFs are allowed in the peripheral region. A minimum of three non-LEOFs shall circumferentially separate LEOFs within the peripheral region. There are no limitations on the number and location of LEOF stored in the inner region.

Because LEOF, by definition, is below the minimum enrichments provided in the FQTs, minimum cooling times for LEOF are obtained by extrapolating the FQT cooling times using an appropriate method. Because minimum cooling times increase with lower enrichments, this extrapolation provides an additional cooling time penalty.

The overall method for application of these FQTs and qualification of LEOF is provided below.

1. *Determine the decay heat of all fuel to be loaded in an EOS-37PTH DSC using NRC Regulatory Guide 3.54, ORIGEN-ARP, or another acceptable method. Confirm the decay heat limit is met for each basket location.*

2. *Determine if LEOF is present in the fuel to be loaded by application of TS Table 7A.*

   a) *Up to 4 LEOF are allowed in the peripheral region. A minimum of three non-LEOFs shall circumferentially separate LEOFs within the peripheral region.*

   b) *There are no limitations on the number and location of LEOF stored in the inner region.*

3. *Verify all fuel to be loaded meets the minimum cooling time of TS Table 7B. Fuel that does not meet the cooling time limitations of this table cannot be loaded.*

4. *For fuel in zone 3 of HLZC 4 or HLZC 7, verify all fuel to be loaded meets the minimum cooling time of TS Table 7C. This table does not apply to HLZC 1, 2, 3, 5, 6, 8, or 9, or to the inner basket locations of HLZC 4 or HLZC 7.*

These FQTs provide an additional constraint to ensure compliance with the dose rate limitations in TS 5.1.2(c).

Examples

Examples to illustrate application of TS Table 7A, TS Table 7B, and TS Table 7C are provided below.

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Example 1 (no LEOF)

This example demonstrates how to determine if a fuel assembly is LEOF and how to determine compliance with TS Table 7B.

A fuel assembly has a burnup (BU) of 50 GWd/MTU, 0.45 MTU, enrichment (E) of 3.5%, and a cooling time (CT) of 4 years. Assume the decay heat has been computed and shown to be acceptable for the basket location of interest.

- *The minimum enrichment for 50 GWd/MTU, per TS Table 7A, is 50/16 = 3.125%, which is rounded down to 3.1%. As E = 3.5% > 3.1%, this fuel assembly is within the minimum enrichment bounds of TS Table 7A and is not LEOF.*
- *Burnup in GWd/FA is (50 GWd/MTU)(0.45 MTU) = 22.5 GWd/FA*
- *Linearly interpolate on enrichment (first) and burnup (second) to determine the minimum cooling time*
- *Linearly interpolating for E = 3.5% in the 22.14 GWd/FA row of TS Table 7B, CT = 2.95 years*
- *Linearly interpolating for E = 3.5% in the 24.6 GWd/FA row of TS Table 7B, CT = 3.29 years*
- *Linearly interpolating for BU = 22.5 GWd/FA between CT = 2.95 years and CT = 3.29 years, the minimum cooling time is CT = 3.00 years.*

Because 4 years > 3.00 years, the fuel assembly meets the TS Table 7B requirements.

Example 2 (with LEOF)

This example demonstrates how to determine if a fuel assembly is LEOF and how to determine compliance with TS Table 7B.

A fuel assembly has a burnup of 50 GWd/MTU, 0.45 MTU, enrichment of 2.9%, and a cooling time of 4 years. Assume the decay heat has been computed and shown to be acceptable for the basket location of interest.

- *The minimum enrichment for 50 GWd/MTU, per TS Table 7A, is 50/16 = 3.125%, which is rounded down to 3.1%. As E = 2.9% < 3.1%, this fuel assembly is LEOF. It is assumed to be the only LEOF to be loaded in this DSC.*
- *Burnup in GWd/FA is (50 GWd/MTU)(0.45 MTU) = 22.5 GWd/FA*
- *Linearly interpolate or extrapolate on enrichment (first) and burnup (second) to determine the minimum cooling time. Because the fuel is LEOF, extrapolation on the enrichment value beyond TS Table 7B is acceptable. Extrapolating to a lower enrichment value increases the minimum cooling time, which is a conservative penalty.*
- *Linearly interpolating for E = 2.9% in the 22.14 GWd/FA row of TS Table 7B, CT = 3.05 years*

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- *Linearly extrapolating for E = 2.9% in the 24.6 GWd/FA row of TS Table 7B for the two nearest enrichments, CT = 3.41 years. Other extrapolation methods could be employed, although the data in this row is following a linear trend.*
- *Linearly interpolating for BU = 22.5 GWd/FA between CT = 3.05 years and CT = 3.41 years, the minimum cooling time is CT = 3.10 years.*

Because 4 years > 3.10 years, the fuel assembly meets the TS Table 7B requirements. Because it is the only LEOF assembly in the DSC, it may be stored in the basket location of interest.

