Enclosure 10 to E-42938

CoC 1004 Amendment 14, Revision 1
UFSAR Changed Pages
(Public Version)

the required actions to be performed following the event depend upon the severity of the event and the resultant cask and trailer/skid damage.

8.2.5.4 Recovery

For drop heights of less than fifteen inches the transfer cask will be loaded back onto the transfer skid/ trailer and moved to the HSM. The DSC will then be transferred to the HSM in the normal manner described previously. For drop heights greater than fifteen inches the transfer cask and contents will be returned to the plant's fuel/reactor building.[1] There the DSC will be inspected for damage, and the DSC opened and the fuel removed for inspection, as necessary. Removal of the transfer cask top cover plate may require cutting of the bolts in the event of a corner drop onto the top end. This operation will take place in the decontamination pit after recovery of the transfer cask. Removal of the DSC cover plates and shield plug assembly are described in Section 5.0.

Following recovery of the transfer cask and unloading of the DSC, the transfer cask will be inspected, repaired and tested as appropriate prior to reuse.

For drop heights approaching the design basis conditions, it may be necessary to develop a special sling/lifting apparatus to move the transfer cask from the drop site to the fuel pool.[1] This may require several weeks of planning to ensure all steps are correctly organized. During this time, additional blankets can be added to the transfer cask to minimize on-site exposure to site operations personnel. The transfer cask will be roped off to ensure the safety of the site personnel.

8.2.6 Lightning

8.2.6.1 Postulated Cause of Event

The likelihood of lightning striking the HSM and causing an off-normal condition is not considered to be a credible event. Lightning protection system requirements are site specific and depend upon 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 HSMs. The addition of simple lightning protection equipment, required by plant criteria, to HSM structures (i.e., grounded handrails, ladders, etc.) is considered a miscellaneous attachment and is acceptable as per Note 9 of the General Arrangement drawing (Dwg. No. NUH-03-6008).

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[1] *The recovery operations listed in this section assume the cask drop occurs during initial transfer and loading of the DSC into the HSM, when the spent fuel pool is still operational and available. If a drop of the Transfer Cask with a loaded DSC from a height greater than fifteen inches occurs during transfer to a transportation cask and an inspection determines that the DSC is damaged with possible fuel confinement boundary breaches and a spent fuel pool is not available onsite, the DSC shall be put into a safe condition. If required, the DSC could be transported offsite to a site licensed for either dry or wet unloading of the DSC.*

8.2.6.2 Analysis of Effects and Consequences

Should lightning strike in the vicinity of the HSM the normal storage operations of the HSM will not be affected. The current discharged by the lightning will follow the low impedance path offered by the surrounding structures. Therefore, the HSM will not be damaged by the heat or mechanical forces generated by current passing through the

9.6 Decommissioning Plan

Decommissioning of a NUHOMS® ISFSI can be performed in a manner consistent with that for decommissioning of the plant itself. It is anticipated that the DSCs will be transported intact to a Federal repository off-site when such a facility is operational. However, should the storage facility not accept the DSCs intact, the NUHOMS® system allows the DSCs to be brought back into the spent fuel pool, *if available,* and the fuel off-loaded to racks for subsequent loading into transport casks provided by the Department of Energy.

All components of the NUHOMS® system are manufactured of materials similar to those found at existing plants (e.g., reinforced concrete, stainless steel, lead). These components can therefore be decommissioned by the same methods in place to handle those materials within the plant. Any of the components that may be contaminated can be cleaned and/or disposed of using the decommissioning technology available at the time of decommissioning.

The NUHOMS® system is a dry containment system that effectively confines all contamination within the DSC. When the DSC is removed from the HSM, the free-standing HSM can be manually decontaminated for any trace activity, dismantled and removed from the site. It is possible that a thin layer of material comprising the inner wall of the HSM could become activated by the neutron flux from the fuel after an extended period of service. Estimates of the potential for activation are difficult due to the variability of rare earths which may be present in the local aggregate. The specific activity of the HSM inner wall surfaces may be measured at the time of decommissioning and compared with the existing guidelines to determine whether the values are below regulatory concern (BRC). Disposal procedures can then be developed which comply with existing guidelines at the time of decommissioning.

Removal of fuel assemblies from the DSC can be accomplished in the plant's spent fuel pool, *if available,* as described in Chapter 5. The DSC is also being qualified for off-site shipment in a compatible transportation cask licensed to 10CFR71. If such transport is made, the DSC may be disposed of as-is at the permanent geologic repository in a suitable overpack container. If the DSC is not compatible with the repository handling or packaging systems, fuel transfer to a suitable container can be performed in a large hot cell or off-site fuel pool.

The general license holder under 10CFR72.210 shall meet the requirements specified in 10CFR72.30.

Table M.2-3a
Initial Enrichment for Type A1 and A2 Basket and Minimum Soluble Boron Loading (NUHOMS®-32PT DSC)

Assembly Class and Type	Soluble Boron Loading (ppm)	0 PRAs (Type A1 and A2) 24 Poison Plate Configuration (0.015 g B-10/cm²) Type A1	(0.020 g B-10/cm²) Type A2
WE 17x17 fuel assembly (without CC)	2500	4.05	4.20
	2800	*NE(1)*	*4.50*
WE 17x17 fuel assembly (with CC)	2500	4.00	4.15
	2800	*NE(1)*	*4.45*
B&W 15x15 Mark B fuel assembly (without CC)	2500	4.00	4.10
B&W 15x15 Mark B fuel assembly (with CC)	2500	3.90	4.10
WE 15x15 fuel assembly (without CC)	2500	4.10	4.20
WE 15x15 fuel assembly (with CC)	2500	4.10	4.20
CE 14x14 fuel assembly (without CC)	1800	3.95	4.10
	2100	4.30	4.45
	2300	4.50	4.70
	2500	4.70	4.90
CE 14x14 fuel assembly (with CC)	1800	3.80	3.95
	2100	4.10	4.25
	2300	4.30	4.50
	2500	4.50	4.70
WE 14x14 fuel assembly (without CC)	1800	4.20	4.20
	2100	4.55	4.60
	2300	4.80	5.00
	2500	5.00	5.00
WE 14x14 fuel assembly (with CC)	1800	4.20	4.35
	2100	4.60	4.75
	2300	4.80	5.00
	2500	5.00	5.00
CE 15x15 fuel assembly	1800	3.50	3.60
	2100	3.75	3.90
	2300	3.95	4.10
	2500	4.10	4.30

Note:
(1) NE: Not Evaluated

M.6.1 Discussion and Results

Figure M.6-1 shows the cross section of the NUHOMS®-32PT DSC. The NUHOMS®-32PT DSC stainless steel basket consists of a welded plate or tube design. The welded plates or tubes form 32 compartments with sufficient space to accommodate aluminum or poison/aluminum inserts and a PWR fuel assembly. The fuel compartment structure is connected to perimeter transition rail assemblies as shown on the drawings in Section M.1.5. The poison/aluminum plates and aluminum plates are located inside the fuel compartments. The poison plates may be arranged in any of the following configurations: a 20 poison plate configuration (base configuration), as shown in Figure M.6-1; an alternate 16 poison plate configuration, as shown in Figure M.6-13; and another alternate 24 poison plate configuration, as shown in Figure M.6-14. Figure M.6-2 through Figure M.6-4 show the fuel compartments that must contain PRAs for loading configurations that require four, eight or sixteen PRAs. The 20 poison plate basket configuration shown in Figure M.6-1 is analyzed as a Type A/B/C/D basket while the 24 poison plate basket configurations shown in Figure M.6-14 is analyzed as an Alternate Type A/B/C/D basket at a boron-10 loading of 0.0070 g/cm^2. The 24 poison plate basket configuration is also analyzed as a Type A1 and Type A2 basket at boron-10 loadings of 0.0150 g/cm^2 and 0.0200 g/cm^2 respectively. The 16 poison plate basket configuration shown in Figure M.6-13 is also analyzed as an Alternate Type A basket.

The analysis presented herein is performed for a NUHOMS®-32PT DSC in the NUHOMS® OS197/197H Transfer Casks (TCs) during normal and accident loading conditions. The NUHOMS® OS197/197H TCs consists of an inner stainless steel shell, lead gamma shield, a stainless steel structural shell and a hydrogenous (liquid) neutron shield. This analysis is applicable to any licensed cask of similar construction. The NUHOMS®-32PT DSC/TC configuration is shown to be sub-critical under normal and accident conditions of loading, transfer and storage.

