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SOUTHERN COMPANY
Energy to Serve Your World℠

June 13, 2005

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NL-05-0984

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant
Technical Specifications Revision
Spent Fuel Cask Loading Requirements
Response to Draft Request for Additional Information

By letter dated May 17, 2005 (NL-05-0740), Southern Nuclear Operating Company (SNC) submitted a request to revise the Joseph M. Farley Nuclear Plant Unit 1 and Unit 2 Technical Specifications. Specifically, the request proposed to revise technical specification requirements in the 10 CFR 50 license to establish cask storage area boron concentration limits, and to restrict the minimum burnup of spent fuel assemblies associated with spent fuel cask loading operations which are scheduled to begin July 1, 2005.

In response to this request, SNC received a draft Request for Additional Information (RAI) from the NRC dated May 27, 2005. The enclosure to this letter contains the draft RAI and SNC's response. The information contained in the enclosure has previously been discussed with the NRC by telephone conference on June 1, 2005. The 10 CFR 50.92 evaluation and the justification for the categorical exclusion from performing an environmental assessment that were included in the May 17, 2005, submittal continue to remain valid.

(Affirmation and signature are on the following page.)

A001

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

Sworn to and subscribed before me this 13 *day of* June *, 2005.*

Notary Public

My commission expires: NOTARY PUBLIC STATE OF ALABAMA AT LARGE
MY COMMISSION EXPIRES: June 10, 2008
BONDED THRU NOTARY PUBLIC UNDERWRITERS

LMS/TMM/sdl

Enclosure: Response to Draft Request for Additional Information

cc: Southern Nuclear Operating Company
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Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley
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Dr. D. E. Williamson, State Health Officer

Enclosure

Joseph M. Farley Nuclear Plant
Technical Specifications Revision
Spent Fuel Cask Loading Requirements
Response to Draft Request for Additional Information

Provided in this enclosure is Southern Nuclear Operating Company's (SNC's) response to the Nuclear Regulatory Commission's (NRC's) draft Request for Additional Information (RAI) dated May 27, 2005, concerning a proposed request to revise technical specification requirements in the 10 CFR 50 license to establish cask storage area boron concentration limits, and to restrict the minimum burnup of spent fuel assemblies to support cask loading operations.

NRC Draft RAI No. 1

Upon reviewing the licensee's amendment request, the staff is unclear regarding the current licensing basis for the Farley spent fuel pools. In order for the staff to make a determination regarding the acceptability of the amendment request, the staff must determine the appropriate regulations that apply. Since the amendment request references NRC's recently issued Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," the staff believes that the applicable regulation is 10 CFR 50.68, "Criticality accident requirements." The staff requests that SNC identify the current licensing basis for the Farley spent fuel pools.

SNC Response to Draft RAI No. 1

Farley Nuclear Plant requested an exemption from the requirements of §70.24 for criticality monitors by letter dated May 31, 1996. In response, the NRC granted the exemption by letter dated July 31, 1996. Section II of the Exemption states, "the basis for the exemption is that inadvertent or accidental criticality will be precluded through compliance with the Farley Technical Specifications, the geometric spacing of fuel assemblies in the new fuel storage facility and spent fuel storage pool, and administrative controls imposed on fuel handling procedures." The last paragraph of Section II of the Exemption goes on to state, "The circumstances for granting an exemption to 10 CFR 70.24 are met because criticality is precluded with the present design configuration, Technical Specification requirements, administrative controls, and the fuel handling equipment and procedures."

By letter dated January 23, 1998, the NRC issued Amendments 133 and 125 to the Farley Unit 1 and Unit 2 facility operating licenses, respectively, to eliminate credit for Boraflex as a neutron poison and establish an alternate method for maintaining the recommended subcriticality margin in the fuel storage pool without relying on the Boraflex material. Issuance of Amendments 133 and 125 resulted in a change to the Farley Technical Specifications to incorporate Limiting Condition for Operation 3.7.13 which requires a minimum spent fuel pool soluble boron concentration ≥ 2000 ppm when fuel assemblies are stored in the fuel storage pool. In addition, Amendments 133 and 125 also incorporated a burnup versus enrichment curve, alternate storage patterns, and design features to maintain the fuel safely subcritical with the appropriate margins.

