Progress Energy

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AUG 1 6 2010

Serial: HNP-10-088
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO SECOND REQUEST FOR ADDITIONAL INFORMATION
REGARDING AMENDMENT TO REMOVE CREDIT FOR BORAFLEX IN
BOILING-WATER REACTOR SPENT FUEL POOL STORAGE RACKS
(TAC NO. ME0012)

References: 1. Letter from C. L. Burton to the Nuclear Regulatory Commission (Serial: HNP-08-075), "Technical Specifications 5.6.1.3.a and 5.6.1.3.b – Incorporation of Updated Criticality Analyses to Reflect Removal of Credit For Boraflex in BWR Spent Fuel Pool Storage Racks," dated September 29, 2008 (ML082800400)

2. Letter from C. L. Burton to the Nuclear Regulatory Commission (Serial: HNP-09-007), "Supplement to Technical Specifications 5.6.1.3.a and 5.6.1.3.b – Incorporation of Updated Criticality Analyses to Reflect Removal of Credit For Boraflex in BWR Spent Fuel Pool Storage Racks," dated January 16, 2009 (ML090230341)

3. Letter from C. L. Burton to the Nuclear Regulatory Commission (Serial: HNP-09-087), "Second Response to Request for Additional Information Regarding Amendment to Remove Credit for Boraflex in the Boiling Water Reactor Spent Fuel Pool Storage Racks (TAC NO. ME0012)," dated January 18, 2010 (ML100250856)

4. Letter from M. Vaaler, Nuclear Regulatory Commission, to C. L. Burton, "Shearon Harris Nuclear Power Plant, Unit 1 – Second Request for Additional Information Regarding Amendment to Remove Credit for Boraflex in Boiling-Water Reactor Spent Fuel Pool Storage Racks (TAC NO. ME0012)," dated July 20, 2010 (ML101940252)

Ladies and Gentlemen:

The Harris Nuclear Plant (HNP) received a request from the NRC dated July 20, 2010, (Reference 4) for additional information needed to facilitate the review of HNP's License Amendment Request to revise its Technical Specifications to incorporate an updated criticality

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analysis reflecting the removal of credit for Boraflex in BWR spent fuel pool storage racks. This original request was submitted as Serial: HNP-08-075 (Reference 1) and supplemented via Serial: HNP-09-007 (Reference 2). The Enclosure to this current submittal contains HNP's responses to this latest request for additional information.

In accordance with 10 CFR 50.91(b), HNP is providing the state of North Carolina with a copy of this response.

This document contains no regulatory commitments.

Please refer any questions regarding this submittal to Mr. Dave Corlett, Supervisor – Licensing/Regulatory Programs, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on AUG 1 6 2010.

Sincerely,

Christopher L. Burton

CLB/kms

Enclosures: 1. Response to Second Request for Additional Information
2. Retyped Technical Specification Pages

cc: Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, Section Chief, DENR
Mr. L. A. Reyes, NRC Regional Administrator, Region II
Ms. M. G. Vaaler, NRC Project Manager, HNP

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Summary

By letter dated September 29, 2008, as supplemented by letters dated January 16, 2009, August 12, 2009, and January 18, 2010, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., submitted a proposed amendment for the Shearon Harris Nuclear Power Plant, Unit 1.

The proposed amendment would modify Technical Specification (TS) Sections 5.6.1.3.a and 5.6.1.3.b to incorporate the results of a new criticality analysis. Specifically the TSs would be revised to add new requirements for the Boiling Water Reactor (BWR) spent fuel storage racks containing Boraflex in Spent Fuel Pools A and B. The requirements for the BWR spent fuel racks as currently contained in TS 5.6.1.3 would be revised to specify applicability to the spent fuel storage racks containing Boral in Spent Fuel Pool B.

The U.S. Nuclear Regulatory Commission staff has determined that it needs responses to the following questions in order to complete its review.

Request 1:

The technical report (Holtec Report No. HI-2043321, *Criticality Safety Analyses of BWR [Boiling-Water Reactor] Fuel Without Credit for Boraflex in the Racks at the Harris Nuclear Power Station)* states that the temperature coefficient of reactivity is positive and the void coefficient is negative.

a.   Were the temperature and void coefficients checked for multiple points covering the range of the proposed burnup credit loading curves and the range of boron concentrations credited in the analysis? If not, describe/justify what was done.

Response: The temperature and void coefficient calculations were performed over the range of enrichments from $1.5 - 4.6$ weight-percent, with burnups from $0 - 55$ gigawatt days per metric ton of uranium (GWD/MTU) and with full rods and partial rods. These calculations were all performed at 4 years cooling time and did not include soluble boron. Reference Table 1 (Attachment to this Enclosure). The analysis only credits 325 parts per million (ppm) soluble boron and this amount is not expected to have a significant impact. Sensitivity studies will be provided to quantify the effect, with additional studies quantifying the effect of soluble boron on the temperature coefficient of reactivity. The technical report will be updated to include all additional information.

b.   The temperature coefficient should include both fuel and water temperatures. It would seem that the only way the temperature coefficient could be positive would be if k_{eff} increased with decreasing water density. This indeed happens with over-moderated arrays. Unless the void coefficient is defined in some unusual way, a negative void coefficient means that as void increases (effectively decreasing water density), k_{eff} goes

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down. It is not clear how a system can have a positive temperature coefficient and a negative void coefficient. Explain how the system has a positive temperature coefficient and a negative void coefficient.

Response: The calculations for the temperature coefficient (i.e. moderator temperature coefficient) assume an equilibrium temperature and therefore both the fuel and moderator are considered at the same temperature. Table 1 (Attachment to this Enclosure) presents the calculations that were performed for the temperature coefficient. These calculations form the basis for the results that are presented in Table 8 (Reactivity Effect of Temperature and Void Content 4.0 wt%, 45.0 MWD/kgU) of HI-2043321, Rev. 6 (submitted with SERIAL: HNP-09-098, dated January 18, 2010, ML100250858). The delta k calculations for the results in Table 8 of HI-2043321, Rev. 6, and those presented in Table 1 (Attachment to this Enclosure), use a different reference temperature. As it can be seen from both Table 8 of HI-2043321, Rev. 6, and Table 1 (Attachment to this Enclosure), the temperature and void coefficient are both positive. It appears that the statement in the text, "At the submerged depth of the storage pool, the maximum temperature at boiling is 120°C, and the void coefficient of reactivity is negative." is incorrect and will be revised or removed.

Request 2:

From the description provided in Section 5.2 of the technical report it appears that the bounding axial burnup profile was determined by averaging 4 profiles selected from among 16 that were provided by Progress Energy. The initial 16 profiles were from assemblies that had initial enrichments around 4 weight-percent uranium-235 (^{235}U) and had accumulated burnup values between 30- and 46-gigawatt days per metric ton of uranium (GWd/MTU). It is not clear that this approach yielded a conservative axial burnup profile. Address the following issues:

a.   Provide justification for using the limited set of 16 profiles provided by Progress Energy. This justification should address coverage of the ranges in fuel designs (i.e., GE3 through GE13), variations of assembly features with the designs (i.e., Gadolinium (GD) rod usage, enrichments, blankets, axial dimensions), variations in relevant fuel depletion history (i.e., power, temperature, void distribution, control rod usage), and the range of the final assembly burnup values to be credit by the burnup credit analysis (i.e., 1.4 to 54.2 GWd/MTU).

b.   Provide justification for the further down-selection to 4 profiles. The justification should address why the 4 selected profiles are bounding compared to the 16 provided by Progress Energy.

c.   It appears to be inappropriate to average the 4 selected profiles. Describe how an average profile can be bounding compared to the 4 profiles used to generate the average profile.

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The information provided in the technical report is not adequate to support a conclusion that the axial burnup profile in Table 6 conservatively bounds the axial burnup profiles for the fuel to be stored in the spent fuel storage racks. Provide a stronger justification supporting use of the specified axial burnup profile, or perform and document a more thorough analysis to determine one or more bounding axial burnup profiles.

Response (all): The method used to determine appropriate axial burnup profiles, as presented in the original and supplemental submittals, was typical and standard methodology approved by the NRC at the time the analysis was generated. However, ongoing changes in the evaluation of spent fuel pool criticality issues have resulted in new precedence identifying the existence of more conservative alternate methods of analysis. Therefore, new axial burnup profiles will be determined, including the performance of sensitivity studies to determine the appropriateness of using the new axial burnup profiles on the various fuel types. These sensitivity studies will include 3D calculations that model the axial variation in fuel assembly design and the appropriate range of operating parameters and fuel storage criteria.

Request 3:

The analysis report should describe assumptions, approximations, and simplifications used in the analysis and should address their impact on the calculated k_{eff} value and on the total uncertainties used to calculate the maximum k_{eff} values. Address the following:

a. Use of CASMO-4 typically includes a "lumped fission product" model. This modeling simplification/approximation is not described in the technical report.

Response: The treatment of the lumped fission products will be addressed using the methodology discussed in Holtec technical report HI-2104598, "Sensitivity Studies to Support Criticality Analysis Methodology." This report, provided in support of the Beaver Valley application currently in NRC review, describes the derivation and treatment of a lumped fission product uncertainty factor that is statistically combined with the other uncertainties in the analysis. The additional studies required to address this Request will be provided.

b. For this analysis, CASMO-4 was used to generate the burned fuel compositions used in the Monte Carlo N-Particle Transport Code (MCNP) models. Were the lumped fission product compositions used in the MCNP models or were they discarded? The modeling of lumped fission products in CASMO and subsequent handling of lumped fission products was not described in the technical report.

Response: The lumped fission products compositions were used in the MCNP models. Reference response to 3.a. above.

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c.   Typically, the fuel composition calculations include either some post-irradiation decay
period or some post-processing of the CASMO burned fuel compositions to remove
xenon and, in some cases, to convert neptunium-239 (^{239}Np) to plutonium-239 (^{239}Pu).
Post-irradiation modeling and adjustment of the burned fuel compositions is not
discussed in the technical report.

Response: The depletion code CASMO-4 is used to calculate the spent fuel isotopic
composition. The spent fuel composition (isotope number densities) calculated by CASMO-4 is
specified as input into MCNP4a. The CASMO-4 depletion calculations to obtain the isotopic
compositions for MCNP4a are performed with one calculation for each enrichment step,
typically in steps of 0.5 weight-percent ^{235}U and for burnup steps in increments of 2.5
GWD/MTU. The isotopic composition for any given burnup is then determined by linear
interpolation. The post-processing of the CASMO-4 isotopic concentrations utilized removes the
isotope xenon-135 (^{135}Xe) and adds the isotopic concentration of ^{239}Np to the isotopic
concentration of ^{239}Pu.

Request 4:

One of the primary products of the analysis is the burnup credit loading curves, which are
presented (1) in equation form in Section 1.0 of the technical report, (2) in Table 5, and (3) in
Figures 2 and 2a. The following issues should be addressed concerning the burnup credit
loading curves:

a.   The loading curves utilize the "initial maximum planar average enrichment" (IMPAE).
This quantity is not defined in the analysis. Provide a clear and complete definition of
IMPAE.

Response: Initial enrichment is defined as the ^{235}U enrichment of the fuel at zero exposure
(i.e., zero burnup, or fresh fuel). Planar average enrichment is defined as the average ^{235}U
enrichment taken over all fuel pins in a plane of fuel. The maximum planar average enrichment
is defined as the greatest planar average enrichment of any plane of fuel in the fuel assembly.
Thus, Initial Maximum Planar Average Enrichment (IMPAE) is the highest average enrichment
over any plane of fuel prior to irradiation of the fuel.

b.   Other than for natural uranium blankets, will any single fuel rod have more than one fuel
enrichment? In other words, do enrichment and the IMPAE vary with axial zone? If so,
describe how axial zone enrichment variation is taken into account in the analysis.