The criticality analysis determines the most reactive configuration for the basket and assembly location. Then criticality calculations evaluate a variety of fuel assembly types, initial enrichments and PRA configurations. Finally, the maximum allowed initial enrichment for each assembly type/PRA configuration is determined. The maximum allowed initial enrichment for each assembly type/PRA configuration is listed in Table M.6-1. The calculations determine k_{eff} with the CSAS25 *and CSAS5* control module*s* of SCALE-4.4 *and SCALE 6.0, respectively,* [6-1 *and 6-4*] for each assembly type/PRA configuration and initial enrichment, including all uncertainties to assure criticality safety under all credible conditions.

Note that the results of the 20-poison plate basket that specify the minimum allowable fuel assembly are not included in Table M.6-1. This implies that the 20-poison plate basket design which is exclusively modeled in most of the criticality evaluations in this chapter is not authorized to store PWR fuel assemblies. However, this basket design is employed in all the criticality sensitivity calculations and is referenced throughout the remainder of this chapter. The results of these sensitivity calculations are still applicable to the 32PT DSC.

The results of the evaluation presented include reconstituted fuel assemblies where fuel pins are replaced with up to 56 solid stainless steel rods or an unlimited number of lower enriched UO_2 rods of the same diameter as the fuel pins.

Additionally, the maximum allowable enrichment is determined for the WE 17x17 assembly class in the Type A2 basket at a soluble boron concentration of 2800 ppm.

Figure M.6-5 is a sketch of each KENO V.a unit showing all materials and dimensions for each Unit and an annotated cross section map showing the assembled geometry units in the radial direction of the most reactive configuration identified in this evaluation. The bounding k_{eff} is calculated with a Westinghouse 17x17 LOPAR/Standard assembly with an initial enrichment of 3.4 wt. % U-235, with no PRAs and 32 BPRAs.

Note that BPRAs are the most relevant CCs for criticality considerations and are utilized in the rest of this chapter to cover all CCs.

M.6.4 Criticality Calculations

This section describes the analysis performed for the criticality analysis. The analyses are performed with the CSAS25 module of the SCALE system. A series of calculations are performed to determine the relative reactivity of the various fuel assembly designs evaluated and to determine the most reactive configuration without PRAs and BPRAs. The most reactive fuel for a given enrichment, as demonstrated by the analyses, is the B&W 15x15 Mark B assembly. The most reactive credible configuration is an infinite array of flooded TCs with minimum fuel compartment inner diameter, minimum basket structure thickness and minimum assembly-to-assembly pitch.

As mentioned in Section M.6.1, the NUHOMS®-32PT DSC (including the 16 poison plate and 24 poison plate alternate configurations) is evaluated to determine the maximum initial enrichment authorized for each assembly class without PRAs, and with four, eight and 16 PRAs, as applicable.

M.6.4.1 Calculational Method

M.6.4.1.1 Computer Codes.

The CSAS25 control module of SCALE-4.4 [6-1] is used to calculate the effective multiplication factor (k_{eff}) of the fuel in the TC. The CSAS25 control module allows simplified data input to the functional modules BONAMI-S, NITAWL-S, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL-S applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0015 for all calculations.

The CSAS5 control module of SCALE 6.0 [6-4] is used in the determination of the maximum allowable enrichment for the WE 17x17 assembly class, 2800 ppm soluble boron and Type A2 basket. 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-S 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 k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0010 for all calculations.

M.6.4.1.2 Physical and Nuclear Data.

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

Table M.6-3 provides the pertinent data for criticality analysis for each fuel assembly evaluated in the NUHOMS®-32PT DSC.

The criticality analysis used the 44-group cross-section library built into the SCALE system. ORNL used ENDF/B-V data to develop this broad-group library specifically for criticality analysis of a wide variety of thermal systems.

M.6.4.1.3 Bases and Assumptions.

The analytical results reported in Section M.3.7 demonstrate that the TC containment boundary and canister basket structure do not experience any significant distortion under hypothetical accident conditions. Therefore, for both normal and hypothetical accident conditions the TC geometry is identical except for the neutron shield and skin. As discussed above, the neutron shield and skin are conservatively removed and the interstitial space modeled as water.

These criticality calculations were performed with CSAS25 of SCALE-4.4. For each case, the result includes (1) the KENO-calculated k_{KENO}, (2) the one sigma uncertainty σ_{KENO}, and (3) the final k_{eff}, which is equal to $k_{KENO} + 2\sigma_{KENO}$.

The criterion for subcriticality is that

$$k_{KENO} + 2\sigma_{KENO} \leq USL,$$

where USL is the upper subcritical limit established by an analysis of benchmark criticality experiments. From Section M.6.5, the minimum USL over the parameter range is 0.9411. From Table M.6-21 for the most reactive case,

$$k_{KENO} + 2\sigma_{KENO} = 0.9388 + 2\ (0.0011) = 0.9410 \leq 0.9411.$$

For the criticality evaluations that were performed with CSAS5 and SCALE 6.0, the maximum k_{eff} is obtained for the WE 17x17 loaded in the Type A2 basket at an enrichment level of 4.50 wt. % U-235, soluble boron concentration of 2800 ppm, no PRAs, and without CCs:

$$k_{eff} = k_{keno} + 2\sigma_{keno} = 0.9362 + 2*(0.0007) = 0.9376 \leq 0.9404$$

The criticality evaluation used the same cross section set, fuel materials and similar material/geometry options that were used in the 121 benchmark calculations as shown in Table M.6-22. The modeling techniques and the applicable parameters listed in Table M.6-24 for the actual criticality evaluations fall within or very close the range of those addressed by the benchmarks in Table M.6-22.

M.6.5.2 Results of the Benchmark Calculations

The results from the comparisons of physical parameters of each of the fuel assembly types to the applicable USL value are presented in Table M.6-24. The minimum value of the USL is determined to be 0.9411 based on comparisons to the limiting assembly parameters as shown in Table M.6-24.

M.6.5.3 Benchmarking of SCALE 6.0

For the criticality evaluations performed using CSAS5 of SCALE 6.0 [6-4], the 92 experimental problems used to perform the benchmarking along with pertinent parameters are listed in Table M.6-58.

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 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), 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 [6-7] 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 is, $|r|$, is nearly 1.0. As demonstrated in Section M.6.5.3.2, 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 [6-8] is used, it can be shown that the k_{eff} values are normally distributed, which is demonstrated in Section M.6.5.3.2.

M.6.5.3.1 Benchmark Experiments and Applicability

The criticality benchmark experiments are obtained from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [6-6]. A brief description of the experiments is provided as follows:

8 configurations (cases 1-8) from LEU-COMP-THERM-001:

These configurations are water moderated U(2.35 wt. % U-235)O_2 fuel rods in 2.032 cm square-pitched arrays. These experiments use open-top carbon steel tank configuration with acrylic support plate, polyethylene lattice plates and aluminum plates as support structure.

5 configurations (cases 1-5) from LEU-COMP-THERM-002:

These configurations are water-moderated U(4.31 wt. % U-235)O_2 fuel rods in 2.54-cm square-pitched arrays. These experiments use open-top carbon steel tank configuration with acrylic support plate, polyethylene lattice plates, and aluminum plates as support structure.

17 configurations (cases 1-17) from LEU-COMP-THERM-008:

These configurations are borated water moderated U(2.459 wt. % U-235)O_2 fuel rods in 1.636-cm square-pitched arrays. These experiments use aluminum tank, aluminum plates as support structure and Pyrex rods (cases 4-9), Vicor rods (case-10), or aluminum oxide rods (cases 11-15) as the absorbing rods.

5 configurations (case 5 and 16-19) from LEU-COMP-THERM-010:

These configurations are water-moderated U(4.31 wt. % U-235)O_2 fuel rods in square-pitched arrays reflected by depleted uranium (case 5) or steel (cases 16-19). These experiments use open-top carbon steel tank configuration with acrylic support plates, polyethylene lattice plates and aluminum plates as support structure.

24 configurations from (cases 3-17, 19-25 and 28-29) LEU-COMP-THERM-017:

These configurations are water-moderated U(2.35 wt. % U-235)O_2 fuel rods in square-pitched arrays reflected by lead (cases 3 and 23-25), depleted uranium (cases 4-9 and 28-29), or steel (cases 10-17 and 19-22). These experiments use open-top carbon steel tank configuration with acrylic support plates, polyethylene lattice plates and aluminum plates as support structure.

7 configurations (cases 1-7) from LEU-COMP-THERM-042:

These configurations are water-moderated, square-pitched arrays of U(2.35 wt. % U-235)O_2 fuel rods, separated by absorption plates made of steel (case 1), borated steel (case 2), BORAL® (case 3), Boroflex (case 4), cadmium (case 5), copper (case 6), or copper with 1% cadmium (case 7), and reflected by steel partitions. These experiments use open-top carbon steel tank configuration with acrylic support plates, polyethylene lattice plates and aluminum plates as support structure.

