Progress Energy

10 CFR 50.90

Serial: RNP-RA/03-0109

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

SUPPLEMENT TO AMENDMENT REQUEST REGARDING
CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON (TAC NO. MB9148)

Ladies and Gentlemen:

By letter dated May 28, 2003, Progress Energy Carolinas, Inc., submitted a request for Technical Specifications change regarding reactivity credit for spent fuel storage pool dissolved boron for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

Requests for Additional Information (RAIs) related to this change were received from the NRC in faxed correspondences dated July 18, 2003 and August 11, 2003. Based on a conference call on August 14, 2003 between NRC and HBRSEP, Unit No. 2, personnel, three of the questions from the August 11, 2003 fax were eliminated. Additionally, one other question required rewording and that reworded question was received by electronic mail on August 21, 2003.

Attachment I provides an Affirmation pursuant to 10 CFR 50.30(b).

Attachment II provides the responses to the RAIs. The responses do not impact the proposed Technical Specifications, No Significant Hazards Consideration Determination, or Environmental Impact Consideration provided in the May 28, 2003 submittal.

In accordance with 10 CFR 50.91(b), the State of South Carolina is being provided a copy of this letter.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

Jan Lucas

Jan F. Lucas
Manager - Support Services - Nuclear

**Progress Energy Carolinas, Inc.**
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A001

Attachments:

I. Affirmation

II. Responses to NRC Requests for Additional Information

RAC/rac

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspectors, HBRSEP
Attorney General (SC)

## AFFIRMATION

The information contained in letter RNP-RA/03-0109 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: Oct 8, 2003

C L Burton

C. L. Burton
Director – Site Operations
HBRSEP, Unit No. 2

## H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

## RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION

### Question 1

You show manufacturing tolerances in Tables 6.5 and 6.6 as ±0.0094 for both the spent fuel and new fuel. How do you account for the uncertainties in the $k_{eff}$ calculation caused by changes to the pellet and cladding geometries during burnup? Please provide a list of all tolerances not included in your $k_{eff}$ calculation, a justification for why they were not considered, and $\Delta k$ values for their contribution to the overall $k_{eff}$ calculation.

### Response 1

The changes in the fuel pellet and cladding geometries as a result of burnup are extremely small and their reactivity effect is negligible. These changes are smaller than the original manufacturing tolerances which also have a negligible effect on reactivity. This conclusion is re-enforced by the analyses of manufacturing tolerances described below.

The analyses of tolerance effects have neglected certain tolerances that historically in many storage rack evaluations have been found to be negligible. However, in response to the Requests for Additional Information (RAIs), these neglected tolerances are listed below and their effect on the reactivity of the racks evaluated.

| [redacted] | Reactivity Effect, $\Delta k$ |
|---|---|
| Fuel Pellet Outside Diameter (O.D.) | ± 0.00002 |
| Clad Inside Diameter (I.D.) | ± 0.00001 |
| Clad O.D. | ± 0.00003 |
| Guide Tube I.D. | ± 0.00003 |
| Guide Tube O.D. | ± 0.00003 |
| Fuel Rod Pitch | Pitch tolerance is limited by overall fuel assembly spacing required to fit into the core and to meet other operating conditions. The tolerance in rod pitch would necessarily be very small. Because of the small average pitch variation possible, the reactivity effect would be negligible. |
| $[\Sigma (tol)^2]^{1/2}$ | ± 0.00006 |

The net effect of these tolerances (±0.00006 Δk) is negligible and would entirely disappear when combined statistically with the more significant tolerance uncertainties. For the checkerboard loading, the tolerance uncertainties were not calculated, but can be assumed to be comparable to those listed above and therefore equally negligible.

The reactivity impact of each listed manufacturing tolerance is shown. The value of each tolerance is not provided because it is proprietary to Framatome ANP, Inc.; however, all tolerances are less than 0.003 inches.

Question 2

For Table 6.5, you indicate an MCNP4a statistical uncertainty at 4.95% enrichment with a fuel burnup of 34,752 MWD/MTU of ±0.0007. In Table 6.6, on the other hand, you use a calculational statistical uncertainty of ±0.0005 for fuel of 4.95% enrichment. Please describe the difference between the methods used to calculate the $k_{eff}$ values for the spent and fresh fuel and account for the difference between the statistical uncertainties. Additionally, please describe why the uncertainties for temperature to 171°F are different for Tables 6.5 and 6.6.