SNC continues to maintain Technical Specifications, geometric spacing of fuel assemblies, and administrative controls imposed on fuel handling procedures to preclude inadvertent or accidental criticality in accordance with the exemption granted to the requirements of § 70.24. It should be noted that § 50.68 was not available as an option to installation of criticality monitors required by § 70.24 until after Amendments 133 and 125 were issued.

Based on the above, the need for an exemption to § 70.24 continued following issuance of Amendments 133 and 125 to the Farley Unit 1 and 2 operating licenses, respectively. The circumstances for granting the exemption to 10 CFR 70.24 (i.e., NRC acceptance criteria) continued to be met because criticality was precluded by the design configuration, Technical Specification requirements, administrative controls, and the fuel handling equipment and procedures as stated in the NRC safety evaluation report (SER) associated with original safety analysis report granting the exemption to § 70.24. In addition, neither the NRC SER associated with Amendments 133 and 125, nor subsequent docketed correspondence from the NRC has been provided notifying SNC that the Farley exemption to the requirements of § 70.24 has been withdrawn or is otherwise invalid. Accordingly, the exemption to the requirements of § 70.24 continues to remain in effect subsequent to the changes incorporated by Amendments 133 and 125 to the Farley Unit 1 and 2 facility operating licenses, respectively, on January 23, 1998.

Based on the above, the licensing bases for criticality control in the Farley Unit 1 and Unit 2 spent fuel pools is described in Amendments 133 and 125, respectively. Additionally, Farley continues to retain its exemption to the requirements of § 70.24 granted by NRC letter dated July 31, 1996. The Technical Specifications, geometric spacing of fuel assemblies, and administrative controls imposed on fuel handling procedures is currently limited to the spent fuel storage racks and the new fuel storage area. NRC approval of the proposed license amendment request will provide the TS requirements, geometric spacing of fuel assemblies, and administrative controls for spent fuel loaded into the spent fuel cask, consistent with the NRC acceptance criteria cited in its SER supporting Farley's exemption granted to § 70.24.

Although RIS 2005-05 went into detail regarding licensee compliance with § 50.68, SNC has not elected to comply with § 50.68 but instead, continues to rely on the exemption to § 70.24. As stated above, the proposed license amendment request proposes to incorporate the required Technical Specifications, geometric spacing, and administrative controls necessary for the exemption to apply to spent fuel loaded into spent fuel casks in the cask storage area.

NRC Draft RAI No. 2

In Section 1.4, "Assumptions," of Enclosure 6, "Westinghouse Calculation Note CN-CRIT-207," SNC states that "Depleted fuel assemblies are conservatively modeled with a UO_2 density equal to 10.412 g/cm^3 (95.0% of theoretical density)." However, no basis is given either in this section or elsewhere in the report to support the stated conclusion that this represents a conservative assumption. Since the basis for the acceptability of the proposed TS changes is focused on the burnup credit analyses, the assumptions input into those analyses must be shown to include appropriately conservative values. Therefore, the staff requests that SNC provide additional information capable of technically supporting the assertion that this assumption is conservative over the range of acceptable assembly burnups provided in proposed Technical Specification Figure 3.7.18-1.

SNC Response to Draft RAI No. 2

The stack density of 95.0 % of theoretical density models fuel pellets with a pellet density equal to 96.1 % of theoretical density and a void fraction of 1.1 %. Sensitivity calculations were performed as a function of stack density. The range of these calculations is 95 to 97.5 % of stack theoretical density. These calculations indicate that the difference in burnup storage limits is extremely small (approximately 20 to 40 MWD/MTU out of 30,000 MWD/MTU) and is in the order of KENO's calculational uncertainty. Over the range between 20,000 to 50,000 MWD/MTU, the isotopics produced based upon 95 % of stack theoretical density produced slightly higher burnup limits. Therefore, for the irradiated fuel assemblies in the Farley Unit 1 and Unit 2 spent fuel pools, the assumption of a stack density of 95% leads to conservative storage burnup limits in the Multi-Purpose Canister (MPC).

NRC Draft RAI No. 3

Additionally, in Section 1.4 SNC states that "The multipurpose canister will be kept sufficiently far away (greater than 1 foot) from other fissile materials such that neutron interactions are precluded." Since the criticality analysis submitted did not account for closer spacing than what was described in the assumption, the staff requests that SNC provide additional information describing operationally how this spacing will be assured prior to and during any cask loading, unloading, or handling operations in the spent fuel pool.