7 configurations (cases 1-7) from LEU-COMP-THERM-050:

These configurations are water-moderated, square-pitched arrays of U(4.738 wt. % U-235)O_2 fuel rods with a central Zircaloy tank that contains either water (cases 1-2) or boron solution (cases 3-7). Cases 8-18 are neglected from the LEU-COMP-THERM-050 experiments as they contain ^{149}Sm solution in the central Zircaloy tank. These experiments use stainless steel tank configuration with aluminum alloy (AG3M), and stainless steel as support structure.

19 configurations (cases 1-19) from LEU-COMP-THERM-051:

These configurations are borated-water-moderated U(2.459 wt. % U-235)O_2 fuel rods in 1.636-cm square-pitched arrays. These experiments use aluminum tank with steel and aluminum plates as support structure. Borated aluminum plates and stainless steel plates are used as absorber plates in these experiments.

All 92 critical experiments, comprised of borated and unborated systems are selected. The experiments selected are:

- *8 configurations (cases 1-8) from LEU-COMP-THERM-001*
- *5 configurations (cases 1-5) from LEU-COMP-THERM-002*
- *17 configurations (cases 1-17) from LEU-COMP-THERM-008*
- *5 configurations (case 5 and 16-19) from LEU-COMP-THERM-010*
- *24 configurations from (cases 3-17, 19-25 and 28-29) LEU-COMP-THERM-017*
- *7 configurations (cases 1-7) from LEU-COMP-THERM-042*
- *7 configurations (cases 1-7) from LEU-COMP-THERM-050*
- *19 configurations (cases 1-19) from LEU-COMP-THERM-051*

M.6.5.3.2 Statistical Analysis and Determination of USL

Determination of parameters responsible for variation in k_{eff} is of primary importance in the criticality analysis. Trending analysis is performed to determine the effectiveness of each parameter in explaining variations in calculated k_{eff} values. The correlation coefficient $|r|$ provides a measure of statistical correlation between each parameter and variations in the calculated k_{eff} values. A correlation value of $|r| = 0$ implies no correlation, and a value of $|r| = 1$ implies strong correlation.

The parameters considered to perform trending analysis against k_{eff} for the experiments are AEG, pitch (cm), assembly separation (cm), moderator-to-fuel ratio and soluble boron concentration (ppm). The $|r|$ values are provided for all the aforementioned parameters in Table M.6-59. The test for normality is performed using the USLSTATS code. As the data are normally distributed, and there is no close correlation with the parameters, the single-sided tolerance limit methodology described in NUREG/CR-6698 [6-8] is utilized to obtain the USL value.

The criticality code validation is performed according to NUREG/CR 7109 [6-5] to obtain the bias that results from the calculation of the benchmark experiments and the bias uncertainty that incorporates several other elements of uncertainty including a confidence interval. According to NUREG/CR-6698 [6-8], the USL may be obtained by computing a single-sided tolerance lower limit above which a defined fraction of the true population (95%) of k_{eff} is expected to lie with a prescribed confidence (95%), and within the area of applicability, a tolerance band, when a relationship between a calculated k_{eff} and independent variable can be determined, or nonparametric statistical treatment when the data do not follow a normal distribution. The independent parameters utilized for the tolerance band method are: AEG, fuel pitch, assembly separation, U-235 enrichment, soluble boron concentration, and moderator-to-fuel volume ratio. It is demonstrated that a close correlation does not exist between calculated k_{eff} and the independent parameters. The k_{eff} values are normally distributed and, therefore, the single-sided tolerance limit methodology described in NUREG/CR-6698 [6-8] is utilized to obtain the USL value. An administrative safety margin of 0.05 is applied.

M.6.5.3.3 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 M.6.5.3.2. The results in Table M.6-59 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 is 0.9404. The results are summarized in Table M.6-60.

M.6.6 Appendix

M.6.6.1 References

6-1 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

6-2 U.S. Nuclear Regulatory Commission, "Criticality Benchmark Guide for Light-Water-Reactor fuel in Transportation and Storage Packages," NUREG/CR-6361, Published March 1997, ORNL/TM-13211.

6-3 U.S. Nuclear Regulatory Commission, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR-5661, Published April 1997, ORNL/TM-11936.

6-4 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.

6-5 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.

6-6 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/.

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

6-8 Dean, J.C., Tayloe Jr., R.W., "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, January 2001.

Table M.6-1
Maximum Initial Enrichment For Each Configuration, wt. % U-235
(Part 2 of 2)

Assembly Class and Type	Soluble Boron Loading (ppm)	0 PRAs (Type A1 and Type A2 24 Poison Plate Configuration Type A1	Type A2
WE 17x17 fuel assembly (without CC)	2500	4.05	4.20
	2800	*NE(1)*	*4.50*
WE 17x17 fuel assembly (with CC)	2500	4.00	4.15
	2800	*NE(1)*	*4.45*
B&W 15x15 Mark B fuel assembly (without CC)	2500	4.00	4.10
B&W 15x15 Mark B fuel assembly (with CC)	2500	3.90	4.10
WE 15x15 fuel assembly (without CC)	2500	4.10	4.20
WE 15x15 fuel assembly (with CC)	2500	4.10	4.20
CE 14x14 fuel assembly (without CC)	1800	3.95	4.10
	2100	4.30	4.45
	2300	4.50	4.70
	2500	4.70	4.90
CE 14x14 fuel assembly (with CC)	1800	3.80	3.95
	2100	4.10	4.25
	2300	4.30	4.50
	2500	4.50	4.70
WE 14x14 fuel assembly (without CC)	1800	4.20	4.20
	2100	4.55	4.60
	2300	4.80	5.00
	2500	5.00	5.00
WE 14x14 fuel assembly (with CC)	1800	4.20	4.35
	2100	4.60	4.75
	2300	4.80	5.00
	2500	5.00	5.00
CE 15x15 fuel assembly	1800	3.50	3.60
	2100	3.75	3.90
	2300	3.95	4.10
	2500	4.10	4.30

Note:
(1) NE: Not Evaluated

Table M.6-57
WE 17x17 Class Assembly Final Results. Type A2 Basket
(24 poison plate configuration, variable soluble boron configuration)

Model Description	k_{keno}	**1σ**	k_{eff}
2500 ppm boron, w/o BPRAs, 4.2 wt% U-235			
40% IMD	0.8799	0.0009	0.8817
50% IMD	0.9083	0.0009	0.9101
60% IMD	0.9259	0.0009	0.9277
70% IMD	0.9333	0.0009	0.9351
80% IMD	**0.9361**	**0.0009**	**0.9379**
90% IMD	0.9332	0.0010	0.9352
100% IMD	0.9274	0.0008	0.9290
2500 ppm boron, w/ BPRAs, 4.15 wt% U-235			
40% IMD	0.8634	0.0009	0.8652
50% IMD	0.8935	0.0009	0.8953
60% IMD	0.9161	0.0009	0.9179
70% IMD	0.9270	0.0009	0.9288
80% IMD	0.9352	0.0011	0.9374
90% IMD	**0.9375**	**0.0008**	**0.9391**
100% IMD	0.9331	0.0009	0.9349
2800 ppm boron, w/o BPRAs, 4.50 wt.% U-235			
40% IMD	*0.8841*	*0.0007*	*0.8854*
50% IMD	*0.9115*	*0.0007*	*0.9129*
60% IMD	*0.9279*	*0.0006*	*0.9291*
70% IMD	*0.9359*	*0.0006*	*0.9372*
80% IMD	***0.9362***	***0.0007***	***0.9376***
90% IMD	*0.9322*	*0.0007*	*0.9337*
100% IMD	*0.9246*	*0.0005*	*0.9255*
2800 ppm boron, w/BPRAs, 4.45 wt.% U-235			
40% IMD	*0.8697*	*0.0007*	*0.8710*
50% IMD	*0.9001*	*0.0006*	*0.9013*
60% IMD	*0.9198*	*0.0006*	*0.9211*
70% IMD	*0.9302*	*0.0008*	*0.9317*
80% IMD	***0.9359***	***0.0008***	***0.9374***
90% IMD	*0.9350*	*0.0007*	*0.9363*
100% IMD	*0.9332*	*0.0008*	*0.9349*

Table M.6-58
Benchmark Experimental KENO V.a Simulation Results for SCALE 6.0
(4 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{keno}	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 M.6-58
Benchmark Experimental KENO V.a Simulation Results for SCALE 6.0
(4 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{keno}	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-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
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