Response: While a single fuel rod may have more than one fuel enrichment (i.e., enrichment
may vary with axial zone), IMPAE does not vary with axial zone. IMPAE, a single planar
enrichment value, accounts for axial zone enrichment variation in the analysis by conservatively
bounding any axial fuel assembly enrichment variation. This is because reactivity increases with

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enrichment and the IMPAE is the maximum planar average enrichment of any axial zone in the assembly.

c.   The analysis uses an average enrichment for each plane rather than the detailed pin-by-pin enrichment distribution. Provide a justification for this modeling simplification. If appropriate, include bias and uncertainty associated with the modeling simplification.

Response: The modeling simplification using an average enrichment for each plane rather than the detailed pin-by-pin enrichment is justified because it is a conservative modeling simplification for BWR fuel assemblies stored in the Harris spent fuel pools. This modeling simplification has previously been reported as conservative for both BWR (i.e., Response to Request for Additional Information No. 27 in AREVA Report ANP-2843Q1NP, ADAMS Accession Number ML101650230) and PWR fuel lattices (i.e., Appendix B.2.1 of AREVA Report ANP-2779-001, ADAMS Accession Number ML083360624).

The reactivity impact of this modeling assumption is also shown as Figure 1 in the Attachment to this Enclosure for the highest planar enrichment GE13 lattice utilized in the highest burnup BWR fuel assembly stored in any Harris spent fuel pool, except that no Gd was included in the model. The calculations were performed with CASMO-4 in the rack geometry following depletion in the core geometry. The results show modeling an average planar enrichment rather than the detailed pin-by-pin enrichment is conservative through fuel burnups of approximately 45 GWD/MTU, which is the highest burnup of any BWR assembly in any Harris spent fuel pool. Since there are no plans to receive additional BWR fuel shipments for storage in the HNP fuels, analysis to this 45 GWD/MTU value is realistic and bounding.

The results do show uniform enrichment can become negligibly non-conservative by 45 GWD/MTU assuming depletion at 0 percent void history. However, the analysis supporting the burnup credit loading curves assumes all nodes deplete at 77 percent void history. The response to Request 9 shows modeling fuel that actually depletes at 0 percent void history as depleting at 77 percent void history is significantly and increasingly conservative with increasing burnup. It should also be noted that only the lower assembly nodes deplete at 0 percent void history in a BWR.

d.   No range is provided for acceptable use of the burnup credit loading curves. It appears that their use should be limited to initial enrichments no lower than 1.5 weight-percent ^{235}U and no greater than 4.6 weight-percent ^{235}U. Confirm the acceptable range for use of the burnup credit loading curves.

Response: The burnup credit loading curves are applicable to fuel with IMPAE below 1.5 weight-percent provided the burnup requirements for 1.5 weight-percent are satisfied. All BWR fuel assemblies in any Harris spent fuel pool with IMPAE below 1.5 weight-percent exceed a burnup of 4.9 GWD/MT, and there are no plans to receive additional shipments of BWR fuel.

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This burnup clearly exceeds the requirements of the burnup credit loading curves for 1.5 weight-percent. Since the burnup credit loading curves are not applicable to fuel with IMPAE greater than 4.6 weight-percent, no BWR fuel assemblies stored in any Harris spent fuel pool exceed an IMPAE of 4.6 weight-percent.

Request 5:

The second bullet in Section 2.0 of the technical report is:

> *Minor structural materials were neglected; i.e. spacer grids were conservatively assumed to be replaced by water.*

Confirm that this modeling simplification was checked over the full range of the burnup credit loading curves and over the full range of soluble boron concentrations credited. If appropriate, include bias and uncertainty associated with the modeling simplification.

Response: While the analysis did not explicitly calculate the effect of neglecting the spacer grids, it is expected that this assumption is conservative for pure water. Calculations will be performed and if the assumption is non-conservative for borated water, credit could be taken for the non-credited soluble boron. The significant margin that exists between the credited soluble boron concentration of 325 ppm and the tech spec limit of 2000 would more than offset any potential non-conservatism.

Request 6:

Section 4.1 of the technical report provides a description of the limiting fuel assembly design. No information is provided on the use of fuel rods that contain gadolinium. Provide information concerning the ranges of the numbers of Gd rods used, the Gd loading, and the axial location of the Gd in the Gd rods.

Response: Detailed gadolinium arrangements are not provided because they are bounded by the supporting analysis. As explained in the response to Request 7 below, since it is conservative to not model gadolinium, the supporting analysis does not model gadolinium.

Request 7:

There appears to be an unstated assumption that it is conservative not to model the gadolinium present in some of the fuel rods. NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers for Pressurized Water Reactor (PWR) Burnup Credit," does present some information for full-length Gd rods in PWR assemblies that one might extrapolate to BWR fuel to support

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such a hypothesis. However, the validity of applying this assumption to BWR fuel has not yet been fully demonstrated. BWR fuel features that may affect this issue include axial zoning of gadolinium and radially varying ^{235}U enrichments that are somewhat correlated with the Gd rod locations.

This issue was raised in earlier requests for additional information (RAIs); the response provided in Enclosure 1 to the licensee's supplemental letter dated January 18, 2010, and its Attachment 1 has been reviewed. The information provided does not fully address the issue. The calculations documented in the RAI response were performed with CASMO-4, a two-dimensional (2D) lattice code. This calculational approach does not address the impact of axially dependent design features such as blankets, part-length rods, axial zoning of Gd, and, if utilized, axial zoning of fuel ^{235}U enrichments.

Further, the RAI response notes that "representative Gadolinium loading and location of Gadolinium rods were used." A three-dimensional (3D) analysis should be performed using gadolinium rod loading and radial and axial locations that are expected to cover the range of acceptable configurations. Provide additional justification supporting the assertion that neglecting gadolinium is a conservative simplification.

Response: Gd displaces fissionable fuel. Uranium oxide fuel poisoned with Gadolinium oxide has lower density than equivalent fuel that contains no Gd, because Gadolinium oxide is less dense than Uranium oxide. Gd poisoned fuel containing the same weight percent ^{235}U enrichment as unpoisoned fuel also contains less total fissionable ^{235}U due to its lower density. Gd does not convert to fissionable isotopes by neutron capture. Additionally, fuel poisoned with Gd incurs a small residual reactivity penalty due to conversion of the highly absorbing ^{155}Gd and ^{157}Gd isotopes to ^{156}Gd and ^{158}Gd via neutron capture. The absorption cross sections of ^{156}Gd and ^{158}Gd are small, but nonetheless produce a small residual reactivity penalty that diminishes very slowly with fuel burnup.

Depletion of a Gd poisoned fuel lattice at different void histories will show that depletion at zero void history results in greater reactivity at the relatively low burnup associated with peak reactivity. This is due to the increased thermal flux at zero void history depleting Gd before the competing effect of higher plutonium production at higher void dominates at increased exposure. However, a Gd poisoned lattice will not become more reactive than one of equal ^{235}U weight-percent enrichment that is not poisoned since Gd displaces fissionable fuel, does not convert to fissionable fuel via neutron capture and imparts a small residual reactivity penalty to the fuel.

Comparisons of peak reactivity, or more directly, negative Gd worth, as a function of burnup for different void histories are only relevant if the negative worth of Gd is credited in a criticality analysis. Since the analysis supporting the Harris burnup credit loading curves does not credit the negative worth of Gd, it is conservative for the analysis to neglect Gd.

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Fuel geometry and specific depletion characteristics do not affect the characteristics of Gd as a burnable absorber (i.e., Gd displaces fissionable fuel, does not convert to fissionable fuel via neutron capture and imparts a residual reactivity penalty to the fuel). For this reason, it is not necessary to explicitly calculate these characteristics for different fuel geometries or depletion characteristics in detail. In-rack CASMO-4 calculation results are presented in Figures 2 through 4 of the Attachment to this Enclosure for 0-, 40- and 77-percent void history depletions for a GE13 lattice of uniform 4.6 weight-percent enrichment containing 11 Gd rods at 5 weight-percent Gd for the purpose of illustrating the trends discussed above.

The impact of blankets and part-length rods will be addressed in the response to Request 8. As explained above, since the impact of axial zoning of Gd does not result in a more reactive lattice than neglecting the modeling of Gd, it is acceptable for the analysis supporting the burnup credit curves to ignore axial Gd zoning. The impact of axial zoning of fuel ^{235}U enrichments does not result in a more reactive fuel assembly than modeling the entire enriched length of the assembly at the higher maximum planar average enrichment. Therefore, it is acceptable for the analysis supporting the burnup credit curves to model all enriched lattices at the IMPAE and to ignore axial ^{235}U zoning. The impact of radial enrichment variation is addressed in the response to Request 4.c.

Request 8:

The text in Section 4.1 of the technical report refers to Tables 7 and 7a showing comparisons of the reactivities of the fuel assembly designs that are or will be stored in the BWR spent fuel storage racks. These tables are used to justify that the GE13 assembly design is the limiting fuel assembly design.

a.   No details are provided in the report concerning how these calculations were performed. Describe the calculations, including information on codes used, geometry, axial burnup distributions used, variations within each assembly design considered, fuel channels, gadolinium modeling, etc. Note that simple 2D comparisons of these assembly designs with varying 3D features may not be appropriate.

Response: The methodology used was a comparison of the fuel assemblies using the CASMO-4 code. Calculations were performed with each fuel assembly type for enrichments between 2 and 4.6 weight-percent with burnups above and below the minimum burnup requirements. The effect of cooling time was also considered. Each fuel assembly was modeled in CASMO-4 for depletion and restarted in the storage rack geometry to yield the infinite multiplication factor. Reference Tables 2, 3, 4 and 5 in the Attachment of this Enclosure. Additional sensitivity calculations will be performed in 3D that consider the impact of specific fuel designs, operating history, axial burnup profiles and effect of blanket length modeling assumptions and the technical report updated accordingly.

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b.   Provide comparisons with and without design-specific fuel channels.

A similar issue was raised in an earlier RAI; the response provided in Enclosure 1 to the licensee's supplemental letter dated January 18, 2010, and its Attachment 1 has been reviewed. The response appears to address only GE13 fuel. The analysis results presented in Tables 7 and 7a should have included results for assemblies with and without channels to establish that GE13 fuel with channels is more reactive than the previous fuel designs with or without channels, provided assemblies of those designs had met their design-specific minimum cooling time requirements. Provide additional justification for the assertion that GE13 fuel with fuel channels is more reactive than previous designs with or without fuel channels.

Response: Additional calculations will be performed to determine the reactivity of all fuel types with and without channels.

c.   Provide comparisons for assemblies with 1.5 weight-percent initial enrichment.

Response: Additional calculations will be performed to provide comparisons for assemblies with 1.5 weight-percent initial enrichment.

d.   The cooling times presented in Table 7a for GE3 through GE10 vary from 9 to 27 years. It appears that the analysis relies on design-dependent post-irradiation cooling times beyond the 4 or 7 year cooling times stated for the burnup credit loading curves. Explain why design-dependent post-irradiation cooling times are not required or describe how they will be implemented. Address why the cooling times in Table 7a vary from the cooling times in Table 7 and in the first paragraph of Section 4.1 of the technical report.

Response: Table 7 (Assembly Design Reactivities (4 years)) and Table 7a (Assembly Design Reactivities (7 years)) were previously provided in Holtec report HI-2043321, Rev. 6 (submitted with SERIAL: HNP-09-098, dated January 18, 2010, ML100250858) as additional justification for using the GE13 fuel assembly as the design basis assembly. Table 7 presents results at 4 years cooling time for all fuel types. Some fuel types have a higher reactivity than the GE13 at 4 years and for those fuel assembly types, Table 7 presents additional results at greater cooling times to show that when longer cooling times are taken into consideration, the GE13 is bounding.