Example 3 (HLZC 1)

This example demonstrates how to determine if TS Table 7C is applicable.

A 2.0 kW fuel assembly will be loaded in HLZC 1 in zone 2 (inner basket location). Assume the decay heat has been computed and shown to be acceptable for this basket location.

TS Table 7C only applies to zone 3 of HLZC 4 and HLZC 7. Because fuel will be loaded in HLZC 1, TS Table 7C does not apply.

Example 4 (HLZC 4 inner)

This example demonstrates how to determine if TS Table 7C is applicable.

A 1.625 kW fuel assembly will be loaded in HLZC 4 in zone 2 (inner basket location). Assume the decay heat has been computed and shown to be acceptable for this basket location.

TS Table 7C only applies to zone 3 of HLZC 4 and HLZC 7. Because fuel will be loaded in zone 2 of HLZC 4, TS Table 7C does not apply.

Example 5 (HLZC 4 peripheral)

This example demonstrates how to determine compliance with TS Table 7C.

The fuel assembly in Example 1 has a burnup of 50 GWd/MTU, 0.45 MTU, enrichment of 3.5%, and a cooling time of 4 years. It is to be loaded in HLZC 4 zone 3 (peripheral region). Assume the decay heat has been computed and shown to be acceptable for the basket location of interest.

It is known this fuel assembly is not LEOF and meets the TS Table 7B per Example 1. However, TS Table 7C applies because fuel is loaded in the peripheral region of HLZC 4.

- *Burnup in GWd/FA is (50 GWd/MTU)(0.45 MTU) = 22.5 GWd/FA*
- *Linearly interpolate on enrichment (first) and burnup (second) to determine the minimum cooling time*

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- *Linearly interpolating for E = 3.5% for the 22.14 GWd/FA row of TS Table 7C, CT = 4.19 years*
- *Linearly interpolating for E = 3.5% for the 24.6 GWd/FA row of TS Table 7C, CT = 4.80 years*
- *Linearly interpolating for BU = 22.5 GWd/FA between CT = 4.19 years and CT = 4.80 years, the minimum cooling time is CT = 4.28 years.*

Because 4 years < 4.28 years, the fuel assembly cannot be loaded in HLZC 4 zone 3.

6.5.2 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 CoC 1042 Appendix A, NUHOMS® EOS System Generic Technical Specifications, Amendment *1*.

All Boxed and/or Shaded Changes are in response to Revised RAI 6-7

6-12 *NUREG/CR-6999, "Technical Basis for a Proposed Expansion of Regulatory Guide 3.54 - Decay Heat Generation in an Independent Spent Fuel Storage Installation," February 2010.*

6-13 *U.S. Energy Information Administration (EIA), Spent Nuclear Fuel GC-859 Database, Accessed January 20, 2019. URL: https:/www.eia.gov/nuclear/spent_fuel/*

6-14 *Unused.*

6-15 *Interagency Agreement DE-SA09-01 SR18976/TVA No. P-01 N8A-249655-001 between the Department of Energy (DOE) and the Tennessee Valley Authority (TVA) for the Off-Specification Fuel Project, April 5, 2001.*

All Boxed and/or Shaded Changes are in response to Revised RAI 6-18

Table 6-40
Irradiation Gases

Element	***PWR Moles of Gas***[(1)]	***BWR Moles of Gas***[(1)]
H	*6.17E-02*	*1.36E-02*
He	*1.58E+00*	*5.50E-01*
N	*2.17E-04*	*7.55E-05*
F	*3.14E-06*	*1.56E-06*
Ne	*1.43E-04*	*6.14E-05*
Cl	*1.02E-05*	*4.68E-06*
Ar	*4.33E-07*	*2.08E-07*
Br	*1.08E-01*	*4.40E-02*
Kr	*3.37E+00*	*1.39E+00*
I	*7.57E-01*	*2.95E-01*
Xe	*3.69E+01*	*1.51E+01*
Rn	*3.06E-13*	*9.48E-14*
Subtotal	*42.7*	*17.4*
He from CCs	*8.34*	*-*
He from IFBA	*8.34*	*-*
Total	*59.4*[(2)]	*17.4*

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) For fuel with a total unirradiated fuel length < 157 inches and maximum fuel loading < 0.450 MTU, this value is 54.3 moles.