Table M.6-58
Benchmark Experimental KENO V.a Simulation Results for SCALE 6.0
(4 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{keno}	σ
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
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

Table M.6-58
Benchmark Experimental KENO V.a Simulation Results for SCALE 6.0
(4 Pages)

Experiment Name	Enrichment (wt. % U-235)	Pitch (cm)	Assembly Separation (cm)	Soluble Boron (ppm)	Mod./Fuel Ratio	AEG	EALF (eV)	k_{keno}	σ
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 M.6-59
Correlation Coefficients |r| for Independent Parameters

Parameter	
U-235 Enrichment	*0.271*
Pitch (cm)	*0.170*
Moderator to Fuel Volume Ratio	*0.108*
AEG	*0.150*
Soluble Boron (ppm)	*0.506*
Assembly Separation (cm)	*0.612*

Table M.6-60
USL Evaluations

Equation Parameter	
K_{eff}	*0.9958*
S_p	*2.70E-3*
U	*2.0*
Δ_{sm}	*0.05*
USL	***0.9404***

T.4.6.10.2 Thermal Evaluation of 61BTH Type 2 DSC with HLZC #10

As shown in Figure T.2-11, HLZC #10 has a maximum heat load of 31.2 kW. Because HLZC #10 has a higher maximum allowable heat load than that for the 61BTH Type 1 DSC of 22 kW, it is only applicable to the 61BTH Type 2 DSC.

Since no other changes are considered to the 61BTH DSC except for the HLZC, the thermal evaluation of 61BTH DSC Type 2 with HLZC #10 is based on a sensitivity study of the normal hot storage condition with 100 °F ambient and the vertical transfer condition with 120 °F ambient. The maximum heat load of 31.2 kW for HLZC 10 is identical to the maximum heat load considered for 61BTH Type 2 DSC in Section T.4.4 and Section T.4.5. Therefore, based on the discussion in the main body of Section T.4.6.10, the DSC shell temperature profiles from the thermal evaluation results, presented for 61BTH DSC at 31.2 kW in Section T.4.4 and Section T.4.5, are used as boundary conditions to evaluate the thermal performance of the 61BTH Type 2 DSC with HLZC #10.

The following table compares the maximum fuel cladding and DSC component temperatures for the 61BTH Type 2 DSC with HLZC #10 to the design basis values presented in Table T.4-12 and Table T.4-14 for normal hot storage condition (DSC in HSM, 100 °F ambient). The design basis values presented for 61BTH Type 2 DSC are based on the bounding temperatures determined for HLZCs #5 through #8 with a maximum heat load of 31.2 kW for the normal hot storage condition (DSC in HSM, 100 °F ambient).

Maximum Component Temperatures for 61BTH Type 2 DSC for Normal Storage in 100 °F Ambient

HLZC	***Fuel Cladding (°F)***	***Fuel Compartment (°F)***	***Neutron Absorber (°F)***	***R45 & R90 Rails (°F)***	***Top Grid (°F)***	***DSC Shell (°F)***
61BTH Type 2 DSC (Design Basis) [Table T.4-12 and Table T.4-14]	*719*	*690*	*689*	*514*	*496*	*434*
61BTH Type 2 DSC HLZC #10	*711*	*687*	*686*	*515*	*497*	*434*
ΔT ($T_{HLZC\ \#10} - T_{Design\ Basis}$)	*-8*	*-3*	*-3*	*1*	*1*	*0*

As shown in the above table, the maximum temperatures of fuel cladding and fuel compartment for 61BTH Type 2 DSC with HLZC #10 are bounded by design basis values listed in Table T.4-12 and Table T.4-14. The 1 °F temperature increase observed for the basket rails and top grid is insignificant and does not affect the thermal or structural performance of the 61BTH Type 2 DSC. Based on this evaluation, the maximum fuel cladding temperature listed for the various storage conditions of a 61BTH Type 2 DSC in Table T.4-12, Table T.4-17 and Table T.4-21 remain bounding for HLZC #10.

A review of the time limits for transfer operations of a 61BTH Type 2 DSC with heat loads greater than 22 kW in Section T.4.5.4 shows that there are two separate time limits for 61BTH Type 2 DSC depending on the HLZC. HLZCs #5, #6 and #8 have transfer time limit of 26 hours, whereas HLZC #7 has a transfer time limit of 13 hours.

To determine the most appropriate time limit for transfer operation with HLZC #10, the 61BTH Type 2 DSC model with HLZC #7 (transfer time limit of 13 hours) was re-evaluated using HLZC #10. No other changes are considered in this evaluation. The following table presents a comparison of the maximum fuel cladding and DSC component temperatures determined previously for HLZC # 7 to those determined in this sensitivity study with HLZC # 10 for the vertical transfer condition (DSC in TC, 120 °F ambient) with a time limit of 15 hours (13 hours excluding the two hour allowance to initiate recovery actions).

Maximum Component Temperatures for 61BTH Type 2 DSC for Vertical Transfer, 120 °F Ambient, 15 hours

HLZC	*Fuel Cladding (°F)*	*Fuel Compartment (°F)*	*Neutron Absorber (°F)*	*R45 & R90 Rails (°F)*	*Top Grid (°F)*	*DSC Shell (°F)*
61BTH Type 2 DSC HLZC # 7	*730*	*701*	*701*	*522*	*493*	*408*
61BTH Type 2 DSC HLZC #10	*722*	*698*	*698*	*524*	*494*	*408*
ΔT ($T_{HLZC\ \#10} - T_{HLZC\ \#7}$)	*-8*	*-3*	*-3*	*2*	*1*	*0*

The results of this evaluation show that the maximum temperature of the fuel cladding and fuel compartment are lowered by 8 °F and 3 °F similar to the storage evaluation presented above. The temperature increase of 1 °F observed for the basket rails and 2 °F observed for the top grid is insignificant and does not affect the thermal or structural performance of the 61BTH Type 2 DSC. Furthermore, the maximum fuel cladding and DSC component temperatures for the 61BTH Type 2 DSC with HLZC # 10 are also below the bounding temperatures listed in Table T.4-12 and Table T.4-14 for the DSC in TC at 120 °F ambient (vertical transfer condition).

Based on this evaluation, the maximum fuel cladding temperatures listed for the various transfer conditions of a 61BTH Type 2 DSC in Table T.4-12, Table T.4-17, and Table T.4-21 remain bounding for HLZC #10. Therefore, the time limits for transfer operations determined for a 61BTH Type 2 DSC with HLZC 7 in Section T.4.5.4 are applicable to a 61BTH Type 2 DSC with HLZC #10.

In addition, the average helium temperature determined for HLZC #10 is 3 °F higher than that used in determining the maximum internal pressure in Table T.4-16. This small change in the average helium temperature does not affect the maximum internal pressure.

Based on this discussion, no further evaluations are required for a 61BTH Type 2 DSC with HLZC #10 and all design criteria described in Section T.4.1 are satisfied.

T.4.6.11 Thermal Analysis of 61BTH DSC with up to 61 Damaged Fuel Assemblies

Figure T.2-9 allows for the storage of up to 61 damaged fuel assemblies within the 61BTH DSC. The damaged fuel assemblies considered for storage in 61BTH DSC ensure that the fuel pellet cannot pass through the opening in the cladding during normal and off-normal conditions as noted in Section T.3.6.3. Further, the damaged fuel assemblies maintain their structural integrity during normal and off normal conditions of storage and on site transfer as noted in Section T.3.6.3.3. This ensures that there is no reconfiguration of the heat generating regions during normal/off-normal conditions. Additionally, the effective thermal conductivity of the fuel assemblies determined in Section T.4.8.1 depends on the physical configuration of the fuel assembly. Since the damaged fuel assemblies maintain the overall physical configuration similar to that of the intact fuel assemblies during normal and off-normal conditions, the effective thermal properties determined in Section T.4.8.1 remain valid for the damaged fuel assemblies during normal and off-normal conditions. Therefore, the thermal evaluation presented in Section T.4.6.6 through Section T.4.6.7 are acceptable for the normal and off-normal conditions, respectively, wherein up to 61 damaged fuel assemblies are stored in 61BTH DSC as shown in Figure T.2-9.

During the accident condition, the thermal evaluation presented in Section T.4.6.8 remains applicable to the damaged fuel assemblies if they maintain their physical configuration during the postulated drop accident. However, the cladding of high burnup damaged fuel assemblies can experience further damages during postulated drop accidents. In the event that they experience further damage, the worst possible scenario is that of the damaged fuel assemblies turning into rubble at the bottom of the DSC.