Table 7a is similar to Table 7 with the exception that it presents the fuel assemblies reactivity's at the longer cooling times only, since Table 7 previously presented that, for short cooling times, other fuel assemblies can be more reactive than the GE13. The only design-dependent post-irradiation cooling time credit being used is to show that when actual cooling times are considered the GE13 remains bounding. Design dependent cooling times are therefore omitted from the analysis since the analysis only credits 4 and 7 years cooling time (the cooling time range for which the GE13 is bounding).

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Request 9:

Section 4.2 of the technical report discusses reactor depletion modeling. The text in this section notes that the void fraction value is taken as the upper bound of the core operating parameters of Brunswick. The text at the top of page 6 includes the following:

> *The neutron spectrum is hardened by each of these parameters, leading*
> *to a greater production of plutonium during depletion, which results in*
> *conservative reactivity values.*

Other BWR plants have performed depletion calculations which showed that for their plants a zero void fraction was conservative. Confirm that calculations were performed for BWR fuel depleted in the Brunswick reactor showing that use of the high void fraction in fuel depletion calculations was conservative.

Response: As described in the response to Request 7 above, zero void fraction depletion calculations are only conservative if the negative worth of Gd is credited in a criticality analysis. The analysis supporting the Harris burnup credit loading curves does not credit the negative worth of Gd. The in-rack reactivity calculation results provided in Request 7, those that neglect Gd, are shown together on the same plot (Figure 5 of the Attachment to this Enclosure). These results show that use of a high void fraction in fuel depletion calculations is conservative relative to use of a lower void fraction.

Request 10:

The simplified fuel storage rack model used in the analysis is described in Section 4.3 and in Figure 3 of the technical report. The simplified model differs from the actual rack geometry in a few significant ways. First, the simplified model creates a flux trap type geometry at the corners of each cell. The actual geometry does not have water between steel plates at the corners. Second, the simplified geometry assumes that the wrapper extends all the way to the corners. This is likely not correct for the actual geometry. Confirm that a detailed rack model calculation was performed showing that the simplified model yields results equivalent or conservative when compared to the detailed model results. Note that this assumption should be checked over the range of the burnup credit loading curve and over the ranges of soluble boron concentrations credited. If appropriate, include bias and uncertainty associated with the modeling simplification.

Response: This Request is similar to a previous RAI (reference SERIAL: HNP-09-007, Request 4, ML090230341). As part of the response to that previous RAI, sensitivity calculations were performed that indicate agreement between the simplified model and the explicit model. The calculations that were previously performed were calculated for 4.6 weight-percent fuel at 42.5 GWD/MTU in pure water conditions. Additional studies will be performed to more

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completely characterize the impact of other enrichment and burnup combinations and soluble boron concentrations.

Request 11:

Simplified fuel assembly models were used in the analysis. Address the following issues related to fuel assembly models:

a.   Were fuel compositions in each axial layer modeled on a pin-by-pin basis or were average compositions used?

Response: Average compositions were used.

b.   Was buildup of Pu in the natural uranium blankets modeled?

Response: Yes the buildup of Pu in the natural uranium blankets was modeled.

c.   What is the impact on in-rack k_{eff} of extending the 8-inch top blanket to 12 inches?

Response: The impact of axial variations in blanket length will be addressed as part of the studies performed for Request #8.

d.   Table 2 indicates that only GE13 fuel assemblies have part-length rods. Did any of the fuel designs other than GE13 have part-length rods or blankets?

Response: The impact of part length rods and axial blankets will be further addressed as part of the studies performed for Request #8. Per the Holtec report HI-2043321, Rev. 6 (submitted with SERIAL: HNP-09-098, dated January 18, 2010, ML100250858), only the GE13 assembly had part-length rods (Table 2: BWR Fuel Characteristics, page 16).

e.   It seems appropriate to use different axial burnup distributions for fuel assemblies with different axial features. Provide justification for use a single axial burnup distribution for modeling fuel assemblies with different axial features.

Response: All fuel assembly modeling simplifications will be addressed in the response to Request #8.

Describe and justify fuel assembly modeling simplifications used. If appropriate, include bias and uncertainty associated with the modeling simplification.

Response: Bias and uncertainty associated with the modeling simplification will be included in the justification of fuel assembly modeling simplifications, if appropriate.

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Request 12:

The next to last paragraph in Section 5.2 of the technical report states that separate CASMO-4 depletion calculations were made for each of the 25 axial segments. The details as to how these calculations were performed are not adequate to support the review. Address how the depletion calculations were performed, including details of Gd rod depletion, and how the CASMO-4 calculated fuel compositions were incorporated into the MCNP models.

Response: The depletion code CASMO-4 is used to calculate the spent fuel isotopic composition. The spent fuel composition (isotope number densities) calculated by CASMO-4 is specified as input into MCNP4a. The CASMO-4 depletion calculations to obtain the isotopic compositions for MCNP4a are performed with one calculation for each enrichment, typically in steps of 0.5 weight-percent ^{235}U, and for burnup steps in increments of 2.5 GWD/MTU for various cooling times. The isotopic composition for any given burnup is then determined by linear interpolation and are assembly average isotopic, not fuel pin specific. The post-processing of the CASMO-4 isotopic concentrations removes the isotope ^{135}Xe and adds the isotopic concentration of ^{239}Np to the isotopic concentration of ^{239}Pu.

With respect to reactivity control components, Gadolinium is not considered in the depletion calculations. However, since the Brunswick reactor uses control rod blades for reactor and power control during operation, the control rod blades are considered in the depletion calculations. Control rod insertion is limited to once and twice burned fuel assemblies since the control blade operating strategy at Brunswick for GE13 fuel did not allow fresh fuel to be placed in a core location that would have planned control rod insertion. Typical fuel is controlled for 3 GWD/MTU intervals then uncontrolled for 3 GWD/MTU intervals. To conservatively bound any control rod insertion, fuel assemblies are modeled with an initial interval of 12 GWD/MTU (i.e., first cycle) of uncontrolled operation followed by intervals of 3 GWD/MTU controlled and uncontrolled operation. This conservatism results from the fuel not actually being controlled for 3GWD/MTU and because the flux suppression of the control rod blade significantly decreases exposure accumulation.

With respect to the how the CASMO-4 depletion calculations are input into MCNP4a, it should be noted that the fuel assembly is broken up by zones from bottom to top: axial blanket zone, zone with full and part length rods, zone with full length rods only, and axial blanket zone. For each zone, the appropriate post-processed CASMO-4 calculated isotopic are input in the MCNP4a model. Finally, the depletion calculations for all axial zones conservatively deplete with the bounding operating parameters, i.e., no credit is taken for actual axial variation in operation parameters.

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Request 13:

A rather limited set of uncertainties due to manufacturing tolerances is presented in Section 6.1 of the technical report. In addition to the items listed on page 10, the uncertainties associated with manufacturing tolerances on the following should have been evaluated: wrapper thickness, wrapper width, Boraflex gap thickness, Boraflex gap width, fuel pin pitch, pellet outer diameter (OD), clad OD and thickness, channel inner diameter and thickness, water tube OD and thickness, initial Gd content, Gd pellet density, and the length and location of each axial zone.

Response: Additional sensitivity studies will be performed to include the effect of these tolerances and they will be included in the k_{eff} calculation.

Request 14:

The text in Section 6.1 of the technical report describes a temperature correction calculated using CASMO, which is applied as a bias to MCNP results in Tables 1 and 1a. Calculation of Δk values related to temperatures changing from 20° C to 150° F using CASMO-4 has not been validated. Further, application of Δk values calculated using a 2D model to a 3D system is questionable. Additionally, an estimate of the uncertainty in the Δk correction due to temperature should be provided. Provide better justification for the temperature correction and provide an estimate in the uncertainty in the correction.

Response: The use of CASMO-4 in the temperature range of 20°C to 150°F for the purpose of determining the temperature bias for MCNP cross sections has been previously performed on many other applications for criticality analysis. It is desirable to use CASMO-4 for this purpose due to the ease at which temperature effects are determined by the code. Conversely, MCNP has several features that make performing temperature effect studies more difficult. Sensitivity studies to support the determination of the temperature bias to correct for cross section temperature dependence will be performed.

Request 15:

The text in Section 6.1.1 of the technical report documents that a 5-percent reactivity decrement uncertainty is included in the uncertainties. The text does not clearly describe how this uncertainty was calculated. Confirm that the depletion uncertainty was calculated as 5-percent of the change in k_{eff} of the 3D spent fuel storage rack due to changing from fresh fuel with no Gd to a burned fuel assembly at its minimum burnup allowed by the loading curves.

Response: The calculations for the depletion uncertainty were performed with 3D MCNP models using fresh fuel uniformly distributed axially for the fresh fuel case and spent fuel at a point above the burnup enrichment curve. The fresh fuel did not contain Gd.

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For each calculation, the determination of the depletion uncertainty is performed by subtracting the reactivity of the spent fuel case from the reactivity of the fresh fuel case and multiplying by 0.05. This depletion uncertainty is then statistically combined with the other uncertainties.

Request 16:

Section 6.2.1 of the technical report discusses the analysis of eccentric location of fuel assemblies in each cell in the spent fuel storage racks. The analysis is inappropriately presented in the section on abnormal and accident conditions. The most reactive fuel assembly position is part of the normal conditions, not an abnormal condition. Second, confirm that the analysis was performed both with and without fuel channels. Lastly, confirm that the analysis included a configuration where all assemblies in a rack module were moved toward a single common interior corner near the center of a rack module.

Response: The eccentric positioning of a fuel assembly is a normal condition, not an accident condition. The calculations performed included the fuel channels. All the calculations were performed using a single cell infinite array model with all the fuel moved toward the corner, essentially creating a 2x2 model with the fuel in the center. Additional calculations will be performed with a larger full rack model with all the fuel assemblies eccentrically positioned toward a common interior corner near the center of the rack.

Request 17:

Section 6.2.3 of the technical report discusses evaluation of accidents involving a misloaded fuel assembly. Detail provided is not sufficient to support review. Provide the following information:

a.   Fully describe the model of the misloaded assembly. Provide information on the IMPAE, final burnup, axial burnup distribution used, elevation-dependent lattice design, presence or lack of a fuel assembly channel, etc.

Response: The misloaded fuel assembly is 4.6 weight-percent with 15 GWD/MTU burnup with a uniform axial burnup profile. The fuel assembly models the part length rods and water rods. It also includes the channel.

b.   Fully describe the rack model into which the misloaded assembly was placed. Provide information on the model extent (e.g., 15x15 spent fuel storage rack module, axial and radial boundary modeling) and burnup of fuel other than the misloaded assembly. Note that the misloaded assembly and normal loaded assemblies should have similar axial fission density profiles. If highly burned fuel assemblies are loaded with a low burnup

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misloaded assembly, neutronic interaction between the most reactive parts of the assemblies will be minimized.

Response: The misloaded fuel assembly model is a 7x7 array of storage rack cells fully loaded with fuel, with the misloaded fuel assembly in the center, and reflective boundary conditions on the rack edge. Each of the fuel assemblies in the model includes the fuel assembly channel. Three calculations were performed with different initial enrichment and burnup combinations for the spent fuel in the storage rack: 2 weight-percent at 10 GWD/MTU, 3 weight-percent at 25 GWD/MTU and 4.6 weight-percent at 50 GWD/MTU. As described in the License Amendment Request, the misloaded fuel assembly was consistently considered as a 4.6 weight-percent with 15 GWD/MTU burnup fuel assembly. All the fuel in the model uses uniform axial burnup profiles.

c.   Describe where inside and outside the rack module the misloaded assembly was modeled. Address the potential for placement of an assembly between and beside the fuel storage racks.

