UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated July 22, 1977, and supplemented on November 30 and December 16, 1977, and June 12, June 28, July 5, and July 14, 1978, Nebraska Public Power District (NPPD, the licensee) requested an amendment to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The request was made to obtain authorization to provide for additional storage capacity in the Cooper Spent Fuel Pool (SFP). The proposed modification would increase the capacity of the SFP from the present capacity of 740 fuel assemblies (about 1-1/3 cores) to a capacity of 2366 fuel assemblies (about 4-1/3 cores). The increased capacity would be achieved by installing new racks with a decreased spacing between fuel storage cavities. Present racks have a nominal center-to-center spacing between stored fuel elements of 11.9 x 6.6 inches. The proposed spent fuel racks are modular aluminum structures comprised of individual cavities that provide a nominal center-to-center spacing of 6-9/16 inches between stored fuel assemblies. The proposed racks have non structural neutron absorbing curtains of Boral in one direction only.

The spent fuel storage pool is located in the Reactor Building. The general arrangement and details of the proposed new spent fuel storage racks are shown in Figures 1 and 2 of the "Licensing Submittal Report for Cooper Nuclear Station High Density Fuel Storage Racks", forwarded as enclosure 1 with the licensee's letter of July 22, 1977.

The expanded capacity of the SFP would allow the Cooper Nuclear Station to operate until about 1991 without shipping spent fuel from the site. This would still leave room for a full core discharge in the SFP.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

The NPPD fuel pool criticality calculations are based on Exxon and GE fuel assemblies and no soluble boron in the water. For the Exxon fuel assemblies it was assumed that the assemblies have no burnable poison and that the fuel is fresh and of an enrichment as high or higher than that of any fuel available (2.74 w/o). For the GE fuel, k_{eff} was evaluated at two points in burnup; fresh fuel with all burnable poison present, and at 7000 MWD/MTU assuming all the burnable poison is gone. It was assumed that the GE fuel was enriched to 2.83 w/o which corresponds to a maximum fuel loading of 14.5 grams of uranium-235 per axial centimeter of fuel assembly. This is the fuel loading which was used in the calculations for the nominal storage lattice.

Nuclear Energy Services, Incorporated (NES) performed the criticality analyses for NPPD. NES made parametric calculations by using the HAMMER computer program to obtain four-group cross sections for EXTERMINATOR diffusion theory calculations.

The blackness theory program, BRM, was used to calculate the thermal and epithermal neutron group cross section for the boron region. The accuracy of this diffusion theory method was checked by comparison with a critical experiment and by comparison with a four group, discrete-ordinates transport theory calculation, where the cross sections for the three higher energy groups were obtained from the GGC-3 computer program and the thermal group cross sections were obtained from the HAMMER program.

The computer programs were first used to calculate the neutron multiplication factor for an infinite array of fuel assemblies in the nominal storage lattice. Then NES performed calculations to determine: (1) the highest neutron multiplication factor as a function of pool water temperature, (2) the effect of a reduction in storage lattice pitch, (3) the effect of a reduction in the boron loading, and (4) the effect of eccentrically positioning fuel assemblies in the storage lattice. NPPD's July 22, 1977 submittal states that by using a criticality uncertainty value of 0.01 combined statistically with the worst case abnormal configuration,

the upper limit on k_{eff} becomes 0.927. In its November 30, 1977 submittal NPPD states that there is a possibility that one of the fabricated fuel racks may contain a Boral plate in which the boron content in the first two inches of width may contain only 0.114 grams of boron per square centimeter of plate rather than the specified minimum of 0.126 gm/cm^2. NSE calculated the effect of having this decreased boron concentration in two inches of the curtain and found that it will increase k_{eff} by about 0.0008.

This increases NES's maximum calculated value of k_{eff} for fuel assemblies stored in these racks to 0.928.

In regard to the possibility of having a Boral curtain missing from one of the racks, NPPD states that the design of the racks is such that visual verification of the curtains between the storage containers is possible. NPPD also states that neutron radiography will be used to verify the boron content of the curtains.

The above described results compare favorably with the results of parametric calculations made with other methods for similar fuel pool storage lattices. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the nominal fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

NES's criticality calculations were based on an infinite array of cells each of which contained a portion of the boron curtain. These calculations did not take into account the possibility of having a fuel assembly put in the space between fuel racks or brought up close to the outside of a filled rack.

In this regard, NPPD states that there will be a six inch thick support beam on the outside of the racks and structural material and bracing between the racks so that it will not be possible to get a fuel assembly close enough to any of the filled racks that would result in an increase of k_{eff} above the maximum calculated value for the infinite array.

NPPD's proposed onsite inspection for the presence of all Boral curtains and verification by neutron radiography that their boron content is above the specified minimum of 0.126 $grams/cm^2$ except for the one plate which has two inches of width where the boron content is greater than 0.114 $grams/cm^2$ provides adequate assurance that the k_{eff} in the fuel pool will not exceed 0.95.