To evaluate the effect of the damaged fuel assemblies turning into rubble on the surrounding intact fuel assemblies during an accident condition (such as a drop accident), several sensitivity studies are performed based on the bounding transfer accident condition. According to Table T.4-21, the accident condition with loss of sun shade, neutron shield and air circulation with the 61BTH Type 2 DSC with 31.2 kW heat load (HLZC #7) represents the bounding transfer accident condition for the 61BTH DSCs. Therefore, this load case is selected for thermal evaluation since it has the maximum fuel cladding temperature and the maximum heat load.

A review of the bounding transfer accident thermal evaluation performed for 61BTH Type 2 DSC shows that the highest temperatures within the basket are around the middle of the DSC. Based on this information, several sensitivity studies are performed wherein intact fuel assemblies are considered in the middle of the basket and damaged fuel assemblies surrounding the intact ones are modeled as rubble.

To bound any possible scenario and to determine the bounding fuel cladding temperature of intact fuel assemblies loaded along with damaged fuel assemblies during accident conditions, five additional thermal evaluations of 61BTH DSC with HLZC #7 are performed. These evaluations consider 16, 40, 52, 60, and 61 damaged fuel assemblies turning into rubble at the bottom of the DSC. The following figure shows the locations identified as "A," "B," and "C1" through "C4." Locations identified as "A" and "B" are the same as those noted in Figure T.2-9. Locations "C" in Figure T.2-9 are subdivided into four groups in the following figure as "C1" through "C4" to easily identify the locations loaded with damaged fuel assemblies in these evaluations. The five additional evaluations are:

- *45 intact / 16 damaged fuel assemblies*

  For this evaluation, locations identified as "A" and "B" are considered to be loaded with damaged fuel assemblies whereas locations "C1" through "C4" are loaded with intact fuel assemblies.

- *21 intact / 40 damaged fuel assemblies*

  For this evaluation, locations identified as "A," "B," and "C1" are considered to be loaded with damaged fuel assemblies whereas locations "C2" through "C4" are loaded with intact fuel assemblies.

- *9 intact / 52 damaged fuel assemblies*

  For this evaluation, locations identified as "A," "B," "C1," and "C2" are considered to be loaded with damaged fuel assemblies whereas locations "C3" and "C4" are loaded with intact fuel assemblies.

- *1 intact / 60 damaged fuel assemblies*

  For this evaluation, locations identified as "A," "B," "C1," "C2," and "C3" are considered to be loaded with damaged fuel assemblies whereas location "C4" is loaded with intact fuel assemblies.

- *61 damaged fuel assemblies*

  For this evaluation, locations identified as "A," "B," "C1," "C2," "C3," and "C4" are considered to be loaded with damaged fuel assemblies.

			C1	C1	C1			
	A	B	C1	C1	C1	B	A	
	B	B	C2	C2	C2	B	B	
C1	C1	C2	C3	C3	C3	C2	C1	C1
C1	C1	C2	C3	C4	C3	C2	C1	C1
C1	C1	C2	C3	C3	C3	C2	C1	C1
	B	B	C2	C2	C2	B	B	
	A	B	C1	C1	C1	B	A	
			C1	C1	C1			

Notes:

1. *See Figure T.2-9 for description of locations "A" and "B."*
2. *Locations identified as "C1," "C2," "C3," and "C4" are the same as Locations "C" in Figure T.2-9. Locations "C" in Figure T.2-9 are subdivided into four groups in this figure as "C1" through "C4" to easily identify the locations loaded with damaged fuel assemblies.*

Location of Damaged Fuel inside 61BTH DSC

To analyze the thermal performance of the 61BTH DSC with intact and damaged fuel assemblies, the thermal model for the 61BTH Type 2 DSC from Section T.4.6.2 is modified as noted below. The DSC shell temperature profile is based on the accident condition with loss of sun shade, neutron shield and air circulation from Section T.4.6.8.2. The heat generation rates for the damaged fuel considered as rubble during accident conditions and intact FAs in the 61BTH DSC are the same as discussed in Section T.4.6.9.2. The bounding effective thermal conductivity of the damaged fuel considered as rubble during accident conditions is calculated as 0.12 BTU/hr-in-°F using the same methodology as discussed in Appendix K, Section K.4.8.1.

The following table shows the maximum temperatures of intact fuel cladding, fuel compartment, and DSC components with 0, 16, 40, 52, 60 and 61 damaged fuel assemblies within a 61BTH Type 2 DSC at 31.2 kW for the bounding transfer accident condition. As shown in the table, the maximum fuel cladding and DSC components temperatures increase with the increase of the total number of damaged fuel assemblies. However, for all evaluations with intact fuel assemblies, the maximum fuel cladding temperatures are well below the limit of 1058°F. For the case with 61 damaged fuel assemblies, since all damaged fuel assemblies are considered as rubble, there are no thermal limits associated with this scenario. Therefore, there is no impact on loading damaged fuel along with intact fuel within the 61BTH DSC.

Maximum Component Temperatures for 61BTH DSC with up to 61 Damaged Fuel Assemblies for the Bounding Transfer Accident Condition

Transfer Condition	*Accident Condition (117°F Ambient, Loss of Neutron Shield and Air Circulation)*					
DSC Type	*61BTH*					
Heat Load, kW	*31.2*					
Fuel Assembly Type	*61 Intact [Table T.4-21 and Table T.4-23]*	*45 Intact / 16 Damaged (1)*	*21 Intact / 40 Damaged (1)*	*9 Intact / 52 Damaged (1)*	*1 Intact / 60 Damaged (1)*	*61 Damaged (1)*
Component	T_{max} *(°F)*					
Intact Fuel Cladding	*830*	*813*	*824*	*875*	*955*	*N/A*
Fuel Compartment	*805*	*787*	*801*	*858*	*943*	*962*
Neutron Absorber	*804*	*787*	*800*	*855*	*942*	*960*
R45 & R90 Rails	*650*	*651*	*683*	*703*	*715*	*717*

Note:
(1) The damaged fuel assemblies are modeled as rubble during accident conditions with the bounding effective thermal conductivity of 0.12 BTU/hr-in-°F using the same methodology as discussed in Appendix K, Section K.4.8.1.

Furthermore, due to cracks in the fuel cladding, the damaged fuel assemblies do not contain any fission/fill gases and, therefore, the maximum internal pressure under accident

conditions is bounded by the evaluation presented for intact fuel assemblies (assume a 100% rupture of the fuel pins) in Section T.4.6.8.5.

T.6.1 Discussion and Results

Figure T.6-1 and Figure T.6-2 show the radial cross section of the NUHOMS®-61BTH Type 1 and Type 2 DSCs. The generic cask consists of an inner stainless steel shell, and lead gamma shield, a stainless steel structural shell and a hydrogenous neutron shield. This analysis is applicable to any licensed cask of similar construction. The NUHOMS®-61BTH DSC/Cask configuration is shown to be subcritical under normal, off-normal and accident conditions.

The criticality calculations assume the General Electric (GE) 10x10-fuel assembly because it is the most reactive fuel assembly allowed by the authorized contents. The calculations determine k_{eff} with the CSAS25 *and CSAS5* control module*s* of SCALE-4.4 *and SCALE 6.0, respectively,* [6.1 *and 6.7*] for various configurations and initial enrichments, including all uncertainties to assure criticality safety under all credible conditions.

The results of the evaluation demonstrate that the maximum k_{eff}–including statistical uncertainty– is 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.

to maximize the reactivity of the damaged fuel assembly and also to qualify fuel assemblies with damaged grids and missing rods to be loaded in the damaged fuel assembly locations.

The most reactive damaged fuel assembly configuration is based on a 10x10 lattice with optimum pitch and 95 fueled rods.

T.6.4.1 Calculational Method

T.6.4.1.1 Computer Codes

The CSAS25 control module of SCALE-4.4 [6.1] was used to calculate the effective multiplication factor (k_{eff}) of the fuel in the cask. The CSAS25 control module allows simplified data input to the functional modules BONAMI-S, NITAWL-S, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL-S applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0016 for all calculations.

Validation and verification of the SCALE 4.4 computer system were performed. Criticality benchmarking calculations were performed.

The CSAS5 control module of SCALE6 [6.7] is used in the determination of the additional damaged fuel configrations performed with the NUHOMS®-61BTH DSC. 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-S 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 k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0010 for all calculations.

T.6.4.1.2 Physical and Nuclear Data

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

Table T.6-3 lists the pertinent data for criticality analysis with the GE12 10x10 fuel assembly in the NUHOMS®-61BTH DSC as loaded in a generic cask described in Section T.6.1.

