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402/636-2000

March 19, 1993
LIC-93-0081

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

References: 1. Docket No. 50-285
2. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated December 7, 1992 (LIC-92-340A)
3. Letter from NRC (S. D. Bloom) to OPPD (T. L. Patterson) dated February 4, 1993

Gentlemen:

SUBJECT: Request for Additional Information Concerning the Fort Calhoun Station Spent Fuel Pool Rerack (TAC No. M85116)

Attached is the Omaha Public Power District (OPPD) response to the subject NRC request (Reference 3). This request concerned the Fort Calhoun Station (FCS) spent fuel storage rack modification proposed in the Reference 2 submittal.

As a result of a February 22, 1993 telephone conversation between Mr. S. D. Bloom of the NRC staff and Mr. J. B. Herman of my staff, OPPD has also included an answer to a question regarding the total person-rem exposure from the 1983/1984 rerack project at FCS.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

WGG/grc

Attachment

c: LeBoeuf, Lamb, Leiby & MacRae
J. L. Milhoan, NRC Regional Administrator, Region IV
S. D. Bloom, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector

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OPPD Response to NRC Questions Concerning the Spent Fuel Pool Rerack Modification

NRC QUESTION

1. The Holtec Licensing Report states that a moderator temperature of 4°C, corresponding to the maximum moderator density, was assumed for the criticality analysis to assure that the true reactivity will always be lower over the expected range of water temperatures. However, since the SCALE cross section library as used by NITAWL has scattering matrices only at 20°C and 277°C, explain how the reactivity at 4°C was obtained.

OPPD RESPONSE

The principal calculations of reactivity at 4°C were completed using the CASMO-3 code and verified using NITAWL-KENO5a. An interpolation routine developed by Oak Ridge National Laboratories (ORNL) called "WORKER" was used to interpolate between the 20°C and 277°C libraries in NITAWL. The reactivity increment between 4°C and 20°C is very small (0.0014 δk, Table 4.8.1 of Reference 2). However, the trends in the KENO calculations (including extrapolation to 4°C) were consistent with and confirm the CASMO calculation for 4°C. Appendix A, page A-4 of Section 4.0 of the Licensing Report (Reference 2) discusses the WORKER Routine.

NRC QUESTION

2. The reactivity calculations for the use of CEAs inserted into fuel assemblies assumed that the CEA was depleted to 75% of its initial boron-10 loading for conservatism. What amount of depletion is anticipated for the cycle life of a typical CEA and what amount of core insertion during CEA lifetime is this based upon?

OPPD RESPONSE

The existing depletion assumptions of the CEAs were as previously reviewed and approved by the NRC in FCS License Amendment No. 133, dated October 1990. The previous analyses supported a depletion to 85% of the initial boron at the end of Cycle 10, based on the assumption of 100% insertion. As stated in the previous analysis:

> The evaluation conservatively assumed full CEA insertion although the rods have limited use for flux shaping, i.e., limited rod insertion. The CEAs were in place for Cycles 1 through 10, which had a combined availability of 3,266 full power days, which means the CEAs were inserted for no more than 663.0 full power days. The minimum amount of B^{10} remaining in the CEAs at 663 full power days is 85%...

For additional conservatism, the present evaluation assumed that only 75% of the B^{10} remained in the CEAs. This is conservative, since a CEA is typically not inserted for a significant length of time. Additionally, only Group 4 CEAs are typically inserted for axial shape control. To ensure a uniform depletion of CEAs and not limit depletion to Group 4, the CEAs are shuffled between groups during each refueling. The Group 4 CEAs (five total) are usually not inserted past a 25% insertion length, with normal operation being in an All Rods Out (ARO) position. OPPD's core reload design methodology is based on the ARO condition.

NRC QUESTION

3. Proposed Technical Specification 2.8(12) requires a minimum boron concentration of 100 ppm in the spent fuel pool whenever storing unirradiated fuel in the pool. However, since credit is taken for soluble boron for accident conditions such as lateral motion of the rack modules under seismic conditions, this Technical Specification should require a minimum boron concentration in the spent fuel pool at <u>all</u> times.

<u>OPPD RESPONSE</u>

The response to this question is based on a March 16, 1993 telephone conversation between Messrs. S. D. Bloom and L. I. Kopp of the NRC and Mr. J. B. Herman and Ms. J. L. Bostelman of OPPD. The spent fuel pool at FCS normally is maintained at the refueling boron concentration specified in the Core Operating Limit Report (approximately 1700 ppm). This is assured through administrative controls as detailed below.

FCS Technical Specification 3.2, Table 3-4, Item 5 requires spent fuel pool boron concentration to be determined once per 31 days. Chemistry Surveillance Test Procedure CH-ST-SFP-001 implements the requirements of Table 3-4, Item 5, and establishes the acceptance criterion. Samples are taken in accordance with CH-SMP-PR-0012 and analyzed in accordance with CH-ANL-GW-0002. If boron concentration is determined to be less than the specified limit, the Shift Supervisor is notified. Chemistry Administrative Procedure CH-AD-0003, specifies the chemistry requirements for plant systems including the spent fuel pool. It states the spent fuel pool is sampled monthly, except during refueling outages when it is sampled weekly or if refueling operations are in progress, every eight hours. Form FC-254, Monthly Chemical Analysis Technical Specification Requirements, documents the sampling results. This form is reviewed and maintained as a Quality Assurance record.

Additionally, the two accident conditions detailed above, lateral rack movement and abnormal location of a fuel assembly, require soluble boron to ensure the .95 reactivity limitation is maintained. The double contingency principle of ANSI N-16.1-1975 and the NRC OT Position Paper (Review and Acceptance of Spent Fuel Storage and Handling Applications, dated April 1978) allow credit for soluble boron under abnormal and accident conditions. Therefore, OPPD has included soluble boron in the Abnormal and Accident Criticality Analysis calculations detailed in Reference 2.

NRC QUESTION (From February 22, 1993 Telephone Conversation)

4. What was the total person-rem exposure for the previous rerack project for Fort Calhoun Station?

OPPD RESPONSE

The previous rerack project was performed in late 1983 and early 1984. This project, including all segments of reracking and decontamination, resulted in a total exposure of 25.44 person-rem.

OPPD is confident that the exposure from the previous rerack project is not representative of expectations for this project. In the previous rerack project, OPPD used a contractor with limited experience in such a project. This lack of experience resulted in some job inefficiencies that resulted in higher than anticipated exposures. The current project contractor, Holtec International, has extensive experience in this type of project and a history of performance that validates an estimate of 5-10 person-rem for this project.

NRC Request for a OPPD Commitment:

In the Request for Additional Information letter the NRC identified that OPPD needs to commit to use Appendix A of Regulatory Guide DG-8006 if divers are used during the rerack.

OPPD Commitment

OPPD commits to adhering to the guidelines stipulated in Regulatory Guide DG-8006 and associated safe diving practices. For diving operations in a high radiation spent fuel pool, Regulatory Guide DG-8006 covers several guidelines which will be encompassed in the appropriate procedures associated with reracking of the spent fuel pool.