SNC Response to Draft RAI No. 3

The Farley spent fuel cask storage area used for loading spent fuel storage casks is physically separated from spent fuel stored in the spent fuel storage racks. The Farley spent fuel pool and cask storage areas are separated by the transfer canal used during refueling operations to transfer fuel to and from the reactor and spent fuel pool. Furthermore, spent fuel is not stored in the transfer canal or in the cask storage area. Accordingly, spent fuel is only moved into the vicinity of the MPC for the purpose of placing the spent fuel in the MPC.

In addition to the above physical separation of the spent fuel storage racks and cask storage area, the design of the FNP auxiliary building requires that the spent fuel cask crane access the cask loading area and cask wash area via a hatch in the auxiliary building roof. Therefore, cask movements inside the auxiliary building are limited by the physical opening of the auxiliary building roof hatch to the cask storage and wash areas.

Accordingly, physical limitations eliminate the possibility of movement of the spent fuel cask into the vicinity of fissionable material contained in the spent fuel storage racks.

NRC Draft RAI No. 4

In Section 1.4, SNC stated that it modeled the poison plate material as Metamic with a minimum Boron-10 areal density of 0.031 g/cm^2. Additionally, SNC stated that this areal density will bound the Boral minimum Boron-10 areal density of 0.0372 g/cm^2 described in the Holtec HI-STORM 100 FSAR. In Section 2.1, "Design Input from SNC," SNC states that the Metamic specifications are taken from the Holtec HI-STORM 100 FSAR. In Table 6.3.4, "Composition of Major Components of the HI-STORM 100," of the Holtec HI-STORM 100 License Amendment Request 2, the Boron-10 density for both Boral and Metamic is listed as 0.0279 g/cm^2. Furthermore, Section 6.4.11, "Fixed Neutron Absorber Material," states that "Metamic is a single layer material with the same overall thickness and the same ^{10}B loading (in g/cm^2) [as Boral] for each basket." Based on the 75 percent credit assumption for fixed neutron absorbers, the nominal areal density for both Boral and Metamic is 0.0372 g/cm^2. Therefore, the staff requests that SNC provide additional information describing how the nominal areal density of Boron-10 in the fixed neutron absorbers was determined and how any uncertainties in the absorbers' properties, such as areal density, thickness, and length were accounted for in the criticality analyses.

SNC Response to Draft RAI No. 4

Certificate of Compliance (CoC) 1014, Appendix B, provides design features that must be met for each cask. Section 3.2.5.2 of CoC 1014, Appendix B, requires that the ^{10}B loading in the neutron absorber be $\geq$ 0.0372 g/cm^2 for Boral and $\geq$ 0.0310 g/cm^2 for Metamic. Section 6.1 of the HI-STORM 100 FSAR, Revision 3, states, "Consistent with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber and 90% of the manufacturer's minimum Boron-10 content for the Metamic neutron absorber." The ^{10}B density of 0.0279 g/cm^2 shown in Table 6.3.4 of Holtec HI-STORM 100 License Amendment Request 2 represents 75% credit for the minimum ^{10}B loading for Boral and 90% credit for the minimum ^{10}B loading for Metamic required by CoC 1014, Appendix B, Section 3.2.5.2. However, for analyses supporting licensing actions under Part 50, there is no corresponding requirement to reduce the areal density. Accordingly, use of 100% credit for the minimum ^{10}B loading required by CoC 1014 for Metamic of 0.0310 g/cm^2 is used. This value bounds the minimum required ^{10}B loading for Boral of 0.0372 g/cm^2. The values used by SNC are Technical Specification minimal acceptable values for areal density. Therefore, it is not necessary to account for manufacturing uncertainties in the analysis.

NRC Draft RAI No. 5

In Section 1.4, SNC stated that it modeled the unborated moderator (water) with a density equal to 1.0 g/cc. The staff agrees that the assumption of full density moderator is conservative if the moderator temperature coefficient (MTC) is negative under nominal loading conditions in the MPC. NRC regulations (10 CFR 50.68) and guidance documents require that the criticality analyses be performed under optimum moderation conditions. Since under some design configurations, the MTC can be positive, the staff requests the licensee describe what analyses it performed to demonstrate that the MTC in a fully loaded MPC was negative and that the full density moderator assumption was conservative.