The criticality analysis used the 44-group cross-section library built into the SCALE system. ORNL used ENDF/B-V data to develop this broad-group library specifically for criticality analysis of a wide variety of thermal systems.

T.6.4.1.3 Bases and Assumptions

The analytical results reported in Section T.3 demonstrate that the cask containment boundary and canister basket structure do not experience any significant distortion under hypothetical accident conditions. The fuel assembly drop analyses documented in Section T.3-5 also demonstrate that the fuel rods do not experience any deformation significant to cause a change in the fuel geometry. Therefore, for both normal and hypothetical accident conditions the cask geometry is identical except for the neutron shield and skin. As discussed above, the neutron shield and skin are conservatively modeled as water.

The cask was modeled with KENO V.a using the permissible geometry options. These options allow a model to be constructed with regular geometric shapes and define the material

The Upper Subcritical Limit (USL) is calculated in accordance with NUREG/CR-6361 [6.4]. USL Method 1 (USL-1) applies a statistical calculation of the bias and its uncertainty plus an administrative margin (0.05) to the linear fit of results of the experimental benchmark data. The basis for the administrative margin is from Reference [6.5]. Results from the USL evaluation are presented in Table T.6-16.

The criticality evaluation used the same cross section set, fuel materials and similar material/geometry options that were used in the 125 benchmark calculations as shown in Table T.6-15. The modeling techniques and the applicable parameters listed in Table T.6-17 for the actual criticality evaluations fall within the range of those addressed by the benchmarks in Table T.6-15.

T.6.5.2 Results of the Benchmark Calculations

The results from the comparisons of physical parameters of each of the fuel assembly types to the applicable USL value are presented in Table T.6-17. The minimum value of the USL was determined to be 0.9410 based on comparisons to the limiting assembly parameters as shown in Table T.6-17.

T.6.5.3 Benchmarking of SCALE 6.0

The system studied here is a GE12 10x10 fuel assembly with 92 rods. It is an unborated system. The upper subcritical limit (USL) determination is made based on the criticality benchmark experiments. From the list of 92 criticality benchmark experiments presented in Appendix M.6 (including borated and unborated systems), 51 unborated experiments are selected. The experiments selected are:

- *8 configurations (cases 1-8) from LEU-COMP-THERM-001*
- *5 configurations (cases 1-5) from LEU-COMP-THERM-002*
- *5 configurations (case 5 and 16-19) from LEU-COMP-THERM-010*
- *24 configurations from (cases 3-17, 19-25 and 28-29) LEU-COMP-THERM-017*
- *7 configurations (cases 1-7) from LEU-COMP-THERM-042*
- *2 configurations (cases 1-2) from LEU-COMP-THERM-050*

The k_{eff} values of the 51 experiments are examined to determine correlation against different independent parameters. The independent parameters utilized are: energy of average lethargy of fission (EALF), fuel pitch, assembly separation, U-235 enrichment, and moderator-to-fuel volume ratio. The results in Table T.6-24 indicate that there is little correlation between k-eff and independent parameter values.

According to the methodology described in NUREG/CR-6698 [6.8], the USL is obtained by computing a single-sided tolerance lower limit above which a defined fraction of the true population (95 %) of k_{eff} is expected to lie, with a prescribed confidence (95 %) and within the area of applicability, a tolerance band, when a relationship between a calculated k_{eff} and independent variable can be determined, or nonparametric statistical treatment when the data do not follow a normal distribution. The k_{eff} values are normally distributed and, therefore, the single-sided tolerance limit methodology described in NUREG/CR-6698 [6.8] is utilized to obtain the USL value. An administrative safety margin of 0.05 is applied. The USL determined is 0.9418. The results are summarized in Table T.6-25.

T.6.6 Appendix

T.6.6.1 References

6.1 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

6.2 Not used.

6.3 Not used.

6.4 U.S. Nuclear Regulatory Commission, "Criticality Benchmark Guide for Light-Water-Reactor fuel in Transportation and Storage Packages," NUREG/CR-6361, Published March 1997, ORNL/TM-13211.

6.5 U.S. Nuclear Regulatory Commission, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR-5661, Published April 1997, ORNL/TM-11936.

6.6 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE 5: 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-725," June 2004. 72.48

6.7 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.

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

T.6.6.4 Additional Damaged Fuel Configurations

This section demonstrates that the NUHOMS® -61BTH DSC is authorized to store additional damaged fuel in basket Types D through Type F provided that:

1. *57 damaged fuel assemblies at a fixed enrichment level of 3.3 wt. % U-235 and four intact fuel assemblies at a maximum allowable enrichment of 5.0 wt. % U-235, placed in four corner locations in the DSC as seen in Figure T.6-9.*

2. *57 damaged fuel assemblies are at a fixed enrichment level of 3.3 wt. % U-235 and four damaged fuel assemblies at a maximum allowable enrichment of 4.2 wt. % U-235 are placed in four corner locations of the DSC as shown in Figure T.6-10.*

As demonstrated, most reactive damaged fuel configuration is obtained with the GE12, 10x10 fuel at optimum pitch with 92 rods, Table T.6-3.

The CSAS5 control module of SCALE 6.0 [6.7] is used to perform the evaluations to authorize the additional damaged fuel loading configurations. Since the USL and values of k_{eff} calculated for the starting case have been calculated using SCALE 4.4 representative cases from Table T.6-12 are reevaluated using SCALE 6.0 [6.7]. The most reactive cases from each set of code evaluations are compared, which demonstrated a reactivity difference that is statistically insignificant. As a result, a code correction factor is not assumed.

Using the starting model, the maximum allowable enrichment for the four corner location damaged fuel assemblies with 57 damaged fuel assemblies (at a fixed enrichment of 3.3 wt. % U-235) is computed. Similarly, the starting model is used to determine the maximum allowable enrichment for four corner location intact fuel assemblies with 57 damaged fuel assemblies (at a fixed enrichment of 3.3 wt. % U-235).

For the loading configuration consisting of 57 damaged fuel assemblies (at a fixed enrichment level of 3.3 wt. % U-235) and four intact fuel assemblies in corner locations, the maximum allowable enrichment for the four corner intact fuel assemblies is 5.0 wt. % U-235. These results are shown in Table T.6-22. The highest value of k_{eff} recorded at 5.0 wt. % U-235 is equal to 0.9384. This is bounded by the USL of 0.9418.

For the loading configuration with 57 damaged fuel assemblies (at a fixed enrichment level of 3.3 wt. % U-235) and four damaged fuel assemblies in corner locations the maximum allowable enrichment for the four corner damaged fuel assemblies is 4.2 wt. % U-235. These results have are presented in Table T.6-23. The highest value of k_{eff} recorded for 4.2 wt. % U-235 is equal to 0.9392. This is bounded by the system USL of 0.9418. The authorized loading combinations are provided in Table T.6-1.

Table T.6-17
USL Determination for Criticality Analysis

Parameter	Value from Limiting GE 10x10 Analysis	Bounding USL
Pin Pitch (cm)	1.2954	0.9416
Water to Fuel Volume Ratio	1.411	0.9421
Average Energy Group Causing Fission (AEG)	< 34[(1)]	0.9433
Assembly Separation (cm)	*1.6383* [(2)]	*0.9410*
Enrichment (wt. % U-235)	3.7 (minimum)	0.9438

1. Examination of the results shows that the value is between 32 and *35* and hence, a conservative value that produces the minimum USL was chosen.
2. Separation Distance= *2*0.075 + 0.105" + 0.09" + 0.3" = 1.6383 cm, calculated using fuel compartment thickness (0.090" for 2X2 compartments, 0.105" for 3X3 compartments), compartment wrapper thickness (0.075") and poison plate thickness (0.3").*

Table T.6-22
Criticality Results with 57 Damaged Fuel Assemblies at 3.3 wt. % U-235, 4 Intact Fuel Assemblies at 5.0 wt. % U-235

Model Description	k_{keno}	1σ	k_{eff}
EMD=1%	***0.9369***	***0.0008***	***0.9384***
EMD=10%	*0.9357*	*0.0008*	*0.9373*
EMD=20%	*0.9368*	*0.0008*	*0.9384*
EMD=30%	*0.9345*	*0.0008*	*0.9361*
EMD=40%	*0.9365*	*0.0009*	*0.9383*
EMD=50%	*0.9353*	*0.0009*	*0.9370*
EMD=60%	*0.9357*	*0.0009*	*0.9375*
EMD=70%	*0.9365*	*0.0008*	*0.9381*
EMD=80%	*0.9361*	*0.0008*	*0.9378*
EMD=90%	*0.9337*	*0.0008*	*0.9352*
EMD=100%	*0.9357*	*0.0008*	*0.9373*