Response: The misloaded fuel assembly model is a 7x7 array of storage rack cells fully loaded with fuel, with the misloaded fuel assembly in the center. No calculations were performed for the misplacement of a fuel assembly (i.e. mislocated fuel assembly) outside of the fuel storage rack since this accident will be bounded by the misloaded fuel assembly due to the neutron leakage for the mislocated case.

Request 18:

Are controls implemented that preclude movement of PWR fuel assemblies near the BWR fuel storage racks? If not, the analysis should include a PWR fuel assembly next to the BWR fuel storage racks as a "normal" condition. It may also be necessary to consider a PWR assembly dropped on to the BWR spent fuel storage racks.

Response: There are currently no controls in place or planned to preclude movement of a PWR fuel assembly near the BWR fuel storage racks. Additional studies will be performed to account for the mislocation of a PWR fuel assembly outside of the BWR racks and determine the potential effect of dropping a PWR fuel assembly on top of a BWR rack.

Request 19:

Are controls implemented that preclude movement of more than one assembly of any type at the same time? If not, the normal and, possibly, abnormal conditions analysis should include consideration of the maximum number of assemblies that may be near the BWR spent fuel storage racks at any one time.

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Response: Yes, controls are in place to preclude movement of more than one assembly of any type at the same time. There is only one fuel handling bridge crane installed in the Harris fuel handling building to service the four spent fuel pools. This bridge crane has one hoist installed on the south side and no hoist installed on the north side.

Request 20:

The text in Section 1 of the technical report states that a soluble boron concentration of 300 parts per million (ppm) is required to maintain k_{eff} below 0.95 under normal conditions and that 325 ppm is required to assure k_{eff} is less than 0.95 for the misload accident. Do these soluble boron concentrations include margin to ensure that k_{eff} will be less than 0.95 with 95-percent probability and a 95-percent confidence level? If not, revise the required soluble boron concentrations to include such margin.

Response: The soluble boron calculations include margin to ensure that k_{eff} will be less than 0.95 with 95% probability and a 95% confidence level.

Request 21:

The analysis does not address interaction between rack modules within the BWR spent fuel storage region and between the BWR spent fuel storage racks and other fuel storage racks such as the PWR spent fuel storage racks. The analysis should have addressed interaction between rack modules or provide justification for not doing so.

The response to Request 6 in Enclosure 1 of the licensee's supplemental letter dated January 18, 2010, addressed a similar question. The RAI response indicates that the interface was already addressed in the PWR Boraflex Rack Criticality Analysis. In the referenced analysis, was the BWR Boraflex rack modeled with no Boraflex and at zero soluble boron? The normal condition, which now will include no Boraflex in the BWR racks, must be subcritical at zero soluble boron.

Further, implementation of the new BWR burnup credit analysis would result in storage of BWR fuel assemblies that will have axial fission density profiles that vary significantly with fuel burnup. The original PWR analysis included only fresh BWR assemblies. If the PWR analysis credited fuel burnup, the PWR/BWR interface model should consider optimizing interaction between the most reactive parts of the PWR and BWR assemblies. For example, one may find that the worst cases would include storage of highly burned BWR fuel next to highly burned PWR fuel or low burnup BWR fuel next to low burnup PWR fuel.

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Accordingly, the response to Request 6 is not sufficient to address the issue. Provide additional information to support that the PWR/BWR interface does not need to be revisited with the implementation of the new BWR fuel storage rack analysis.

Response: The rack interaction analysis referenced in response to Request 6 in Enclosure 1 of Harris supplemental letter dated January 18, 2010, performed a deboration (zero ppm boron) rack interaction analysis, which assumed the BWR rack Boraflex had fully degraded. The analysis concluded the interface condition was subcritical at zero soluble boron.

Axial fission density profile variation does not impact the conclusions of the referenced rack interaction analysis, because the referenced analysis considered a more reactive fuel assembly than the burned assemblies permitted for storage by the burnup credit loading curves. The referenced rack interaction analysis assumed GE13 BWR fuel assembly geometry uniformly loaded at 1.5 weight-percent enrichment with no burnup, no integral Gadolinium absorber, no part length rods (i.e., part length rods were extended to the full active fuel length) and blankets replaced with enriched fuel. The burnup credit loading curves require an assembly with IMPAE of 1.5 weight-percent to incur burnup in order to be acceptable for storage in the Boraflex racks. Because the referenced rack interaction analysis assumed a more reactive BWR assembly than could be stored in the BWR Boraflex racks following approval of the burnup credit loading curves, it is not necessary to further optimize interaction between BWR and PWR assemblies.

Request 22:

Table 2a of the technical report provides Brunswick core operating parameters used for depletion. The text in Section 4.2 identifies these parameters "as the upper bound (most conservative) of the core operating parameters of Brunswick." Are these parameters bounding on a local level or on a core average basis? The values used should be bounding locally, not just on average.

Response: The analysis conservatively considered the operating parameters along the full length of the fuel. These operating parameters were considered conservative, but were not developed as maximum possible values. In conjunction with the response to Request #8, additional calculations will be performed with maximum local parameters.

Request 23:

Provide the following information related to the specific power used for depletion:

a.   The information submitted by letter dated January 16, 2009, stated that maximum specific power for Brunswick was 26.7 MW/MTU. Table 2a of the technical report shows that 30 MW/MTU was used. Confirm that the assumed specific power is

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conservative considering the presence of fission products. Also, confirm that the assemblies depleted at pre-extended power uprate level are bounded.

Response: Additional studies will be performed to determine the impact of specific power on reactivity in the storage racks.

b. Show how specific power was calculated.

Response: In CASMO-4, specific power is input in units of watts/gram. Therefore, specific power is calculated as follows (Core Thermal Power)/(Mass Uranium) accounting for the units.

Request 24:

Table 4 includes the following text for the temperature increase "condition":

150 ° F used for normal storage condition. Higher temperatures are accident conditions with credit for soluble boron allowed.

What is a credible higher temperature and how much soluble boron is required to address the higher temperature abnormal condition identified in Table 4 of the technical report?

Response: The Holtec criticality analysis is performed at the maximum normal operating temperature of 150 °F. Higher temperatures are classified as accident conditions. Section 3.2, Accident Analysis, of Enclosure 2 to HNP-09-087 identifies 325 ppm as being the minimum soluble boron concentration required to maintain k_{eff} less than 0.95 in the BWR Boraflex racks under all accident conditions. This would include the higher temperature abnormal condition identified in Table 4. Additional research will need to be performed to verify the amount of soluble boron included in the higher temperature abnormal condition analysis.

Per FSAR section 9.1.3.1, "Design Basis":

 f)   The Standard Review Plan pool temperature requirement for the Normal Case, assuming a single active failure, is 140°F. The minimum decay time prior to movement of irradiated fuel in the reactor vessel will address both radiological and decay heat considerations. Administrative controls are placed on the minimum cooling time before transfer of spent fuel to the pools, to limit the fuel pool temperature to less than or equal to 150°F. The pool temperature requirement for the Abnormal Case is to be below boiling. The pool concrete design temperature is 150°F, but has been evaluated to 160°F.

Additionally, FSAR Section 9.1.3.3, "Safety Evaluation," provides a discussion of the pool temperature increase for a loss of cooling caused by CCW header isolation for a LOCA. Using

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the heat load listed for normal operations in pools "A" and "B", the pool temperature is predicted to reach 160°F at 7.2 hours after accident initiation. The CCW flow to the fuel pool cooling is available for restoration at five hours after the LOCA event initiation. This allows 2.2 hours to complete the manipulations to restore the nonessential header flow from the available CCW pumps. The heatup rate is conservatively based on the decay heat present in the spent fuel pools early in a fuel cycle. The starting temperature of the spent fuel pool is conservatively taken as 125.7 °F, based on a CCW supply temperature of 105°F and the listed heat load for the Normal Operations case.

Request 25:

Table 4 of the technical report describes the consequences of a dropped assembly and of a seismic event as "negligible". How much seismic movement (i.e., rack module sliding) is considered credible? Were calculations performed to show that the consequences of these conditions are negligible? If not, provide justification for not performing calculations.

Response: The design basis analysis model is a single cell infinite lattice model that uses 12 inches of water above and below the active fuel region (i.e., an infinite reflector). No credit is taken for rack-to-rack gaps. For the case were a fuel assembly might be dropped and rest horizontally on top of the racks, the result of a dropped fuel assembly would not result in a more reactive condition than the design basis model used in the analysis because the minimum distance from the top of the active length to the top of the storage rack is greater than 12 inches. A study will be performed to prove more than 12 inches of distance exists in the worst case stack up of tolerances on both fuel assembly dimensions (for all fuel types) and rack dimensions. As long as more than 12 inches exist, the scenario will be bounded by the design basis model.

Additionally, the scenario where seismic activity may cause the racks to slide with respect to each other cannot result in a more reactive condition than the design basis model, an infinite lattice that does not credit the gap between racks. Therefore, since these accident conditions are bounded by the design basis model, no additional calculations are required. Additional information will be transmitted to support the dimensional conclusion.

Request 26:

The validation analysis presented in Appendix A of the technical report should have evaluated bias trends as a function of plutonium content [e.g., g Pu/(g Pu + g U)] and soluble boron concentration. Evaluate the bias trends for these parameters or provide justification for not evaluating the trends using these parameters. Where appropriate, incorporate parameter dependent biases and uncertainties.

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Response: A new benchmark that includes the "Haut Taux de Combustion" (HTC) experiments has been performed. The new bias and bias uncertainty derived from this new Holtec benchmark for MCNP will be incorporated into the k_{eff} calculations.

Request 27:

Provide the following information related to the Technical Specifications (TS):

a. It appears that TS 5.6.1.3.a.5 and TS 5.6.1.3.a.6 can be in conflict (i.e., an assembly might meet TS 5.6.1.3.a.6 but not TS 5.6.1.3.a.5) since TS 5.6.1.3.a.6 does not specify a burnup requirement. The enrichment requirement is in the loading curve.

Response: The proposed TS 5.6.1.3.a will be revised. The specifications that are specific to the BWR fuel assembly conditions will be combined to eliminate the conflict.

b. Provide the NRC staff Safety Evaluation which approved the basis for TS 5.6.1.3.b.3, which states that "BWR assemblies are acceptable for storage in BWR Boral storage racks provided the maximum planar average enrichments are less than 3.2 weight-percent U235."

Response: The current Tech Spec 5.6.1.3 is written for both the Boraflex and Boral poisoned BWR racks with no enrichment limit identified. The proposed TS with this LAR separated the current spec 5.6.1.3 into two sections to separately address the Boraflex and the Boral BWR racks.

The criticality analysis that is the basis for this current spec is the original Westinghouse analysis for the BWR Boraflex racks, with an enrichment limit of 3.2 weight-percent. HNP initially proposed the addition of this enrichment limit of 3.2 weight-percent to the new TS 5.6.1.3.b merely as a clarification of the requirements for storage of fuel in the BWR Boral racks in SFP B. This original Westinghouse analysis would have been provided to the NRC for review prior to issuance of the initial operating license. If the NRC does not have access to this in their document repository, HNP will need to obtain it from Westinghouse (if available). This will require negotiation and issuance of a contract to Westinghouse.

c. TS Figure 5.6-3 should specify "Initial Maximum Planar Average Enrichment (wt% U-235)" instead of just "Enrichment." Having just "Enrichment" might be confused with assembly average enrichment.

Response: The x-axis title for Tech Spec figure 5.6-3 will be changed to incorporate this recommendation.

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Request 28:

Confirm that HNP currently only stores fuel depleted at Harris, Robinson, or Brunswick. Also, discuss the licensing approach to be used if, in the future, HNP decides to restart the storage of fuel from other facilities, which may include fuel designs not covered by this license amendment request.