All Boxed and/or Shaded Changes are in response to Revised RAI 8-2

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.

7-2 U.S. Nuclear Regulatory Commission, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," NUREG-1536, Revision 1, July 2010.

7-3 Scaglione, J.M., Mueller, D.E., Wagner, J.C., and Marshall, W.J., "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions," NUREG/CR 7109, U.S. Nuclear Regulatory Commission, April 2012.

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.

7-7 *TN Americas LLC, "Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," Revision 16, USNRC Docket Number 72-1004, July 2017.*

7-8 CoC 1042 Appendix A, NUHOMS® EOS System Generic Technical Specifications, Amendment 1.

All Boxed and/or Shaded Changes are in response to Revised RAI 7-1

Table 7-51

The detailed information associated with this table can be found in CoC 1042 Amendment 1 Technical Specifications Table 4

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, *Lifting Lug Plate*	Stainless steel ASME SA-240 Type 304, 316, 2205 or UNS S31803	72.48
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	Stainless steel ASTM A240 Type 304, 316, or 2205 or UNS S31803	72.48
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	ASTM B221 Alloy 6061	
R45 Angle Plates	ASTM A516 Gr 70	
Tie Rod/Flat Head Screws	ASTM A193 Gr B7	
Hex Nuts	ASTM A194 Gr 7	
Miscellaneous Washers	Carbon Steel	
Neutron absorber plates	MMC or BORAL® (89BTH only)	
Fuel Spacers	*Stainless steel ASTM A240 Type 304 or Aluminum ASTM B209, Type 6061 or 1100*	

All Boxed and/or Shaded Changes are in response to Revised RAI 9-2

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 High-Strength *Low-Alloy* Steel for Basket Structure

The basket structural material shall be a High-Strength Low-Alloy (HSLA) steel meeting one of the following requirements A, B, or C:

A. *ASTM A829 Gr 4130 or AMS 6345 SAE 4130, quenched and tempered at not less than 1050 °F, 103.6 ksi minimum yield, and 123.1 ksi minimum ultimate stress. This material is qualified as described in [10-31].*

B. *ASME Code edition 2010 with 2011 addenda, SA-517 Gr A, B, E, F, or P. This material is qualified by the material properties at elevated temperature in ASME Section II, Part D, which exceed the values of yield and ultimate strength in UFSAR Table 8-10.*

C. *Other HSLA steel, with the specified heat treatment, meeting these qualification and acceptance criteria:*

i. If quenched and tempered, the tempering temperature shall be at no less than 1000 °F,

ii. Qualified prior to first use by testing at least two lots and demonstrating that the fracture toughness value $K_{JIc} \geq 150$ ksi √in at -40 °F with 95% confidence based on the methodology in Reference [10-31] for HSLA steel.

iii. Qualified prior to first use by testing at least two lots and demonstrating that the 95% lower tolerance limit of yield and ultimate strengths ≥ the values in UFSAR Table 8-10 based on the methodology in Reference [10-31] for HSLA steel.

iv. Meet production acceptance criteria based on the 95% lower tolerance limit of yield strength and ultimate strength at room temperature as determined by qualification testing described in Section 10.1.7.iii.