SNC Response to Draft RAI No. 5

Westinghouse performed sensitivity calculations at 68, 100, and 185°F. Note that the nominal temperature range for the water in the spent fuel pool is 50 to 180°F. The 185°F temperature was chosen to conservatively bound the upper end of the temperature range. The KENO calculation performed at 68°F was employed for the reference condition. The KENO calculation at 100°F demonstrated that the reactivity decreased by 0.77 % delta k_{eff} from the reference condition. The KENO calculation at 185°F demonstrated that the reactivity decreased by 1.02 % delta k_{eff} from the reference condition. Therefore, the reactivity of the dry storage cask is maximized by selecting the lowest water temperature.

NRC Draft RAI No. 6

In Table 2-1, "MPC-32 Cell Dimensions," SNC lists the important design parameters used in its model for the criticality analysis. Only two of the parameters listed contain manufacturing tolerances that were subsequently included in the criticality analysis to determine the maximum effective multiplication factor (k_{eff}). Since the appropriate statistical accounting of manufacturing tolerances and uncertainties is essential for ensuring that the maximum k_{eff} is determined and NRC regulations are satisfied, the staff requests that SNC provide additional information to justify not including tolerances and uncertainties in the criticality analyses for the other design parameters in Table 2-1.

SNC Response to Draft RAI No. 6

The figures for the MPC in the FSAR do not contain the tolerance values in question. As stated in the June 1, 2005, teleconference between SNC and the NRC, the analysis was performed for an upper limit k_{eff} value, including all biases and uncertainties, less than or equal to 0.97 in full density unborated water. Therefore, it is concluded that the added margin to the k_{eff} limit of unity is more than sufficient to cover the statistical combination of the uncertainties in question.

NRC Draft RAI No. 7

In Section 3.1.1, "KENO Model for Multi-Purpose Canister," SNC states that a 2-foot water reflector was modeled on all sides of the MPC-32 and vacuum boundary conditions were applied in all directions. The 2-foot water reflector does not appear to be consistent with the 1-foot assumption listed in Section 1.4. Therefore, the staff request that SNC describe the real-world significance of each of these assumptions and how they will be assured during MPC loading, unloading, and handling operations in the SFP. Likewise, the licensee is requested to describe the basis for the vacuum boundary condition used in the KENO model.

SNC Response to Draft RAI No. 7

The KENO model for the MPC employed a two foot reflector in both the radial and axial directions. The neutron flux which arrives at the external surface of a two foot water reflector is nearly zero. Therefore, the use of void boundary conditions at the surface of the two foot water reflector is appropriate. Note that if reflective or periodic boundary conditions were employed the calculated results would be nearly identical to the ones generated with void boundary conditions.

Note that the two foot water reflector employed for the KENO model and the 1-foot assumption listed in Section 1.4 are not related. The 1-foot assumption listed in Section 1.4 is provided to preclude neutronic interaction with fuel assemblies in the spent fuel pool and the two foot water reflector is employed to maximize the calculated reactivity results.

NRC Draft RAI No. 8

In Section 3.2, "Design Basis Fuel Assembly," SNC stated that "Based on scoping calculations for the U-235 loading and storage configuration considered here, the most reactive fresh fuel assembly design is the Westinghouse 17 x 17 Standard fuel assembly." (Emphasis Added). Additionally, SNC added that for the postulated accidents analyzed, the Westinghouse Optimized Fuel Assembly is conservatively modeled as the misloaded fuel assembly. Since the determination of the most limiting assembly design is essential for ensuring that the regulatory and safety limits for k_{eff} are met, the staff requests that SNC provide the following information:

a. A technical description and the results of the methodology used to compare the design basis fuel assemblies under normal and accident conditions and to determine the bounding fuel assembly design.

b. A technical basis for why a comparison of fresh fuel assemblies was used to determine the bounding fuel assembly design when storage of the assemblies will be restricted based on a burnup limit.