Response:

HNP currently only stores fuel depleted at Harris, Robinson and Brunswick nuclear plants. Since the license for the cask used for trans-shipping fuel has expired, the population of fuel from Robinson and Brunswick stored at HNP will not increase above the current inventory. Additionally, Robinson and Brunswick are implementing dry spent fuel storage. There are no plans to restart the transshipment of spent fuel from Robinson or Brunswick nuclear plants to HNP.

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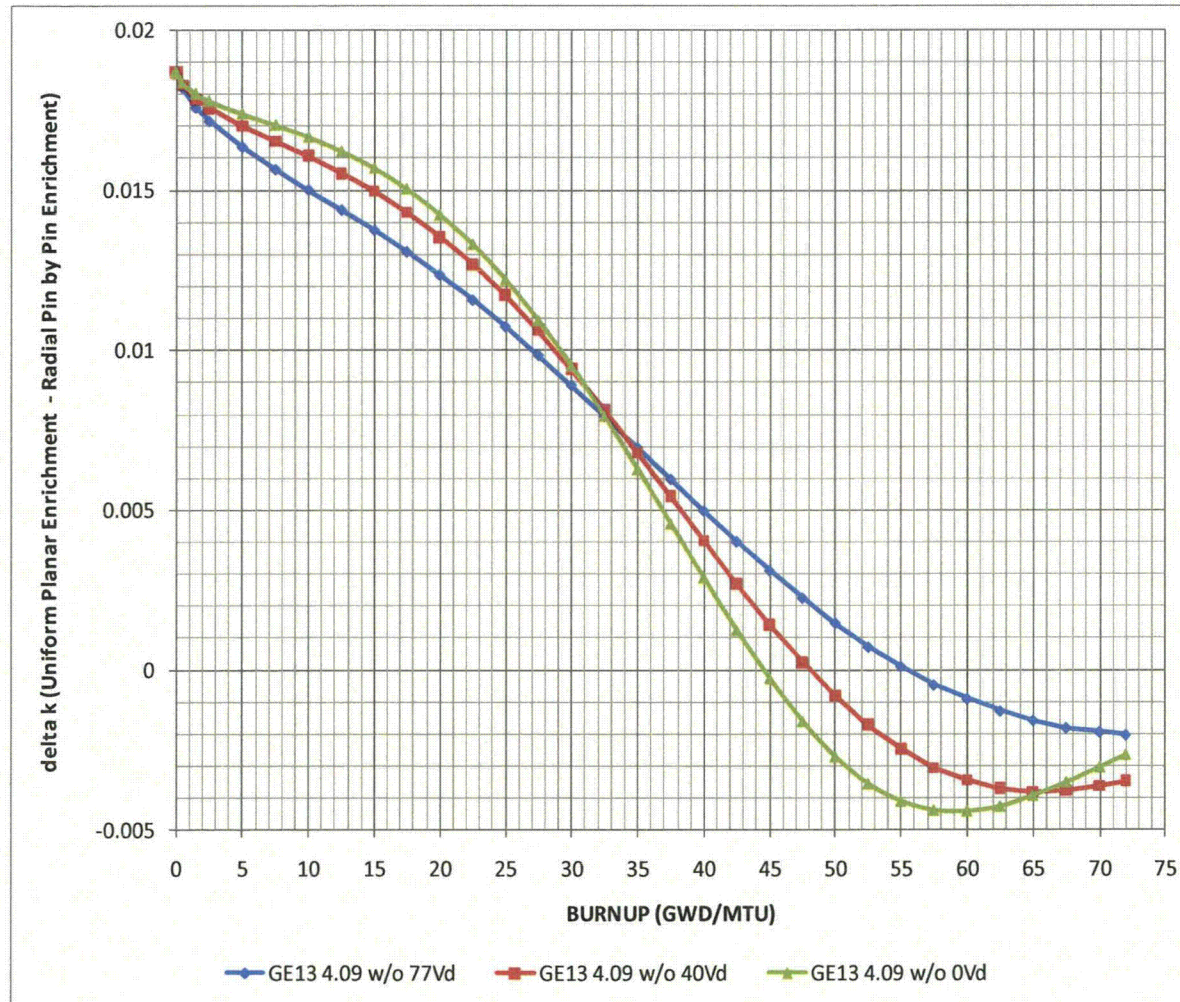ATTACHMENT – TABLES AND FIGURES
(10 Pages)

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Table 1: Calculations performed for the temperature coefficient (Response 1.b)

Cooling Time			4.0											
Tolerance			32		39.2		80.33		150		254		254+10% Void	
Burnup	Enrichment	Description	g	□k	h	□k	i	□k	z	□k	j	□k	k	□k
0.0	1.5	Full Rods	0.97352	-0.0023	0.97381	-0.0020	0.97579	0.0000	0.97972	0.0039	0.98627	0.0105	0.9897	0.0139
5.0	1.5	Full Rods	0.93423	-0.0033	0.93467	-0.0029	0.93754	0.0000	0.94303	0.0055	0.95258	0.0150	0.95483	0.0173
10.0	2.0	Full Rods	0.96972	-0.0038	0.97025	-0.0033	0.97352	0.0000	0.97941	0.0059	0.98912	0.0156	0.98956	0.0160
15.0	2.0	Full Rods	0.93965	-0.0043	0.94025	-0.0037	0.94392	0.0000	0.95042	0.0065	0.9611	0.0172	0.9609	0.0170
20.0	2.5	Full Rods	0.96061	-0.0043	0.96123	-0.0037	0.96495	0.0000	0.97132	0.0064	0.98146	0.0165	0.98009	0.0151
25.0	2.5	Full Rods	0.9246	-0.0046	0.92527	-0.0040	0.92922	0.0000	0.93602	0.0068	0.94679	0.0176	0.94517	0.0159
25.0	3.0	Full Rods	0.9722	-0.0044	0.9728	-0.0038	0.97656	0.0000	0.98285	0.0063	0.99252	0.0160	0.99026	0.0137
30.0	3.0	Full Rods	0.93993	-0.0048	0.94063	-0.0041	0.94471	0.0000	0.9515	0.0068	0.96195	0.0172	0.9595	0.0148
35.0	3.5	Full Rods	0.95238	-0.0048	0.95308	-0.0041	0.95718	0.0000	0.96384	0.0067	0.97387	0.0167	0.97069	0.0135
40.0	3.5	Full Rods	0.92565	-0.0050	0.92639	-0.0043	0.93067	0.0000	0.93763	0.0070	0.94814	0.0175	0.94489	0.0142
40.0	4.0	Full Rods	0.96044	-0.0048	0.96115	-0.0041	0.96524	0.0000	0.97179	0.0066	0.98144	0.0162	0.97766	0.0124
45.0	4.0	Full Rods	0.93107	-0.0049	0.9318	-0.0041	0.93592	0.0000	0.94257	0.0067	0.9525	0.0166	0.94871	0.0128
50.0	4.6	Full Rods	0.93936	-0.0048	0.94007	-0.0041	0.94414	0.0000	0.95059	0.0065	0.96002	0.0159	0.95572	0.0116
55.0	4.6	Full Rods	0.91262	-0.0050	0.91336	-0.0042	0.91758	0.0000	0.92429	0.0067	0.93419	0.0166	0.92991	0.0123
0.0	1.5	Part Rods	0.9566	-0.0019	0.95681	-0.0017	0.95847	0.0000	0.96218	0.0037	0.96887	0.0104	0.9742	0.0157
5.0	1.5	Part Rods	0.91433	-0.0031	0.91471	-0.0027	0.91738	0.0000	0.92284	0.0055	0.9329	0.0155	0.93706	0.0197
10.0	2.0	Part Rods	0.95178	-0.0036	0.95226	-0.0031	0.95539	0.0000	0.96135	0.0060	0.97177	0.0164	0.97413	0.0187
15.0	2.0	Part Rods	0.91876	-0.0042	0.91933	-0.0036	0.92295	0.0000	0.92969	0.0067	0.94135	0.0184	0.94308	0.0201
20.0	2.5	Part Rods	0.94207	-0.0043	0.94266	-0.0037	0.94636	0.0000	0.95302	0.0067	0.96419	0.0178	0.96471	0.0184
25.0	2.5	Part Rods	0.9016	-0.0046	0.90225	-0.0040	0.90624	0.0000	0.91338	0.0071	0.92535	0.0191	0.92563	0.0194
25.0	3.0	Part Rods	0.95555	-0.0044	0.95616	-0.0038	0.95991	0.0000	0.96649	0.0066	0.97724	0.0173	0.97681	0.0169
30.0	3.0	Part Rods	0.91857	-0.0048	0.91925	-0.0041	0.92335	0.0000	0.93049	0.0071	0.94211	0.0188	0.94154	0.0182
35.0	3.5	Part Rods	0.93351	-0.0048	0.9342	-0.0041	0.93832	0.0000	0.94537	0.0071	0.95661	0.0183	0.95523	0.0169
40.0	3.5	Part Rods	0.9034	-0.0051	0.90413	-0.0044	0.90849	0.0000	0.91591	0.0074	0.92774	0.0193	0.92627	0.0178
40.0	4.0	Part Rods	0.94393	-0.0048	0.94463	-0.0041	0.94876	0.0000	0.95571	0.0069	0.96657	0.0178	0.96451	0.0157
45.0	4.0	Part Rods	0.91192	-0.0049	0.91263	-0.0042	0.91686	0.0000	0.92398	0.0071	0.93522	0.0184	0.93316	0.0163
50.0	4.6	Part Rods	0.92264	-0.0049	0.92334	-0.0042	0.9275	0.0000	0.93441	0.0069	0.94514	0.0176	0.94248	0.0150
55.0	4.6	Part Rods	0.89177	-0.0051	0.89251	-0.0043	0.89684	0.0000	0.90406	0.0072	0.91532	0.0185	0.91271	0.0159

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Figure 1: The reactivity impact of the modeling assumption (Response 4.c)

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Figure 2: In-rack CASMO-4 calculation results for 0 percent void history (Response 7)

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Figure 3: In-rack CASMO-4 calculation results for 40 percent void history (Response 7)