Response 2

The differences between the MCNP4a statistical uncertainties stated for these two cases are due to the random nature of the Monte Carlo method. These two calculations use the same original enrichment, but are modeling different isotopic mixtures (Table 6.5 refers to spent fuel and Table 6.6 refers to fresh fuel) and different storage geometries (Table 6.5 refers to unrestricted storage and Table 6.6 refers to checkerboard storage). These differences influence the convergence of the Monte Carlo confidence interval for $k_{eff}$ and therefore cause the two cases to have slightly different statistical uncertainties.

Question 3

In Fig. 1-1 and 1-1a, you provide a chart listing acceptable burnups for unrestricted storage of spent fuel. How do you plan to store spent fuel that does not meet the minimum burnup requirements? Additionally, please describe the methods that will be in place, either administratively or experimentally, to independently confirm the fuel burnup before the fuel is placed in the storage racks.

Response 3

Fuel that does not meet the minimum burnup (specific to its original enrichment) for unrestricted storage will be stored in restricted storage. Section 1 of the criticality analysis, provided as Attachment VII to the May 28, 2003 letter, states that in any location, fuel of a lower reactivity may be used in lieu of the fuel otherwise specified.

Fresh fuel with enrichments less than 4.95±0.05%, or spent fuel of any burnup, may be used in lieu of the fresh 4.95% (nominal) enriched fuel.

The burnup for a given fuel assembly is taken from the Special Nuclear Material (SNM) database, which is updated based on the core monitoring system. Appropriate uncertainty penalties are applied to account for the uncertainty in exposure records.

## Question 4

In Figures 1.1 and 1.1a, you provide data for the minimum burnup as a function of enrichment. Please provide information describing the methodology used to calculate these limits. Additionally, please demonstrate that the data presented represent the most bounding or limiting condition.

## Response 4

Figure 1.1 and the line in Figure 1.1a were generated by plotting the data shown in Table 6.5 and fitting a linear equation using a least squares fit. The resulting line was then modified to insure that all data points from Table 6.5 are confirmed to be conservatively bounded. The data in Table 6.5 are conservative estimates based on specific calculations of the bounding cases, conservatively including the maximum credible uncertainties.

## Question 5

The submittal does not discuss the interface between the fresh fuel and the spent fuel in the various storage locations. Please review the worst case reactivity conditions that could result from this interface and either describe why this condition is bounded by current analyses or submit calculations which demonstrate that the requirements of 10 CFR 50.68 will be satisfied.

## Response 5

In all cases, the interfaces between racks provide spacing between fuel assemblies in storage that is very large (15.75 inches between fuel centerlines in the new high-density racks and the fuel centerlines in the existing low-density racks, and 12 inches between centerlines of assemblies in adjacent high-density racks). These large spacings are more than adequate to provide neutronic isolation between racks. For mixed storage of fresh (checkerboard) and spent fuel in the same rack module, the May 28, 2003 submittal specifies a row of empty (water-filled) cells between the two arrays.

## Question 6

In the submittal, you determine the maximum effective multiplication factors by statistically combining $k_{eff}$ uncertainties. However, the submittal does not contain the equation used to perform this combination. Please provide a detailed description of the statistical method employed and an example of your implementation of this method.

## Response 6

An equation for the combination of individual uncertainties may be written as:

$$\text{Statistical sum} = [\Sigma_i (\Delta k_i)^2]^{1/2}$$

Where $\Delta k_i$ are the reactivity effects associated with the independent tolerances (see also ANS/ANSI Standard 8-17 (1997), "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors"). In practice, each tolerance effect ($\Delta k$) is separately evaluated (as permitted in the letter from B. K. Grimes to All Power Reactors, dated April 14, 1978, and the memorandum "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," USNRC Internal Memorandum, L. Kopp to Timothy Collins, dated August 19,1998) and each $\Delta k$ value is squared and added to the other $(\Delta k)^2$ terms. The square root of the summed $(\Delta k)^2$ values is taken as the total statistical uncertainty. This is the conventional method of statistically combining uncertainties.