SNC Response to Draft RAI No. 8

a. For the storage of fuel assemblies in the MPC, the design basis fuel assembly is the Westinghouse standard assembly in the MPC environment. Under accident conditions, the most limiting accident is the loading of a fresh 5.0 w/o U-235 Westinghouse OFA assembly into a cask that contains Westinghouse standard assemblies. This is also discussed in Sections 3.2, 3.5.1, and 3.6.3 of the criticality analysis report (CN-CRIT-207, provided as Enclosure 6 of the SNC license amendment request dated May 17, 2005, NL-05-0740). This is discussed further below.

b. For the storage of fuel assemblies in the MPC discussed in the criticality analysis report (CN-CRIT-207), the design basis assembly was determined by comparing k_{eff} of fresh Westinghouse standard and OFA fuel assemblies each with enrichments of 2.27 w/o and 3.0 w/o U-235. The standard assembly was found to be more reactive. This result was considered to be applicable to fuel of higher initial enrichments and burnups. During the discussions between SNC and the NRC in a teleconference on June 1, 2005, the NRC requested the results of an additional confirmatory calculation with burnt OFA fuel at 5.0 w/o to compare to the burnup limit of the design basis standard assembly at this same enrichment. This is discussed further below.

An additional sensitivity calculation was performed to determine the most reactive fuel assembly, or design basis fuel assembly, for loading of the MPC at an enrichment of 5.0 w/o. The sensitivity calculation was performed with a burnt Westinghouse OFA fuel assembly design. The OFA assembly at 5.0 w/o is slightly more reactive than the standard fuel assembly. However, the method used for determining the burnup requirement at 5.0 w/o for the standard fuel assembly in Table 3-5 and Figure 4-1 of the criticality analysis report (CN-CRIT-207) conservatively overestimates the burnup requirement. The conservatism is sufficient to offset the higher reactivity of the OFA fuel assembly such that the burnup requirements for the OFA assembly would remain bounded by the requirements in Table 3-5 and Figure 4-1 of the criticality analysis report (CN-CRIT-207).

Separately, it was determined that the most reactive fuel mishandling event occurred by misloading a fresh Westinghouse OFA fuel assembly into the center of the MPC. Note that the misloading event was also simulated with the Westinghouse Standard fuel assembly. The Westinghouse OFA fuel assembly design produced a slightly higher reactivity result (by 0.17 % delta k_{eff}) than the Westinghouse Standard fuel assembly design.

NRC Draft RAI No. 9

In Section 3.1, SNC stated that the DIT code was used to generate the isotopic concentrations for each segment of the axial profile. Specifically, SNC stated that the fuel and moderator temperatures used in the analysis were based on mid-cycle temperature profiles for a typical 1000 MWe pressurized water reactor. Since the in-core conditions during irradiation and depletion can have a significant effect on the burnup profile a spent fuel assembly, the staff requests that the licensee describe the analyses performed to demonstrate that the profiles used are appropriately representative of the historical operation at Farley. Additionally, the staff requests that SNC provide additional information that demonstrates that the assumptions used in the DIT depletion analyses for fuel and moderator temperatures and RCS soluble boron concentrations result in an appropriately conservative prediction of burnup for the fuel assemblies to be loaded in the MPC-32 at Farley.

SNC Response to Draft RAI No. 9

The following parameters were employed for the DIT calculations which simulated the in-core production of the fuel isotopics :

1) A constant soluble boron concentration, employed to simulate mid-cycle or the time averaged soluble boron concentration, equal to 800 ppm.

2) An average water temperature, employed to simulate the core average water temperature, equal to 579.95°F.

3) An average fuel temperature, employed to simulate the core average fuel temperature, equal to 944.12°F.

4) The following core operational data from Farley Unit 1, which is also typical for Farley Unit 2, was used:

Farley Unit 1	Vessel Hot Full Power Average (°F)	Cycle Average Boron (ppm)
Cycle 1	570.5	460
Cycle 2	570.5	480
Cycle 3	570.5	480
Cycle 4	570.5	500
Cycle 5	570.5	460
Cycle 6	575.0	650
Cycle 7	575.0	775
Cycle 8	575.0	721
Cycle 9	575.0	778
Cycle 10	575.0	769
Cycle 11	575.0	781
Cycle 12	575.0	740
Cycle 13	575.0	751
Cycle 14	575.0	821

Cycle 15	575.0	766
Cycle 16	574.3	705
Cycle 17	572.2	747
Cycle 18	572.2	781
Cycle 19	572.2	793
Cycle 20	572.2	778

The soluble boron concentration of 800 ppm employed in the DIT calculations conservatively bounds the cycle average soluble boron concentration for all prior Farley Unit 1 operations except for Cycle 14. Note that most fuel assemblies are irradiated for more than one cycle and therefore assemblies loaded in Cycle 13 or Cycle 14 would experience a (time-averaged) cycle average soluble boron concentration less than 800 ppm. Lastly, it should be stated that higher soluble boron concentrations produce a harder neutron spectrum and more conservative burnup limits.