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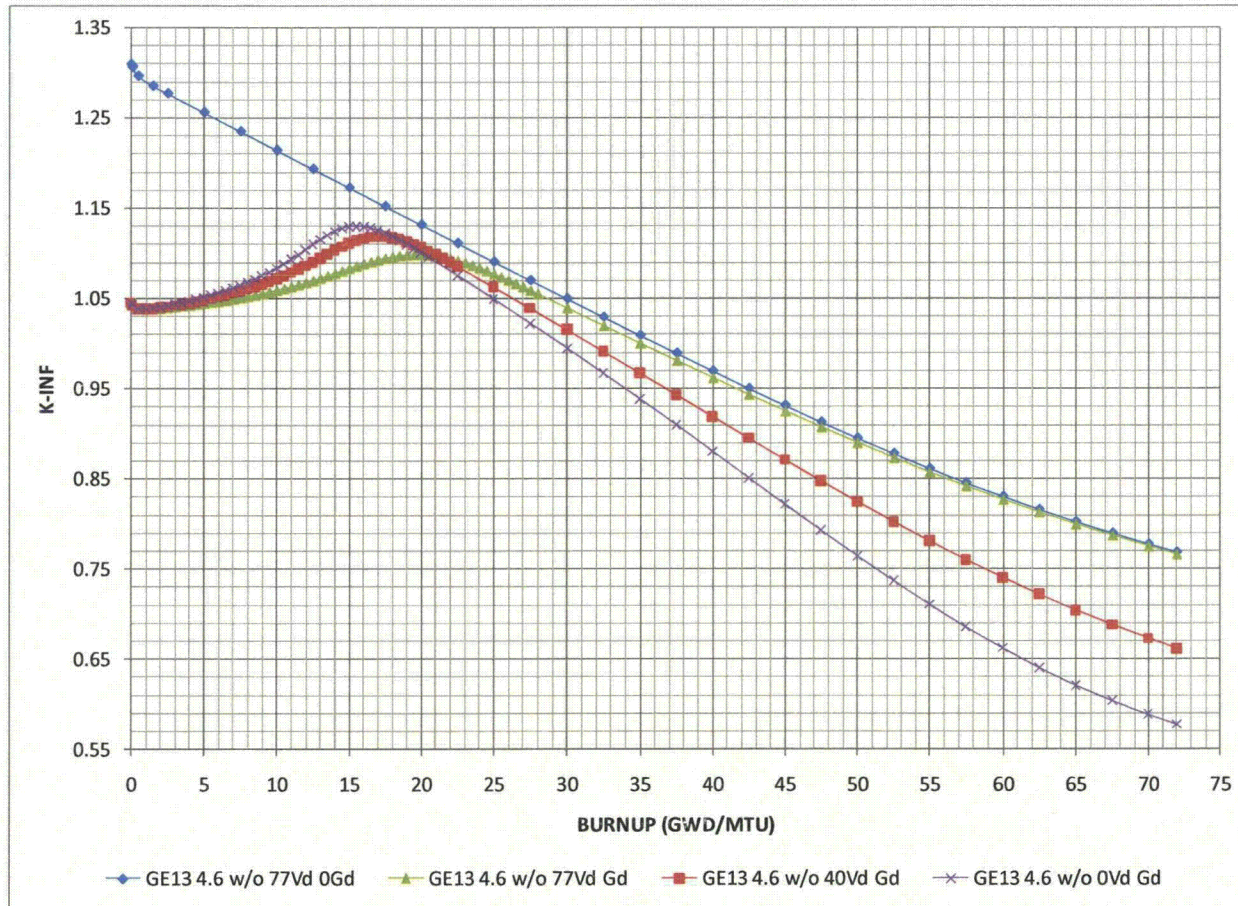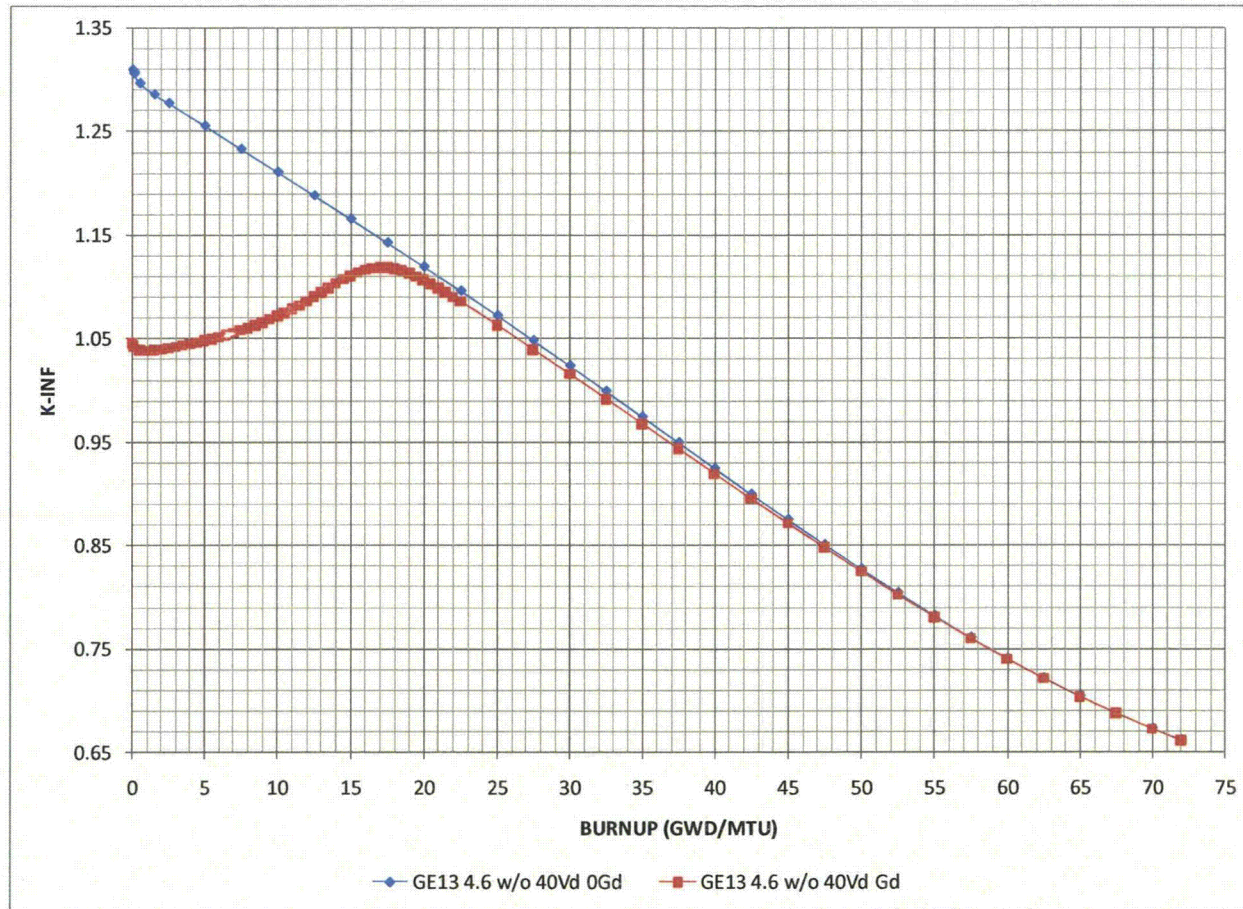Figure 4: In-rack CASMO-4 calculation results for 77 percent void history (Response 7)

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Table 2 - Design Basis Fuel Assembly Calculations for 2 wt% (Response 8.a)

Burnup [GWD/MTU]	GE3 4 YEARS	GE3 26 YEARS	GE4 4 YEARS	GE4 23 YEARS	GE7 4 YEARS	GE7 12 YEARS	GE8 4 YEARS	GE9 4 YEARS	GE10 4 YEARS	GE13 4 YEARS
Filename	HS-GE3-20	HS-GE3-20-26	HS-GE4-20	HS-GE4-20-23	HS-GE7-20	HS-GE7-20-12	HS-GE8-20	HS-GE9-20	HS-GE10-20	HS-GE13-20
0.0	1.0822	1.0822	1.0763	1.0762	1.0754	1.0752	1.0726	1.0715	1.0583	1.0728
0.1	1.0776	1.0777	1.0718	1.0717	1.0708	1.0707	1.0681	1.0670	1.0538	1.0683
0.5	1.0696	1.0697	1.0638	1.0638	1.0628	1.0627	1.0600	1.0589	1.0458	1.0603
1.5	1.0604	1.0604	1.0547	1.0546	1.0535	1.0534	1.0507	1.0497	1.0365	1.0511
2.5	1.0498	1.0494	1.0442	1.0437	1.0430	1.0426	1.0401	1.0391	1.0259	1.0407
5.0	1.0195	1.0160	1.0140	1.0107	1.0125	1.0106	1.0096	1.0086	0.9955	1.0105
7.5	0.9880	0.9794	0.9827	0.9746	0.9806	0.9762	0.9775	0.9764	0.9634	0.9788
10.0	0.9563	0.9415	0.9510	0.9373	0.9482	0.9407	0.9447	0.9436	0.9307	0.9466
12.5	0.9247	0.9032	0.9197	0.8997	0.9159	0.9051	0.9119	0.9107	0.8979	0.9144
15.0	0.8938	0.8655	0.8890	0.8627	0.8842	0.8699	0.8794	0.8781	0.8654	0.8827
17.5	0.8638	0.8288	0.8593	0.8268	0.8533	0.8356	0.8476	0.8464	0.8337	0.8518
20.0	0.8352	0.7937	0.8309	0.7925	0.8237	0.8027	0.8171	0.8158	0.8032	0.8221
22.5	0.8082	0.7608	0.8041	0.7602	0.7957	0.7717	0.7881	0.7868	0.7743	0.7939
25.0	0.7831	0.7302	0.7793	0.7304	0.7697	0.7428	0.7611	0.7597	0.7473	0.7677
27.5	0.7600	0.7023	0.7565	0.7031	0.7458	0.7164	0.7363	0.7349	0.7226	0.7435
30.0	0.7392	0.6772	0.7359	0.6785	0.7243	0.6926	0.7139	0.7124	0.7004	0.7216
32.5	0.7206	0.6549	0.7175	0.6567	0.7050	0.6714	0.6939	0.6925	0.6806	0.7020
35.0	0.7042	0.6353	0.7011	0.6374	0.6880	0.6528	0.6764	0.6750	0.6632	0.6848
37.5	0.6898	0.6182	0.6868	0.6206	0.6733	0.6366	0.6613	0.6598	0.6483	0.6698
40.0	0.6774	0.6035	0.6745	0.6061	0.6606	0.6227	0.6483	0.6468	0.6355	0.6569
42.5	0.6667	0.5908	0.6637	0.5937	0.6496	0.6108	0.6372	0.6358	0.6247	0.6459
45.0	0.6575	0.5801	0.6546	0.5830	0.6404	0.6007	0.6279	0.6265	0.6155	0.6365

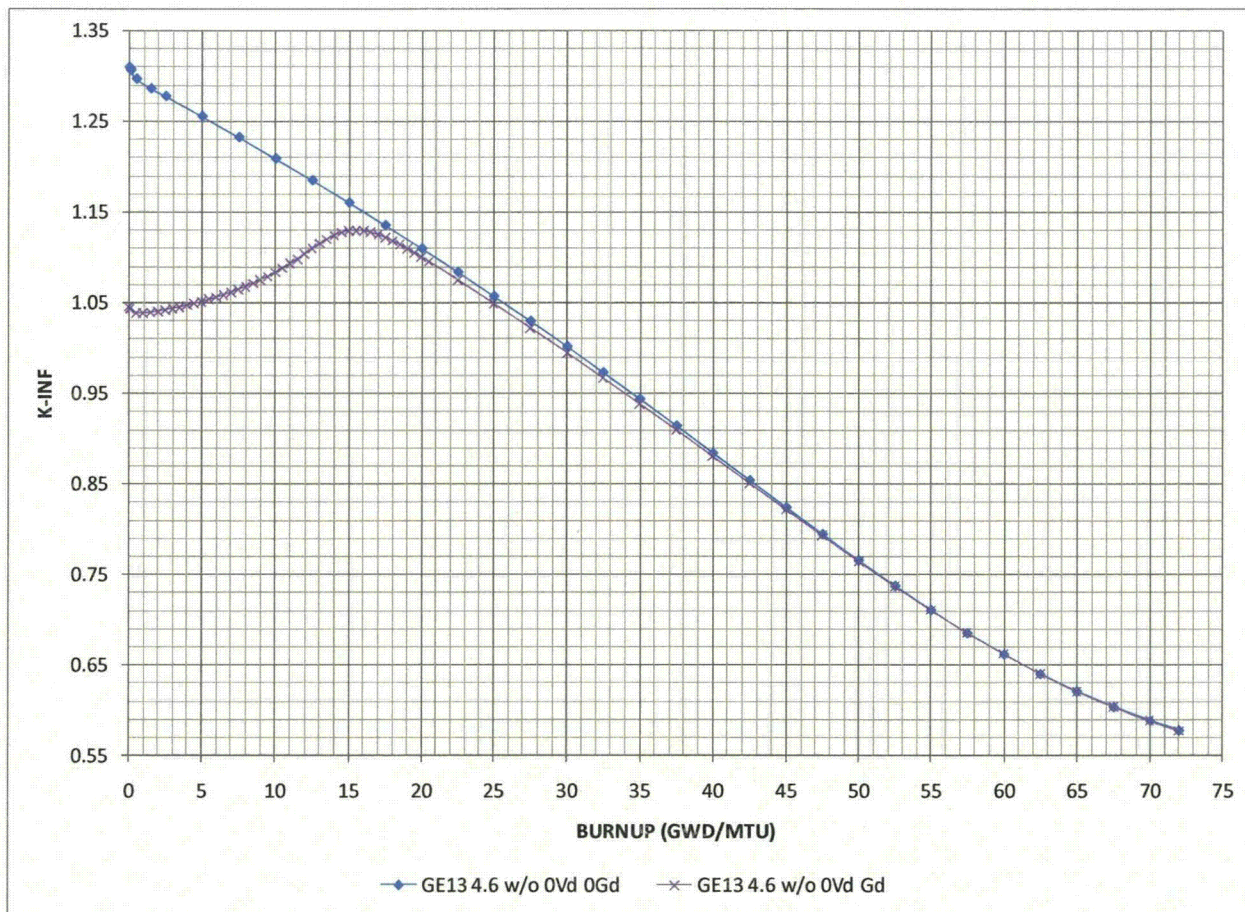SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
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Table 3 - Design Basis Fuel Assembly Calculations for 3 wt% (Response 8.a)

