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Docket No. 50-285

Mr. W. C. Jones
Division Manager, Production Operations
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

In conducting our review of your January 21, 1983 request relating to Spent Fuel Modification for Increased Storage Capacity at the Fort Calhoun Station, Unit No. 1, we have determined that we will need the additional information identified in the enclosure to continue our review.

In order for us to maintain our review schedule, your response is requested within 30 days of your receipt of this letter.

The information requested in this letter affects fewer than 10 respondents; therefore OMB clearance is not required under P. L. 96-511.

Please contact us if you have any questions concerning this request.

Sincerely,

Original signed by
Robert A. Clark
Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Request for Additional Information

cc w/enclosure
See next page

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Omaha Public Power District

cc:

Marilyn T. Shaw, Esq.
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D. C. 20036

Mr. Jack Jensen
Chairman, Washington County
Board of Supervisors
Blair, Nebraska 68023

U.S. Environmental Protection Agency
Region VII
ATTN: Regional Radiation
Representative
324 East 11th Street
Kansas City, Missouri 64106

Metropolitan Planning Agency
ATTN: Dagnia Prieditis
7000 West Center Road
Omaha, Nebraska 68107

Mr. Larry Yandell
U.S.N.R.C. Resident Inspector
P. O. Box 309
Fort Calhoun, Nebraska 68023

Mr. Charles B. Brinkman
Manager - Washington Nuclear
Operations
C-E Power Systems
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, Maryland 20814

Regional Administrator
Nuclear Regulatory Commission, Region IV
Office of Executive Director for Operations
611 Ryan Plaza Drive Suite 1000
Arlington, Texas 76011

REQUEST FOR ADDITIONAL INFORMATION

FORT CALHOUN SPENT FUEL POOL EXPANSION

PART A

1. Provide justification for the conclusion that the degree of agreement with diffusion theory calculations of the Pacific Northwest Laboratories (PNWL) high leakage critical experiments can be properly extrapolated to the Fort Calhoun spent fuel rack infinite array.

2. Since the absorber plate reactivity worths in the fixed neutron poison critical experiments analyzed were much lower than the Boraflex worths in the spent fuel racks, provide justification for the conclusion that the benchmarking signifies that absorption effects were properly treated.

3. The experiments performed by Babcock & Wilcox (M. N. Baldwin, et al, Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel, BAW-1484-7, July 1979) include much higher absorber worths. Have they been analyzed as part of your verification? If so please provide results of the analyses.

4. Have comparisons been made to higher order calculations (e.g., KENO-IV with the AMPX-NITAWL 123 group cross section set)? If so provide results of such comparisons.

5. The statement is made that the use of the simple assembly average exposure can result in an over-estimate of the fuel assembly k_{eff} by about +.015 $\Delta k/K$. Is this based on an actual calculation? If not, how does the k_{eff} of the pancake region consisting of the lower exposure end of the fuel assemblies compare to that calculated based on an assembly average exposure?

PART B

1. Provide a sketch showing the thickness of material in the rack cans.

2. Describe or provide a more detailed sketch of the bounding bar or angle at the top of the rack around the perimeters. What are the stresses in this bar?

3. Provide a tabulation of actual as well as allowable stresses for welds at key points in the racks for pertinent loading conditions.

4. Was local buckling of the cans considered? What acceptance criteria was buckling compared against? Where is the potential for local buckling greatest in the cans?

5. Provide a tabulation of actual buckling stresses compared with allowable stresses in the cans for local buckling.

6. Provide a tabulation of actual and allowable stresses for the key structural components of the racks.

7. Describe the seismic input load for these racks. Were the three components of earthquake input to the ANSYS model? Describe the method in detail.

PART C

1. Regarding the use of the shutdown cooling system to cool the spent fuel pool, provide the following:

The licensee states that the shutdown cooling system can be aligned to provide cooling for the spent fuel pool four (4) hours after receipt of a high pool temperature alarm. The licensee did not specify the condition of the reactor (operating mode) during the time the shutdown cooling system is aligned for spent fuel cooling. Verify that the reactor will be in cold shutdown prior to alignment of the shutdown cooling system for spent fuel pool cooling.

2. The licensee stated in his submittal that the spent fuel pool temperature would be maintained below 120°F. The licensee did not use NUREG-0800, Standard Review Plan, Section 9.1.3 and Branch Technical Position ASB 9-2 for calculating the decay heat loads. Consequently, we are not sure how much conservatism is in the licensee's analysis. Therefore provide the following information with the heat exchangers expected fouling factor and pluggage factor for the life of the plant:

a discussion of the capability and procedure to remove the spent fuel pool cooling system heat exchanger from service for tube cleaning, tube plugging or retubing. The spent fuel pool cooling system consists of two pumps and a single heat exchanger. Include in the discussion of the time available to perform these tasks without exceeding any pool temperature alarm setpoints.

