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SUBJECT: Forwards response to NRC 990429 RAI re license amend request to place spent fuel pools C & D in service,dtd 981223.Info does not change initial determination that proposed license amend represents no significant hazards consideration.

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United States Nuclear Regulatory Commission
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SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE
HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Dear Sir or Madam:

By letter dated April 29, 1999, the NRC issued a request for additional information (RAI) regarding the Harris Nuclear Plant (HNP) license amendment request, submitted by CP&L letter Serial: HNP-98-188, dated December 23, 1998, to place spent fuel pools C and D in service. The HNP response to the NRC RAI is enclosed. The enclosed information is provided as a supplement to our December 23, 1998 license amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

Donna B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

A001

KWS/kws

Enclosure

c:

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SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Requested Item 1

Although the burnup criteria for storage in Pools C or D will be implemented by administrative procedures to ensure verified burnup prior to fuel transfer into these pools, an administrative failure should be assumed and evaluation of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per Technical Specifications (TS) Figure 5.6.1) should be analyzed.

Response to Requested Item 1

The presence of soluble boron in the spent fuel pool water will assure that the reactivity is maintained substantially less than the design limitation in the event of a misloading event as described above. The Double Contingency Principle provides that neither the utility nor the staff is required to assume two unlikely, independent, concurrent events. Therefore, a failure of the administrative controls related to fuel assembly placement and the inadvertent dilution of the spent fuel pool water need not be considered to occur simultaneously. As a result, credit for the presence of soluble boron in the spent fuel pool water may be taken for an assembly misloading event as described. A minimum spent fuel pool boron concentration of 2000 ppm is maintained in accordance with HNP chemistry procedure CRC-001. This minimum boron concentration is more than adequate to offset the reactivity addition from a postulated fuel assembly misloading event. Based on analysis performed by Holtec International, it has been determined that a soluble boron concentration of 400 ppm would be sufficient to maintain k_{eff} less than 0.95 in the event of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1).

Requested Item 2

How will the burnup requirements needed to meet TS Figure 5.6.1 be ascertained for fuel assemblies shipped from other PWR plants (Robinson)?

Response to Requested Item 2

The burnup curve (proposed TS Figure 5.6.1) applies to the Robinson 15 x 15 fuel assembly types identified in Table 4.3.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98.

The selection of spent fuel for shipment to Harris is made in accordance with procedure NFP-NGGC-0003, entitled "Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask." The purpose of this procedure is to assure that the requirements of the IF-300

Cask Certificate of Compliance No. 9001 are met with regard to the selection of irradiated fuel to be shipped and that the fuel selected for shipment is acceptable for storage at CP&L's Harris plant. This procedure has been in use since 1990 for Robinson spent fuel shipments.

A computer program, which has also been in use since 1990 for Robinson spent fuel shipments, is used in conjunction with the above-referenced fuel selection procedure. For candidate assemblies to be shipped, the program retrieves the fuel type, enrichment, burnup, and decay heat from the special nuclear materials database. The initial enrichment data for each fuel assembly is contained in this database along with the other fuel data, and this data is based on manufacturing records. The burnup data for each fuel assembly is also included in the database along with the other isotopic inventories, and this data is obtained from the core monitoring software used for the Robinson plant. The special nuclear material database and core monitoring software have also been in use since 1990 for Robinson shipments.

The burnup curve proposed as TS Fig. 5.6.1 for pools C and D has already been programmed into the software for use in conjunction with fuel selection procedure NFP-NGGC-0003; however, this version is not yet in production as testing and documentation per CP&L's computer code quality assurance requirements are in progress. This new version will screen candidate PWR (Robinson) fuel against the burnup curve.

Revision to fuel selection procedure NFP-NGGC-0003 to reflect criticality screening requirements for fuel to be stored in Harris pools C or D has begun, but will not be completed until after: (1) the software changes identified above have been tested and the revised software placed in production status, and (2) the NRC has approved CP&L's license amendment application to place spent fuel pools C and D in service.

Requested Item 3

The fuel enrichment tolerance is specified in Section 4.5.2.5 as +0.0/-0.05. Why isn't a positive tolerance of +0.05 assumed (i.e., 5.0+0.05 weight percent U-235)?