Table T.6-23
Criticality Results with 57 Damaged Fuel Assemblies at 3.3 wt. % U-235, 4 Damaged Fuel Assemblies at 4.2 wt. % U-235

Case Description	k_{keno}	1σ	k_{eff}
EMD=1%	*0.9350*	*0.0008*	*0.9365*
EMD=10%	*0.9373*	*0.0008*	*0.9388*
EMD=20%	*0.9356*	*0.0011*	*0.9378*
EMD=30%	*0.9359*	*0.0008*	*0.9375*
EMD=40%	*0.9359*	*0.0008*	*0.9374*
EMD=50%	*0.9349*	*0.0008*	*0.9364*
EMD=60%	*0.9358*	*0.0008*	*0.9375*
EMD=70%	*0.9362*	*0.0008*	*0.9378*
EMD=80%	***0.9374***	***0.0009***	***0.9392***
EMD=90%	*0.9361*	*0.0008*	*0.9377*
EMD=100%	*0.9369*	*0.0008*	*0.9386*

Table T.6-24
Correlation Coefficients |r| for Independent Parameters

Parameter	Correlation Coefficient (|r|)
U-235 Enrichment	0.218
Pitch	0.053
Moderator to Fuel Volume Ratio	0.141
EALF	0.340
Assembly Separation	0.487

Table T.6-25
USL Determination

Equation Parameter	Value
k_{eff}	0.9964
S_p	2.23E-3
U	2.065
Δ_{sm}	0.05
USL	**0.9418**

T.8.1 Procedures for Loading the Cask

T.8.1.1 Preparation of the Transfer Cask and DSC

Notes:

- If using the OS200/OS200 FC TC for transfer of the NUHOMS®-61BTH DSC, verify that it has been fitted with an internal aluminum sleeve (refer to Drawing NUH-08-8004-SAR provided in Appendix U.1, Section U.1.5). This step, if required, can be performed at any time prior to placing the DSC in the TC.

1. Prior to placement in dry storage, the candidate intact and damaged fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification 2.1.

2. Prior to being placed in service, the transfer cask is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification 5.2.4.d.

3. Place the transfer cask in the vertical position in the cask decon area using the cask handling crane and the transfer cask lifting yoke.

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

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

5a. If using OS200/OS200FC TC to load, verify that a cask spacer of appropriate height (Refer to Drawing NUH-08-8005-SAR provided in Appendix U.1, Section U.1.5) is placed at the bottom of the TC.

6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed. Record the DSC serial number which is located on the grapple ring. Verify the correct DSC type, basket type, and poison material types against the DSC serial number. Verify that the DSC is appropriate for the specific fuel loading campaign per Technical Specification 2.1.

   CAUTION: If loading fuel assemblies through the basket hold down ring (HDR) or top grid assembly (TGA), verify that the lifting grapple will be able to release fuel assemblies while inside the HDR/TGA.

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

8. If damaged fuel assemblies are to be included in a specific loading campaign, place the required number of bottom end caps provided (up to a maximum of *61*) into the bottom

T.8.1.2 DSC Fuel Loading

1. Lift the TC/DSC and position it over the cask loading area of the spent fuel pool in accordance with the plant's 10CFR50 cask handling procedures.

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

3. Place the cask in the designated location of the fuel pool.

4. Disengage the lifting yoke from the cask lifting trunnions and move the yoke. Spray the lifting yoke 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 damaged and/or intact fuel assemblies 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 failed, damaged, and/or intact fuel assemblies meet the burnup, enrichment and cooling time parameters of Technical Specification 2.1.
   - 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 with the fuel movement schedule.
   - If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations before fuel load.
   - If failed fuel is to be loaded in the DSC, place the empty failed fuel cans (refer to drawing NUH61BTH-72-1105) in the appropriate locations in the 61BTH DSC. (Note: If the failed fuel is to be loaded into the failed fuel can prior to loading into the DSC, skip this step.)

6. Prior to insertion of a spent fuel assembly into the DSC, the identity of the assembly is to be verified by two individuals using an underwater video camera or other means. Read and record the fuel assembly identification number from the fuel assembly and check this identification number against the DSC loading plan which indicates which fuel assemblies are acceptable for dry storage.

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Steps 6 and 7 for each SFA loaded into the DSC. A maximum of *61* damaged fuel or 4 failed fuel assemblies may be loaded into the appropriate 2x2 compartments of the 61BTH DSC basket per Technical Specification 2.1. If loading failed fuel, ensure that the failed fuel can lids are installed. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly in the DSC. If

U.1 General Discussion

This Appendix to the NUHOMS® Updated Final Safety Analysis Report (UFSAR) addresses the *important* to *safety* aspects of adding the NUHOMS®-32PTH1 system to the Standardized NUHOMS® *System* described in the UFSAR.

The NUHOMS®-32PTH1 *System* is a modular canister based spent fuel storage and transfer system, similar to the Standardized NUHOMS®-24PTH *System* described in Appendix P of the UFSAR. *It is designed to accommodate up to 32 intact, up to 16 damaged, or up to 16 failed fuel canisters,* with characteristics as described in Chapter U.2.

The NUHOMS® 32PTH1 System consists of the following new or modified components:

- A 32PTH1 DSC, with three alternate configurations, described in detail in Section U.1.2, provides confinement, an inert environment, structural support, and criticality control for the 32 PWR fuel assemblies,
- A modified HSM-H module, described in Section U.1.2, is provided for environmental protection, shielding and heat rejection during storage, and
- OS200 or OS200FC TC for onsite transfer of the 32PTH1 DSCs.

The NUHOMS®-32PTH1 *dry shielded canister* (DSC) is a dual purpose (Storage/-Transportation) canister, with three alternate configurations depending on the canister length: a short length (185.75 *in.)* DSC designated as Type 32PTH1-S DSC, a medium length (193.0 *in.)* DSC designated as Type 32PTH1-M DSC, and a long (198.5 *in.)* DSC designated as Type 32PTH1-L DSC. The 32PTH1 DSC is designed for a maximum heat load of 40.8 kW.

The 32PTH1 DSC basket design is provided with two alternate options: a Type 1 basket with solid aluminum rails and a Type 2 basket with steel transition rails including aluminum inserts. The solid aluminum rail configuration of the Type 1 basket facilitates heat transfer and is the preferred option for canisters with high decay heat loads. For criticality control, the NUHOMS®-32PTH1 basket is provided with three alternate neutron absorber plate materials: a *borated aluminum* alloy, or *boron carbide/aluminum metal matrix composite* (MMC) or Boral®. In addition, for each neutron absorber material, the NUHOMS®-32PTH1 DSC basket is analyzed for five alternate basket configurations, depending on the boron content provided, to accommodate the various fuel enrichment levels (designated as Type A for the lowest B-10 loading to Type E for the highest B-10 loading).

The 32PTH1 DSC is stored in a modified version of the *horizontal storage module* (HSM-H) described in Appendix P of the UFSAR. The diameter of the HSM-H access door is increased to accommodate the larger diameter of the 32PTH1 DSC. In addition, spacers are provided to accommodate the three different lengths of the 32PTH1 DSCs. All of the key design features of the HSM-H, which provide enhanced shielding and heat rejection capabilities remain unchanged from those described in Appendix P of the UFSAR.

U.1.1 Introduction

The NUHOMS®-32PTH1 *System* is designed to store up to 32 (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR fuel assemblies. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years. Each of the 32PTH1 DSC types is designed to store up to 32 Control Components (CCs) which include *burnable poison rod assemblies* (BPRAs), *thimble plug assemblies* (TPAs), *control rod assemblies* (CRAs), *rod cluster control assemblies* (RCCAs), *axial power shaping rod assemblies* (APSRAs), *orifice rod assemblies* (ORAs), *vibration suppression inserts* (VSIs), and *neutron source assemblies* (NSAs). The design characteristics, including physical and radiological parameters of the payload, are described in Chapter U.2.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or 32 lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods or 32 Zr rods or Zr pellets or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC as intact fuel assemblies with a slightly longer cooling time than that required for a standard assembly. The maximum number of reconstituted fuel assemblies with irradiated stainless steel rods per DSC is four.

Provisions have been made for storage of up to 16 damaged fuel assemblies in lieu of an equal number of intact assemblies in the cells located at the center of the 32PTH1 basket. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The DSC basket cells *that* store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

Provisions have also been made for storage of up to 16 FFCs in cells located at the corners of the interior 4x4 compartment cells of the 32PTH1 basket or in a checkerboard pattern, as described in Chapter U.2.