3. On April 14, 1978 a generic letter was sent to all licensees which provided guidance concerning the information to be provided by the utility when requesting spent fuel pool modifications for the purpose of increasing the number of fuel bundles to be stored in the pool. The licensee submittal did not contain all of the information requested by the generic letter. Therefore, provide the following information.

   a. With respect to Section 1.2, verify that no combination of events, and/or failures will result in a K_{eff} of the spent fuel storage arrangement of greater than .95.

   b. Provide a discussion of the onsite tests which will be performed to confirm the presence and retention of the neutron absorber in the racks. The results of the verification tests shall show within a 95% confidence level that there is sufficient amount of neutron absorber to maintain K_{eff} at or less than .95.

   c. Provide a discussion of the periodic surveillance testing to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain K_{eff} at or below .95. The testing should be performed on a statistically acceptable sample size. The frequency of testing should be specified.

4. Provide the maximum uplift forces imposed by the spent fuel handling crane including the consideration of these forces in the design of the racks and the effects on attachments to the pool liner.

5. In the submittal the storage of control rods was not addressed. Our concern is if the control rods were stored in the fuel bundle in the pool, then the additional weight might have a significant effect on the seismic analysis. Verify whether or not more than one control rod at any time will be stored in the spent fuel pool. If control rods will be stored in the spent fuel pool, verify that the seismic analysis provided in the submittal represents the maximum pool liner loadings on the walls and floor, and the maximum inter-rack reactions with consideration being given to the additional weight of the control rods.

6. In July 1980 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was published which requires a generic review of the methodology used in routine moving of heavy loads. A heavy load is defined as any load which weighs more than a fuel bundle and its handling tools. The re-racking of a spent fuel pool is not considered routine and therefore is not within the scope of NUREG-0612. Concerning the moving of spent fuel storage racks, we have the following questions:

   . Verify that procedures are developed, which include the safe load paths, for the removal of existing spent fuel storage racks and installation of new spent fuel storage racks. Verify that all safe load paths for these operations are clearly marked on the floor. Provide drawings which show the load paths of the new and existing spent fuel racks and all other heavy loads associated with this modification of the spent fuel pool.

7. The information provided in the licensee's submittal dated March 12, 1982 did not include a discussion of the capability of the component cooling water system and the raw water system to remove the additional heat from the spent fuel pool. Based on the heat loads using NUREG-0800, Standard Review Plan, Section 9.1.3 and Branch Technical Position ASB 9-2, provide the results of a revised FSAR analysis which shows the increased heat loads from the spent fuel storage expansion on the component cooling water system and the raw water system. Include information which shows the design heat load capacities and the imposed heat loads for normal operation with normal refuelings and for all design basis accident heat loads. For each system, include an analysis of the capability of the system to remove the new spent fuel cooling heat loads and all normal and accident heat loads while maintaining the original design margins for tube fouling and pluggage. No single failure should prevent proper spent fuel pool cooling or safe shutdown. No credit can be taken for any redundant train or component which is not properly qualified for the accident being considered, such as a safe shutdown earthquake, or which requires operator action within 20 minutes (30 minutes if the single operator action is required outside of the control room).

8. Section 3.5 identifies the sequence of rack replacement but no drawing was provided to identify the racks in the pool. Provide a drawing which shows the racks in the pool, the identification of each rack and which racks are defined as Region I and which are defined as Region II.

9. Provide a drawing which shows the load path for each rack and the locations of stored fuel for each movement of a rack.

10. Provide a discussion of your procedure for handling discharged fuel which does not meet the burnup criteria for being relocated into Region II. Assuming one fuel assembly per refueling does not meet the burnup criteria, is failed, or is damaged, provide a discussion of your procedure for offloading the core.

11. The first sentence in Section 8.6 of the submittal states that: "It will be verified." To what does the "it" refer? When will "it" be verified and the information provided for the staff's review?

12. Section 6.3 of the submittal states that an "analysis of the fuel drop accident will be performed." When will the analysis be performed and when will the results of the analysis be provided for the staff's review? Verify that the referenced analysis will include the dropping of the handling tools listed in Table 6.2.

13. Describe the procedure for "measuring for fuel depletion."