Burnup [GWD/MTU]	GE3 4 YEARS	GE3 26 YEARS	GE4 4 YEARS	GE4 23 YEARS	GE7 4 YEARS	GE7 12 YEARS	GE8 4 YEARS	GE9 4 YEARS	GE10 4 YEARS	GE13 4 YEARS
Filename	HS-GE3-30	HS-GE3-30-26	HS-GE4-30	HS-GE4-30-23	HS-GE7-30	HS-GE7-30-12	HS-GE8-30	HS-GE9-30	HS-GE10-30	HS-GE13-30
0.0	1.2030	1.2029	1.1972	1.1970	1.1981	1.1980	1.1965	1.1956	1.1800	1.1950
0.1	1.1988	1.1987	1.1930	1.1929	1.1939	1.1938	1.1923	1.1915	1.1759	1.1909
0.5	1.1893	1.1893	1.1835	1.1835	1.1845	1.1844	1.1829	1.1821	1.1666	1.1815
1.5	1.1788	1.1789	1.1731	1.1732	1.1741	1.1740	1.1726	1.1718	1.1564	1.1713
2.5	1.1687	1.1687	1.1631	1.1630	1.1641	1.1639	1.1627	1.1619	1.1465	1.1614
5.0	1.1415	1.1402	1.1360	1.1347	1.1370	1.1363	1.1359	1.1351	1.1199	1.1347
7.5	1.1135	1.1095	1.1080	1.1043	1.1090	1.1070	1.1081	1.1072	1.0922	1.1070
10.0	1.0853	1.0776	1.0797	1.0726	1.0805	1.0767	1.0796	1.0787	1.0639	1.0788
12.5	1.0569	1.0448	1.0513	1.0402	1.0517	1.0457	1.0507	1.0499	1.0351	1.0504
15.0	1.0283	1.0115	1.0229	1.0073	1.0228	1.0143	1.0215	1.0206	1.0060	1.0217
17.5	0.9995	0.9776	0.9944	0.9741	0.9936	0.9826	0.9920	0.9911	0.9766	0.9928
20.0	0.9710	0.9440	0.9660	0.9409	0.9644	0.9508	0.9623	0.9614	0.9470	0.9638
22.5	0.9427	0.9105	0.9378	0.9080	0.9353	0.9191	0.9326	0.9316	0.9173	0.9348
25.0	0.9147	0.8774	0.9101	0.8755	0.9065	0.8876	0.9031	0.9020	0.8878	0.9061
27.5	0.8873	0.8450	0.8829	0.8438	0.8783	0.8568	0.8739	0.8728	0.8587	0.8778
30.0	0.8607	0.8137	0.8566	0.8131	0.8507	0.8268	0.8454	0.8442	0.8303	0.8501
32.5	0.8351	0.7836	0.8313	0.7836	0.8242	0.7979	0.8178	0.8165	0.8027	0.8233
35.0	0.8107	0.7550	0.8071	0.7556	0.7988	0.7702	0.7913	0.7900	0.7763	0.7976
37.5	0.7877	0.7282	0.7843	0.7293	0.7748	0.7442	0.7661	0.7648	0.7512	0.7732
40.0	0.7663	0.7032	0.7631	0.7048	0.7523	0.7199	0.7426	0.7412	0.7279	0.7504
42.5	0.7464	0.6802	0.7434	0.6822	0.7315	0.6975	0.7209	0.7195	0.7063	0.7292
45.0	0.7283	0.6593	0.7253	0.6616	0.7125	0.6770	0.7010	0.6996	0.6867	0.7097

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Table 4 - Design Basis Fuel Assembly Calculations for 4 wt% (Response 8.a)

| Burnup [GWD/MTU]	GE3 4 YEARS	GE3 26 YEARS	GE4 4 YEARS	GE4 23 YEARS	GE7 4 YEARS	GE7 12 YEARS	GE8 4 YEARS	GE9 4 YEARS	GE10 4 YEARS	GE13 4 YEARS
Filename	HS-GE3-40	HS-GE3-40-26	HS-GE4-40	HS-GE4-40-23	HS-GE7-40	HS-GE7-40-12	HS-GE8-40	HS-GE9-40	HS-GE10-40	HS-GE13-40
0.0	1.2760	1.2759	1.2703	1.2702	1.2725	1.2724	1.2717	1.2709	1.2539	1.2691
0.1	1.2723	1.2722	1.2666	1.2665	1.2688	1.2687	1.2680	1.2672	1.2503	1.2654
0.5	1.2628	1.2628	1.2572	1.2571	1.2594	1.2593	1.2586	1.2579	1.2411	1.2561
1.5	1.2521	1.2522	1.2466	1.2466	1.2489	1.2488	1.2483	1.2475	1.2308	1.2457
2.5	1.2431	1.2432	1.2375	1.2376	1.2399	1.2399	1.2395	1.2387	1.2221	1.2369
5.0	1.2194	1.2190	1.2139	1.2135	1.2166	1.2163	1.2164	1.2157	1.1992	1.2139
7.5	1.1952	1.1933	1.1897	1.1878	1.1925	1.1915	1.1927	1.1919	1.1757	1.1901
10.0	1.1708	1.1665	1.1652	1.1613	1.1681	1.1659	1.1685	1.1678	1.1517	1.1661
12.5	1.1463	1.1391	1.1407	1.1340	1.1434	1.1399	1.1440	1.1433	1.1273	1.1418
15.0	1.1216	1.1111	1.1160	1.1063	1.1186	1.1134	1.1193	1.1185	1.1027	1.1173
17.5	1.0968	1.0826	1.0912	1.0780	1.0935	1.0865	1.0943	1.0935	1.0778	1.0925
20.0	1.0716	1.0536	1.0662	1.0494	1.0683	1.0592	1.0688	1.0680	1.0525	1.0676
22.5	1.0465	1.0244	1.0410	1.0206	1.0427	1.0316	1.0431	1.0422	1.0268	1.0423
25.0	1.0212	0.9949	1.0159	0.9915	1.0169	1.0036	1.0170	1.0161	1.0008	1.0168
27.5	0.9957	0.9652	0.9906	0.9624	0.9909	0.9755	0.9907	0.9897	0.9745	0.9911
30.0	0.9702	0.9356	0.9653	0.9332	0.9648	0.9473	0.9641	0.9631	0.9480	0.9653
32.5	0.9448	0.9061	0.9402	0.9042	0.9388	0.9191	0.9374	0.9364	0.9213	0.9393
35.0	0.9197	0.8769	0.9152	0.8756	0.9129	0.8910	0.9107	0.9097	0.8947	0.9135
37.5	0.8949	0.8481	0.8907	0.8474	0.8873	0.8633	0.8842	0.8831	0.8682	0.8878
40.0	0.8706	0.8201	0.8667	0.8200	0.8621	0.8362	0.8581	0.8569	0.8421	0.8625
42.5	0.8471	0.7929	0.8434	0.7933	0.8376	0.8097	0.8324	0.8312	0.8165	0.8377
45.0	0.8243	0.7668	0.8209	0.7677	0.8138	0.7841	0.8075	0.8062	0.7916	0.8136

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Table 5 - Design Basis Fuel Assembly Calculations for 4.6 wt% (Response 8.a)

Burnup [GWD/MTU]	GE3 4 YEARS	GE3 26 YEARS	GE4 4 YEARS	GE4 23 YEARS	GE7 4 YEARS	GE7 12 YEARS	GE8 4 YEARS	GE9 4 YEARS	GE10 4 YEARS	GE13 4 YEARS
Filename	HS-GE3-46	HS-GE3-46-26	HS-GE4-46	HS-GE4-46-23	HS-GE7-46	HS-GE7-46-12	HS-GE8-46	HS-GE9-46	HS-GE10-46	HS-GE13-46
0.0	1.3076	1.3076	1.3020	1.3019	1.3048	1.3047	1.3043	1.3036	1.3016	1.3012
0.1	1.3041	1.3041	1.2986	1.2985	1.3013	1.3012	1.3009	1.3002	1.2982	1.2978
0.5	1.2948	1.2949	1.2894	1.2893	1.2921	1.2921	1.2917	1.2910	1.2891	1.2887
1.5	1.2841	1.2843	1.2787	1.2788	1.2816	1.2815	1.2813	1.2806	1.2787	1.2782
2.5	1.2755	1.2758	1.2702	1.2703	1.2732	1.2732	1.2731	1.2724	1.2705	1.2700
5.0	1.2536	1.2536	1.2483	1.2481	1.2516	1.2514	1.2519	1.2512	1.2493	1.2487
7.5	1.2313	1.2302	1.2260	1.2248	1.2295	1.2288	1.2301	1.2294	1.2275	1.2269
10.0	1.2088	1.2059	1.2035	1.2006	1.2071	1.2056	1.2080	1.2073	1.2054	1.2049
12.5	1.1863	1.1811	1.1809	1.1759	1.1846	1.1819	1.1857	1.1850	1.1830	1.1827
15.0	1.1637	1.1557	1.1582	1.1507	1.1619	1.1579	1.1632	1.1625	1.1604	1.1603
17.5	1.1409	1.1299	1.1354	1.1251	1.1390	1.1335	1.1405	1.1398	1.1376	1.1378
20.0	1.1180	1.1037	1.1125	1.0991	1.1159	1.1087	1.1174	1.1167	1.1144	1.1150
22.5	1.0948	1.0770	1.0894	1.0727	1.0925	1.0835	1.0941	1.0933	1.0908	1.0920
25.0	1.0714	1.0500	1.0660	1.0461	1.0688	1.0580	1.0703	1.0695	1.0669	1.0687
27.5	1.0478	1.0228	1.0426	1.0191	1.0449	1.0322	1.0462	1.0453	1.0426	1.0450
30.0	1.0240	0.9952	1.0189	0.9920	1.0207	1.0061	1.0217	1.0209	1.0179	1.0212
32.5	1.0001	0.9676	0.9951	0.9648	0.9963	0.9797	0.9970	0.9961	0.9929	0.9970
35.0	0.9760	0.9398	0.9713	0.9375	0.9718	0.9532	0.9720	0.9710	0.9676	0.9727
37.5	0.9521	0.9122	0.9476	0.9104	0.9472	0.9267	0.9468	0.9458	0.9422	0.9483
40.0	0.9283	0.8847	0.9241	0.8835	0.9227	0.9003	0.9216	0.9206	0.9167	0.9239
42.5	0.9047	0.8576	0.9008	0.8569	0.8984	0.8741	0.8964	0.8953	0.8912	0.8996
45.0	0.8816	0.8310	0.8779	0.8309	0.8744	0.8483	0.8714	0.8703	0.8659	0.8754