The moderator temperature employed in the DIT calculations is slightly higher than the core average moderator temperature for all prior Farley Unit 1 operations and therefore would produce slightly conservative burnup limits.

Operational fuel temperature values are not available and therefore a direct comparison is not possible. However, the fuel temperature employed in the DIT calculations is based upon a fuel temperature correlation that is dependent on linear heat generation rate and assembly burnup. The linear heat rate employed to generate the fuel temperature value is approximately the same as the current Farley Unit 1 core average linear heat (after the power uprate). Therefore, it is concluded that the fuel temperature value employed in the DIT calculations will produce accurate or slightly more conservative burnup limits for the multi-purpose canister.

NRC Draft RAI No. 10

In Footnote 10 on Table 3-3, "k_{eff} for the Various Physical Tolerance Cases for the MPC-32," SNC states that "... uncertainty calculations are performed at a nominal ^{235}U enrichment of 2.08%." Since the enrichment chosen for performing the uncertainty analyses can have an effect on the calculated value of the uncertainty, the staff requests that the licensee provide additional information that demonstrates that the enrichment chosen will result in a conservative prediction of the uncertainties.

SNC Response to Draft RAI No. 10

As stated in Section 3.4, the following methodology was employed to determine the delta k_{eff} value associated with an enrichment uncertainty equal to 0.05 w/o U-235.

1) A quadratic fit of k_{eff} versus initial enrichment values for fresh fuel assemblies was performed from 2.1 to 4.0 w/o U-235.

2) The derivative of k_{eff} with initial enrichment was evaluated at 3.0 w/o U-235. This derivative value was then multiplied by 0.05 w/o U-235 to determine the enrichment uncertainty.

Note that this approach is conservative for all initial enrichments greater than 3.0 w/o U-235. Also, the enrichment uncertainty for 2.09 w/o U-235 (the highest initial enrichment which does not require assembly burnup) would have approximately the same effect as the uncertainty calculated for 3.0 w/o U-235 when statistically combined with the remaining uncertainties. Therefore, it is concluded that evaluating the enrichment uncertainty at the "lower end" (compared to the lowest initial enrichment) of the initial enrichment range is sufficiently conservative.

NRC Draft RAI No. 11

In Section 4.1, "Allowable MPC-32 Storage Conditions," SNC provides a list of requirements for storage of assemblies in the MPC-32. Since the staff's approval of the proposed amendment is contingent on continued compliance with the requirements of the HI-STORM 100 Certificate of Compliance (CoC), the staff requests that the licensee verify that the requirements listed in Section 4.1 are in accordance with any limitations, conditions, or technical specifications in the HI-STORM 100 CoC and that nothing proposed in this amendment is intended to contradict or violate any Part 72 regulations, the CoC, or any other NRC regulations or requirements.

SNC Response to Draft RAI No. 11

Section 4.1 of Westinghouse Calculation Note Number CN-CRIT-207 provides analysis of configurations that, if utilized in conjunction with the burnup versus enrichment limits contained in Figure 4.1, will meet the NRC criteria of $k_{eff} \leq 0.95$ when flooded with borated water and $k_{eff} < 1.0$ when flooded with unborated water. SNC recognizes that the analysis may include components or configurations not currently authorized by CoC 1014. However, incorporation of these components or configurations in the analysis does not invalidate the conclusions of the analysis or authorize loading of non-approved components into MPC-32s. SNC also recognizes that the approved content for the MPC-32 is described in CoC 1014, Appendix B, and changes to the approved content requires prior NRC approval in the form of an amendment to the CoC. Accordingly, SNC will comply with both the proposed Part 50 Technical Specifications and the Part 72 Certificate of Compliance (CoC) 1014 requirements during cask loading operations in the cask storage area.