Response to Requested Item 3

A maximum U-235 enrichment of 5.0 weight percent was specified, because it is the maximum enrichment allowed by both the Robinson and Harris Technical Specifications. Robinson TS 4.3.1.1.a states that the spent fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Robinson TS 4.3.1.2.a states that the new fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Harris TS 5.3.1 states that the initial core loading shall have a maximum enrichment of 3.5 weight percent U-235 and that reload fuel shall have a maximum enrichment of 5.0 weight percent U-235.

Also, the manufacturing facility of Siemens Power Corporation (SPC), the current fuel supplier for both the Robinson and Harris plants, is limited by license to a maximum U-235 enrichment of 5.0 weight percent. The SPC manufacturing tolerance is 0.05 weight percent U-235. Therefore, for enrichments with a tolerance of +/- 0.05%, the nominal design enrichment may not exceed

4.95 weight percent U-235 to ensure that the nominal plus the tolerance does not exceed 5.0 weight percent. The fuel enrichment and density tolerances specified in Section 4.5.2.5 appropriately supports a maximum allowable enrichment of 5.0 weight percent U-235.

<u>Requested Item 4</u>

Justify that the allowance that was assumed for possible differences between the fuel vendor and the Holtec calculations is sufficient to also encompass burnup calculational uncertainties.

<u>Response to Requested Item 4</u>

The Criticality Safety Calculations for the BWR Fuel Racks are summarized in Table 4.2.2 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98. An uncertainty on depletion was not explicitly included in the uncertainties summarized in Table 4.2.2. Instead, the 0.01 additive allowance for comparisons to vendor calculations discussed in Section 4.4.2.2 also accounts for burnup uncertainty. This practice is acceptable for the following two reasons:

First, the BWR calculations consider the peak reactivity during burnup. The k_{inf} in the rack corresponding to a peak k_{inf} in the Standard Cold Core Geometry (SCCG) of 1.32 was calculated in the analysis. The burnup corresponding to this peak reactivity value is simply a by-product of this calculation and, in contrast to PWR analysis, burnup is not used as a criteria for establishing acceptability for fuel storage. Any uncertainty in the burnup calculation would simply decrease or increase, with burnup, the location of the peak reactivity. However, the k_{inf} in the SCCG and the k_{inf} in the rack would remain the same at the peak in reactivity. As a result, an additional uncertainty on depletion is not necessary.

Second, the fuel vendor performs similar depletion calculations to those discussed in Section 4. Therefore any uncertainty in depletion is an inherent part of the comparison between those calculations in Section 4 and those performed by the vendor to determine the peak k_{inf} in SCCG as a function of burnup. Again, it is noted that the actual burnup at which the peak occurs is not used in the BWR acceptable fuel storage criteria.

<u>Requested Item 5</u>

The summary of criticality safety calculations shown in Tables 4.2.1 and 4.2.2 indicates that the total uncertainty is a statistical combination of the manufacturing tolerances but do not indicate methodology biases and uncertainties. Were these included?

<u>Response to Requested Item 5</u>

Section 4.4.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98, discusses the fact that CASMO-3, because it is a two-dimensional code, can not be directly compared to critical experiments and as a result a calculational/methodology bias is not available for CASMO-3. This section also discusses MCNP, which is a full three-dimensional Monte Carlo code, which has been benchmarked against critical experiments. CASMO-3 was used as the

primary method of calculation and the results from CASMO-3 were compared to the regulatory limit of $k_{eff} \leq 0.95$ in Tables 4.2.1 and 4.2.2. As noted, the methodology bias and uncertainty were not included in these tables. However, these factors were implicitly included in a code-to-code comparison between CASMO-3 and MCNP shown in Table 4.5.1.

As discussed above, a methodology bias can not be developed for CASMO-3. Therefore, CASMO-3 results were compared to MCNP results to either verify that it produces conservative results relative to the benchmarked MCNP, or to determine a code-to-code bias. This comparison is discussed in Sections 4.5.1 and 4.6.1 with the results presented in Table 4.5.1. In the comparison between MCNP and CASMO-3, the methodology bias, uncertainty on the bias, calculational statistics, and a correction from 20°C to 4°C were added to the MCNP results. These results indicate that CASMO-3 is conservative relative to the benchmarked code MCNP and therefore the code-to-code bias was 0.0 for CASMO-3. Since the code-to-code bias was 0.0, it was not included in Tables 4.2.1 and 4.2.2. In conclusion, it can be stated that even though a methodology bias and uncertainty were not directly included in the final results shown in Tables 4.2.1 and 4.2.2, they were implicitly included through comparison of CASMO-3 and the benchmarked MCNP, provided in Table 4.5.1.