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Figure 5: In-rack reactivity calculation results from Request 7 shown on same plot as those that neglect Gd (Response 9)
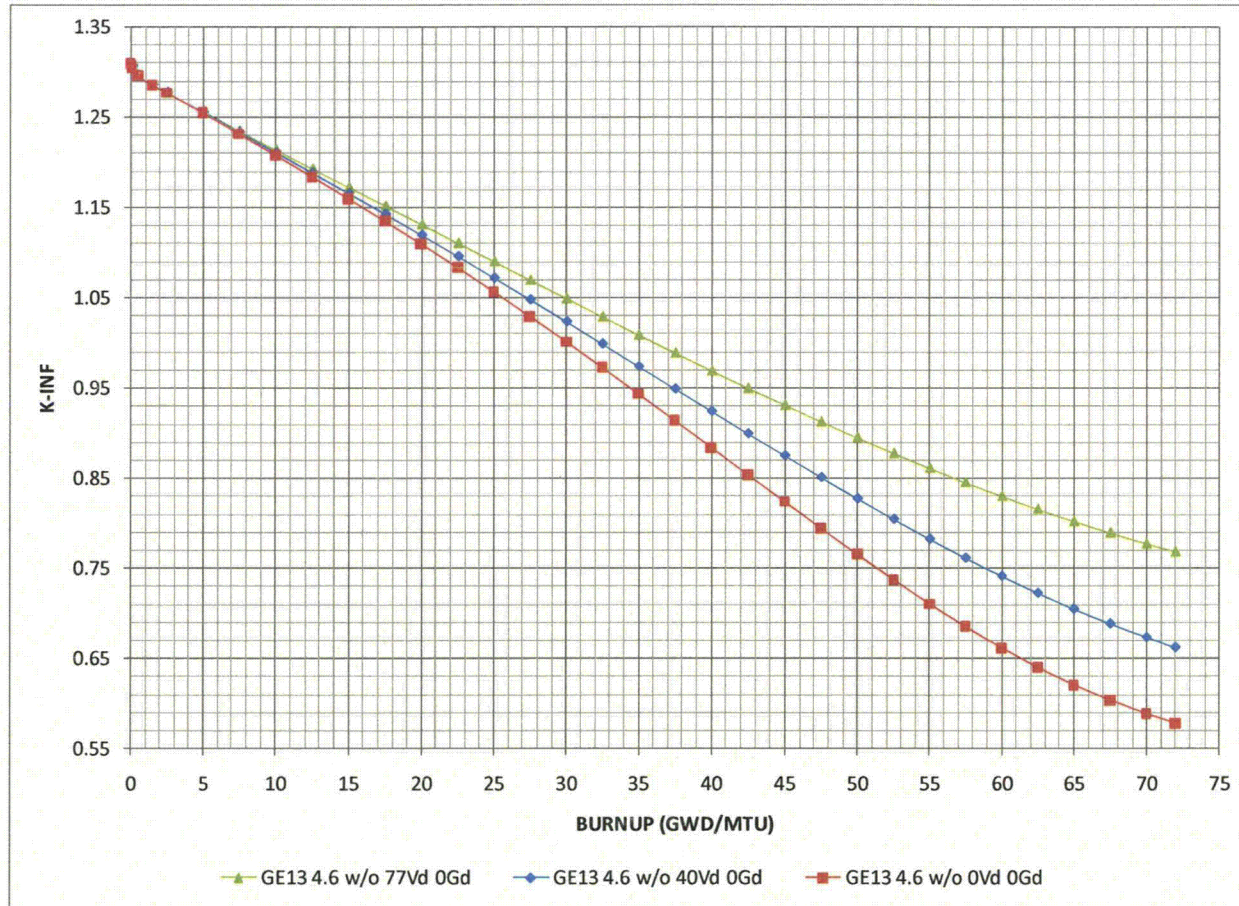
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RETYPED TECHNICAL SPECIFICATION PAGES
(4 Pages)

PLANT SYSTEMS

3/4.7.15 FUEL STORAGE

LIMITING CONDITION FOR OPERATION

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3.7.15 The combination of initial enrichment and burnup of each fuel assembly stored in the Fuel Handling Building storage racks shall meet the initial enrichment and burnup requirements of Specification 5.6 Fuel Storage.*

APPLICABILITY: Whenever any fuel assembly is stored in a fuel storage rack in the Fuel Handling Building.

ACTION:

   a. With the requirements of LCO 3.7.15 not met, immediately initiate action to move the non-complying fuel assembly to meet the requirements of Specification 3.7.15.

   b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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4.7.15.1 Prior to storing the fuel assembly in the Fuel Handling Building storage racks, verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with the requirements of Specification 5.6 for the intended storage rack and storage configuration.*

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* Burnup requirements are not applicable to BWR fuel assemblies stored in Boral storage racks in SFP "B" and "C".

5.6 Fuel Storage

CRITICALITY

5.6.1 3. BWR Storage Racks in Pools "A" and "B"

    a. BWR Boraflex storage racks.

        1. K_{eff} less than or equal to 0.95 if fully flooded with water borated to 2000 ppm.

        2. K_{eff} less than 1.0 if flooded with unborated water.

        3. Nominal 6.25 inch center-to-center spacing between fuel assemblies for fuel stored in the BWR Boraflex racks.

        4. BWR assembly storage requirements for BWR Boraflex Racks:

            i. BWR assemblies must have a minimum cooling time of 7 years.

            ii. BWR assemblies must have a maximum planar average initial enrichment of less than or equal to 4.6wt.% U^{235} and be within the "acceptable burnup domain" of the burnup restriction shown in Figure 5.6-3.

        5. BWR assemblies are acceptable for storage in BWR Boraflex storage racks provided the K_{inf} is less than or equal to 1.32 for the standard cold core geometry (SCCG).

    b. BWR Boral storage racks

        1. K_{eff} less than or equal to 0.95 when flooded with unborated water.

        2. The reactivity margin is assured for BWR Boral racks in pool "B" by maintaining a nominal 6.25 inch center-to-center distance in the BWR Boral storage racks.

        3. BWR assemblies are acceptable for storage in BWR Boral storage racks provided the maximum planar average enrichments are less than 3.2wt.% U^{235}.

   4. PWR and BWR racks in pools "C" and "D"

    a. k_{eff} less than or equal to 0.95 when flooded with unborated water.

    b. The reactivity margin is assured for pools "C" and "D" by maintaining a nominal 9.017 inch center-to-center distance between fuel assemblies placed in the non-flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks.

DESIGN FEATURES

5.6 Fuel Storage (continued)

4. PWR and BWR racks in pools "C" and "D" (continued)

    c. The following restrictions are also imposed through administrative controls:

        1. PWR assemblies must be within the "acceptable range" of the burnup restrictions shown in Figure 5.6-1 prior to storage in pools "C" and "D".

        2. BWR assemblies are acceptable for storage in pool "C" provided the maximum planar average enrichments are less than 4.6 wt.% U235 and K_{inf} is less than or equal to 1.32 for the standard cold core geometry (SCCG).

  5. In each case, k_{eff} includes allowances for uncertainties as described in Section 4.3.2.6 of the FSAR.

DRAINAGE

5.6.2 The pools "A", "B", "C" and "D" are designed and shall be maintained to prevent inadvertent draining of the pools below elevation 277.

CAPACITY

5.6.3.a Pool "A" contains six (6 x 10 cell) flux trap type PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. Pool "B" contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) flux trap style PWR racks and seventeen (11 x 11 cell) BWR racks and is licensed for one additional (11 x 11 cell) BWR rack that will be installed as needed. The combined pool "A" and "B" licensed storage capacity is 3669 assemblies.

The graph shows Burnup (MWD/MTU) on the y-axis versus Initial Maximum Planar Average Enrichment (wt% U-235) on the x-axis.

Acceptable Burnup Domain
(Minimum required cooling
time is 7 years)

$Burnup = 0.3850*E^3-3.8506*E^2+28.063*E-32.905$

Unacceptable Burnup Domain

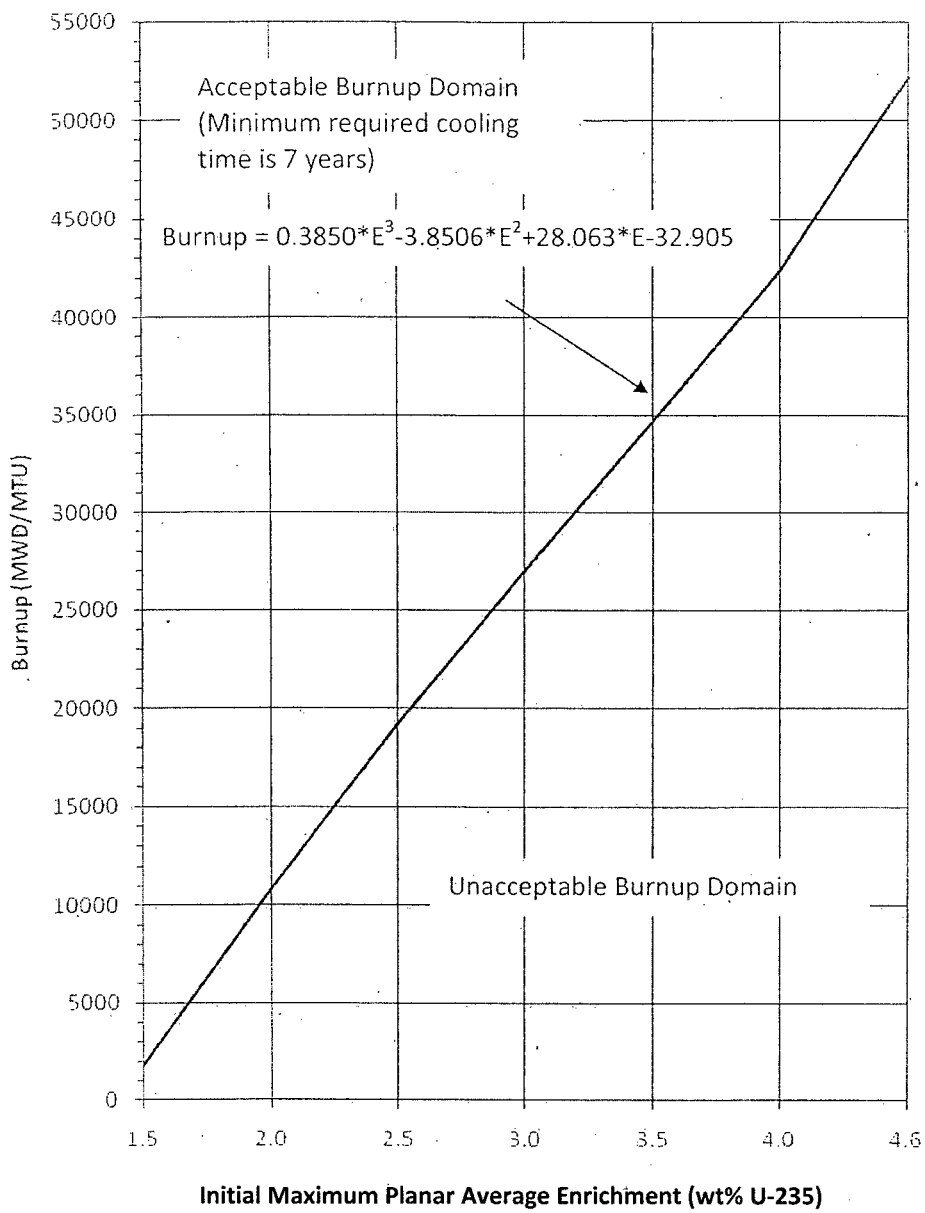**Initial Maximum Planar Average Enrichment (wt% U-235)**

FIGURE 5.6-3

POOLS "A" AND "B" BURNUP VERSUS ENRICHMENT FOR BWR BORAFLEX RACKS