The NUHOMS®-32PTH1 *System* consists of the following new or modified components:

- A 32PTH1 DSC, with three alternate configurations, described in detail in Section U.1.2, provides confinement, an inert environment, structural support, and criticality control for the 32 PWR fuel assemblies,
- A modified HSM-H module, described in Section U.1.2, is provided for environmental protection, shielding and heat rejection during storage, and
- OS200 or OS200FC TC for onsite transfer of the 32PTH1 DSCs.

The NUHOMS®-32PTH1 *System* requires the use of non-safety related auxiliary transfer equipment similar to those described in Section 1.3.2.2 (for OS200 TC) and Appendix P (for OS200FC TC) of the UFSAR. There is no change to any of the design features of the auxiliary transfer equipment except for the dimension changes necessary to accommodate the larger OS200 TC relative to the OS197 TC.

The OS200/OS200FC TC is provided with an aluminum internal sleeve to accommodate onsite transfer of the smaller diameter NUHOMS®-61BT, 32PT, 24PTH and 61BTH DSCs described in Appendix K, M, P and T respectively. This inner sleeve is provided with slots to accommodate the existing rails inside the TC and is fitted with rails inside the sleeve on which the smaller diameter DSCs can slide during the horizontal loading or unloading from the TC to the HSM-H. Drawing NUH-08-8004-SAR provided in Section U.1.5 shows the fabrication details of the internal sleeve. A docking ring adapter as shown in Appendix U.3, Figure U.3.7-39 is used to dock the OS200/OS200FC TC with HSM Models 80, 102, 152 and 202 when loading these smaller diameter DSCs. To accept the varying length of the DSCs, stainless steel or aluminum spacers are provided to limit axial movement of the payload (See Drawing NUH-08-8005-SAR provided in Section U.1.5).

U.1.2.2 Operational Features

U.1.2.2.1 General Features

The NUHOMS®-32PTH1 System is designed to safely store up to 32 intact, up to 16 damaged, or up to 16 failed fuel canisters. The NUHOMS®-32PTH1 DSC is designed to maintain the fuel cladding temperature below allowable limits during normal storage, short-term accident conditions, short-term off-normal conditions and fuel loading/transfer operations.

The criticality control features of the NUHOMS®-32PTH1 DSC are designed to maintain the neutron multiplication factor k-effective less than the upper subcritical limit equal to 0.95 minus benchmarking bias and modeling bias under all conditions.

U.1.2.2.2 Sequence of Operation

The sequence of operations to be performed in loading fuel into the NUHOMS®-32PTH1 DSCs is presented in Chapter U.8.

U.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

U.1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry, soluble boron in the spent fuel pool and by utilizing fixed neutron absorber material in the fuel basket. During storage, with the DSC cavity dry and sealed from the environment, criticality control measures within the installation are not necessary because of the low reactivity of the fuel in the dry NUHOMS®-32PTH1 DSC and the assurance that no water can enter the DSC cavity during storage.

U.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS®-32PTH1 system.

U.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS®-32PTH1 DSC system is a totally passive system so that consideration of operation shutdown modes is unnecessary.

U.1.2.2.3.4 Instrumentation

No change to Section 5.1.3.4.

U.1.2.2.3.5 Maintenance Techniques

No change to Section 5.1.3.5.

U.1.2.3 Cask Contents

The NUHOMS®-32PTH1 System is designed to store up to 32 intact, up to 16 damaged, or up to 16 failed fuel canisters. The fuel that may be stored in the NUHOMS®-32PTH1 DSC is presented in Chapter U.2.

Chapter U.3 provides the structural analysis. Chapter U.4 includes the thermal analysis. Chapter U.5 provides the shielding analysis. Chapter U.6 covers the criticality safety of the NUHOMS®-32PTH1 DSC system and its contents, listing material densities, moderator ratios, and geometric configurations.

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
3A	REVISED FOR AMENDMENT 14	11/09/15
2	REVISED PER FCN 721004-1063	01/08/14
1	REVISED PER FCN 721004-831	01/26/12
0	INITIAL ISSUE PER FCN 721004-466 (SHOWN BY CLOUDS), 721004-664 (SHOWN BY CLOUDS), FCN 721004-756 (FCN 721004-756 INCORPORATES AMD 10)	01/28/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1
TRANSPORTABLE CANISTER
PWR FUEL BASKET ASSEMBLY

DRAWING NO. NUH32PTH1-1003-SAR | SCALE NONE | SHEET 1 OF 5

NUH32PTH1-1003-SAR 2 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1003-SAR	2 OF 5	3A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH32PTH1-1003-SAR 4 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1003-SAR	4 OF 5	3A

8 7 6 5 4 3 2 1

DRAWING NO.	SHEET
NUH32PTH1-1003-SAR	5 OF 5

H G F E D C B A

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SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO	SHEET	REVISION
NUH32PTH1-1003-SAR	5 OF 5	3A

8 7 6 5 4 3 2 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0B	REVISED DURING AMENDMENT 14 RAI'S TO INCLUDE WELD DETAILS	11/09/15
0A	FIRST ISSUE FOR AMENDMENT 14	04/15/15

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INTERPRET WELD SYMBOLS

AREVA

SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1F
TRANSPORTABLE CANISTER FOR PWR FUEL
FAILED FUEL CAN

DRAWING NO.	SCALE	SHEET
NUH32PTH1-1006-SAR	NONE	1 OF 2

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH32PTH1-1006-SAR | SHEET 2 OF 2 | REVISION 0B

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

0B	REVISED DURING AMENDMENT 14 RAs TO INCLUDE 16 FFCs	11/09/15
0A	FIRST ISSUE FOR AMENDMENT 14	04/15/15
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS

AREVA

SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1F
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET ASSEMBLY MODIFICATIONS FOR FAILED FUEL CAN

DRAWING NO. NUH32PTH1-1007-SAR | SCALE NONE | SHEET 1 OF 2

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1007-SAR	2 OF 2	0B

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding, and criticality evaluations for these rods are bounded by those of the regular fuel rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or 32 lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods, or 32 Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC as intact fuel assemblies with a slightly longer cooling time than that required for a standard assembly. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel replacement rods or 32 with UO_2 replacement rods.

The NUHOMS®-32PTH1 DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the center cells of the DSC as shown in Figure U.2-1 through Figure U.2-3. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks, or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means and retrievability is assured following normal and off-normal conditions. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

The NUHOMS®-32PTH1F DSC, an alternate version of the NUHOMS®-32PTH1 DSC, is designed to accommodate failed fuel in up to a maximum of four failed fuel cans (FCCs) placed in the corner cells of the interior 4x4 compartment cells of the basket, as shown in Figure U.2-5 or in a checkerboard loading as shown in Figure U.2-3. Failed fuel is defined as fuel rods that have been removed from a fuel assembly and placed in a secondary container, such as a rod storage basket. Failed fuel may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means. Individual fuel rods that are not failed can be stored directly in the FFC without a secondary container such as an RSB. The maximum number of fuel rods that may be stored in a failed fuel can is 100, with a total uranium loading limited to 250 kg initial uranium. The total weight of the failed fuel can plus all its contents shall be less than 1715 lbs, 1625 lbs, and 1665 lbs, for the 32PTH1-L, 32PTH1-M, and 32PTH1-S DSCs, respectively.

A 32PTH1 DSC containing less than 32 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

A summary of the alternate poison loadings considered and the corresponding credit taken in the criticality analysis for each poison material as a function of basket types is presented below:

Poison Type	32PTH1 Basket Type[(1)]	Poison Loading (B10 mg/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum Alloy/MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	20	
	1D or 2D	32	
	1E or 2E	50	
Boral®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	25	
	1D or 2D	NA	
	1E or 2E	NA	

(1) Type 1A = Basket Type 1 with solid aluminum transition rails and Type A poison plate configuration;
Type 2A = Basket Type 2 with steel transition rails including aluminum inserts and Type A poison plate configuration.

The maximum assembly average initial enrichment as a function of soluble boron concentration and basket neutron poison requirements for intact fuel are summarized in Table U.2-4. The maximum assembly average initial enrichment as a function of soluble boron concentration and basket neutron poison requirements for damaged fuel and damaged fuel with up to four failed fuel cans are summarized in Table U.2-5. The maximum assembly average initial enrichment as a function of soluble boron concentration and basket neutron poison requirements for up to sixteen failed fuel cans are summarized in Technical Specification Table 1-1dd1.

The detailed information associated with this figure can be found in CoC 1004 Amendment 14 Technical Specifications Figure 1-28.

Figure U.2-3