## Question 7

In this submittal, you use a Westinghouse 15x15 fuel assembly and an Advanced Framatome-ANP 15x15 fuel assembly as bounding assemblies. What processes do you have in place to ensure that your calculations remain bounding for current and future core reload fuel types?

## Response 7

The core reload process used at Progress Energy Carolinas, Inc., requires that any such constraints be identified in the criticality analyses, accident analyses, peaking limits, etc. Any future fuel changes would require these analyses to be verified to insure that the analyses remain bounding. If it is determined that the analyses would not bound all aspects of a new fuel design, then a new analysis will be prepared and submitted, if required.

## Question 8

On page 1, you describe limitations of the MCNP4a calculations which prevent modeling certain fission product cross-sections in the criticality analyses. You then state that you model an equivalent Boron-10 concentration to compensate for these limitations. Please demonstrate that this methodology is conservative and provides bounding results.

Response 8

Neither MCNP4a nor NITAWL-KENO5a have all of the fission product cross-sections in their applicable libraries. CASMO4 tracks the concentrations of the most important 49 actinide and fission product nuclides. Fission product nuclides that are not tracked in CASMO4 are collected together and described by two pseudo-fission product nuclides, called LFP1 and LFP2. In addition to the two pseudo-fission products, MCNP4a and KENO5a do not have the following six nuclides in their libraries:

| | | |
|---|---|---|
| U-239 | Np-239 | Ba-140 |
| La-140 | Pm-148m | Eu-148 |

Of these six nuclides, only Pm-148m is significant, and for conservatism the remaining five nuclides (together with Xe-135) are set to zero concentration. There are three nuclides that have no cross-section libraries in either MCNP4a or KENO5a, Pm-148m and the two pseudo-fission products. For the past several years, in many licensing applications reviewed and accepted by the NRC, it has been standard practice to calculate an equivalent Boron-10 concentration to compensate for the absence of cross-sections for those nuclides in the MCNP4a and KENO5a libraries.

Question 9

In the amendment request letter, you indicated that a temperature of 171°F was assumed for a criticality calculation. What is the maximum bulk pool temperature at a full core off-load during a refueling outage? If the temperature exceeds 150°F, provide technical justifications for exceeding a gross temperature of 150°F in accordance with the guidance in the ACI Code 349 for long term operation.

Response 9

The H. B. Robinson Steam Electric Plant, Unit No. 2, Technical Requirements Manual (TRM) limits the Spent Fuel Pool (SFP) to a maximum of 150 °F. If the temperature exceeds 150 °F, the TRM requires fuel to be moved back into the containment. Calculation RNP-M/MECH-1646 was performed to determine the effects of increasing Service Water temperature on the Component Cooling Water System. The calculation assumed 1 and 1/3 cores were discharged into the SFP with no actions taken to limit temperature. The calculation resulted in a maximum SFP temperature of 170.4 °F. The criticality analysis used 171 °F for conservatism. This analysis was not done to allow SFP temperatures to exceed 150 °F; it only used the higher temperature to provide a conservative limit. Therefore, an evaluation in accordance with ACI Code 349 is not warranted.

Question 10

Lateral motion of the storage racks under postulated seismic conditions could potentially alter the spacing between racks. Indicate whether you have performed a structural rack dynamic analysis to calculate the required spacing between racks for your criticality calculations. If you have performed, discuss the methodologies and assumptions used for the analysis, and provide the results of the analysis. If you have not performed, explain the reasons why you don't need to perform an analysis.

Response 10

No specific rack dynamic analysis was performed for this criticality analysis. The expected motion of the racks during a seismic event is only 0.3 inches. In all cases, the interfaces between racks that provide spacing between fuel assemblies in storage are very large (15.75 inches between fuel centerlines in the new high-density racks and the fuel centerlines in the existing low-density racks, and 12 inches between centerlines of assemblies in adjacent high-density racks). A postulated seismic condition is an accident condition for which credit for the soluble boron in the SFP water is allowed. In the unlikely event that a seismic event could alter the spacing between racks, the soluble boron would preclude any criticality concern, and the very large water-gaps present would safely accommodate any reasonable seismic-induced motion without approaching spacing important for criticality safety.