The basket HSLA material shall also meet the following production acceptance criteria:

- *Weld repair shall not be permitted.*
- *Impact testing shall be performed at -40 °F*
  - *Charpy testing per ASTM A370, minimum absorbed energy 25 ft-lb average, 20 ft-lb lowest of three (modify these acceptance criteria for sub-size specimens per A370-17 Table 9), or*
  - *Dynamic tear testing per ASTM E604 with acceptance criterion of a minimum 80% shear fracture appearance.*
- *Test specimen location, orientation, and sampling rate per ASTM A6 or ASTM A20 for production acceptance testing.*

All Boxed and/or Shaded Changes are in response to Revised RAI 8-7

8 7 6 5 4 3 2 1

H G F E D C B A

Proprietary and Security Related Information
for Drawing MX01-5000-SAR, Rev. 0C
Withheld Pursuant to 10 CFR 2.390

8 7 6 5 4 3 2 1

Proprietary Information on Pages A.4-36, A.4-37, A.4-68, A.4-69, A.4-118, A.4-119, A.4-120 and A.4-121
Withheld Pursuant to 10 CFR 2.390

A.8-18 ASTM A588/A588M, "Standard Specification for High-Strength Low-Alloy Structural Steel, up to 50 ksi [345 MPa] Minimum Yield Point, with Atmospheric Corrosion Resistance."

All Boxed and/or Shaded Changes are in response to Revised RAI 8-6

Table A.8-2
Material Properties, ASTM A572 Grade 50 Steel

Temp (°F)	E [(2)] (10^3 ksi)	S_y [(3)] (ksi)	S_u [(4)] (ksi)	α_{AVG} [(5)] (10^{-6} °F^{-1})	ρ (lb/in^3) [(6)]
-20					0.280
70	*29.0*[(7)]	50.0[(1)]	65.0[(1)]		
100	*29.0*	48.5	65.0	6.3	
150					
200	*28.4*	46.0	63.7	6.5	
250					
300	*27.8*	44.0	65.0	6.7	
350					
400	*27.3*	42.5	65.0	6.9	
450					
500	*26.7*	41.5	65.0	7.1	
550					
600	*26.1*	41.0	62.4	7.2	
650					
700	*25.5*	40.0	53.3	7.4	

Notes

1. Reference [A.8-9].
2. Based on lowest rate of reduction provided in [A.8-8] Figure 7.5.
3. Based on lowest rate of reduction provided in [A.8-8] Figure 7.3.
4. Based on lowest rate of reduction provided in [A.8-8] Figure 7.4.
5. Based on lowest rate of reduction provided in [A.8-8] Figure 7.6.
6. ASME Section II Part D [A.8-2].
7. *Based on AISC, Table B4.1 [A.8-7].*

All Boxed and/or Shaded Changes are in response to Revised RAI 8-6

Table A.8-3
Material Properties, ASTM A992 Grade 50

Temp (°F)	E [2] (10^3 ksi)	Yield Strength[3] (ksi)	Tensile Strength[4] (ksi)	ρ [5] (lb/in^3)
-20				*0.280*
70	29.0[6]	50.0 [1]	65.0 [1]	
100	29.0	48.5	65.0	
150				
200	28.4	46.0	63.7	
250				
300	27.8	44.0	65.0	
350				
400	27.3	42.5	65.0	
450				
500	26.7	41.5	65.0	

Notes

1. Reference [A.8-14].
2. Based on lowest rate of reduction provided in [A.8-8] Figure 7.5.
3. Based on lowest rate of reduction provided in [A.8-8] Figure 7.3.
4. Based on lowest rate of reduction provided in [A.8-8] Figure 7.4.
5. ASME Section II Part D, Table PRD [A.8-2].
6. *Based on AISC, Table B4.1 [A.8-7]*

All Boxed and/or Shaded Changes are in response to Revised RAI 8-6

Table A.8-4
Material Properties, ASTM A588

Temp (°F)	*E [2] (10^3 ksi)*	*Yield Strength[3] (ksi)*	*Tensile Strength[4] (ksi)*	*Density[5] (lb/in^3)*
-20				0.280
70	29.0 [6]	50.0 [1]	70.0 [1]	
100	29.0	48.5	70.0	
150				
200	28.4	46.0	68.6	
250				
300	27.8	44.0	70.0	
350				
400	27.3	42.5	70.0	
450				
500	26.7	41.5	70.0	

Notes

1. *Reference* [A.8-18].
2. *Based on lowest rate of reduction provided in [A.8-8] Figure 7.5.*
3. *Based on lowest rate of reduction provided in [A.8-8] Figure 7.3.*
4. *Based on lowest rate of reduction provided in [A.8-8] Figure 7.4.*
5. *ASME Section II Part D, Table PRD [A.8-2].*
6. *Based on AISC, Table B4.1 [A.8-7]*

All Boxed and/or Shaded Changes are in response to Revised RAI 8-6