We find that all factors that could affect the neutron multiplication factor in this pool have been conservatively accounted for and that the maximum neutron multiplication factor in this pool with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a spent fuel pool. This 0.95 acceptance criterion is based on the uncertainties associated with the calculational methods and provides sufficient margins to preclude criticality in the fuel pool.

We find that when any number of the fuel assemblies which NPPD described in these submittals, which have no more than 14.5 grams of uranium-235 per axial centimeter of fuel assembly, are loaded into the proposed racks, the neutron multiplication factor will be less than 0.95.

We have amended the plant's Technical Specifications to prohibit the storage of fuel assemblies that contain more than 14.5 grams of uranium-235 per axial centimeter of fuel assembly. On this basis, we conclude that the rack design will preclude criticality.

2.2 Spent Fuel Cooling

The licensed thermal power for the Cooper Nuclear Station is 2381 MWt. NPPD plans to refuel this plant annually. This will require the replacement of about 112 of the 548 fuel assemblies in the core every year.

In its July 22, 1977 submittal, NPPD assumed a seven day (168 hour) time interval between a reactor shutdown and the time a one-quarter core refueling offload is completed and a thirt n day (312 hour) time interval between a reactor shutdown and th time a full core offload is completed. These time intervals were obtained by assuming 120 hours of in-core cooldown and a transfer rate of three assemblies per hour.

For these cooling times, NPPD states that the maximum possible heat load in the spent fuel pool due to annual refueling will be 7.7×10^6 BTU/hr and that the heat load due to a full core offload will be 19.8×10^6 BTU/hr. NPPD also states that the total heat load for either case was found to be insensitive to the number of assemblies which have been cooling in the fuel pool for more than a few years.

As indicated in Section X-5.5 of the FSAR, the spent fuel cooling system consists of two pumps and two heat exchangers in parallel. Each pump is designed to pump 475 gpm (2.38×10^5 pounds per hour).

Each heat exchanger is designed to transfer 3.2×10^6 BTU/hr from 125°F fuel pool water to 95°F water in the Reactor Building Closed Cooling Water System. For higher heat loads, such as for the full core offload, which would result in the fuel pool outlet water temperature going above 150°F, NPPD states that an operator will be able to connect the Residual Heat Removal System (RHR) to the spent fuel cooling system for additional cooling capacity. Based on the flow rates and design temperature which NPPD stated in its November 30, 1977 response to our request for additional information, we find that the RHR system has the capability for transferring 41.5×10^6 BTU/hr of spent fuel heat to 85°F service water when the spent fuel pool outlet water temperature is at 150°F.

Section X-5.5 of the FSAR indicates that instrumentation is provided in the spent fuel cooling system which will monitor pool water level, pump operation, pool temperature, and system flow. A loss of pump discharge pressure will actuate an alarm in the main control room. Also a detection system is provided to monitor fuel pool water leakage.

Table X-5-1 of the FSAR indicates that the volume of the fuel pool is 4.2×10^4 cubic feet, and Section X-11.3.1 indicates that the makeup water system, which could be used as an emergency source of water for the fuel pool, is designed to provide 125 gpm of demineralized water.

NPPD stated in its submittal that the maximum heat load in the spent fuel pool is relatively insensitive to the number of fuel assemblies which have been cooling in the fuel pool for more than a few years. While we generally agree with this view, we find that the heat generated by the older assemblies cannot be completely neglected. For example, we find that the maximum incremental heat load in this spent fuel pool that will be added by increasing the number of fuel assemblies stored in the pool from 740 to 2366 assemblies will be 2.0×10^6 BTU/hr.

This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools. Therefore, we find that, for a thirteen day cooling time, the peak heat load for the full core offload that essentially fills the pool is 21.9×10^6 BTU/hr. Also, we find that the peak heat loads for annual refuelings will be 8.7×10^6 BTU/hr when sufficient storage capacity is saved for a full core offload and 9.1×10^6 BTU/hr for seventeen annual reloads. These heat loads represent maximum heat loads since they were calculated for a plant capacity factor of one hundred percent.

We calculate that with both spent fuel cooling pumps operating at design capacity and with the peak heat load for any annual refueling (i.e., 9.1×10^6 BTU/hr), the maximum spent fuel pool outlet water temperature will be 138°F. NPPD will use the RHR system for cooling the spent fuel pool when a full core is offloaded.

Since the RHR system has the capability for removing 41.5×10^6 BTU/hr while maintaining the fuel pool outlet water temperature at 150°F, the removal of 21.9×10^6 BTU/hr is well within its heat removal capacity while maintaining the fuel pool outlet water temperature below 150°F.

For this plant, there are two postulated "worst case" cooling accidents that define the maximum emergency actions that could be required. One is for annual refueling offloads. The other is for the full core offload. Both of these postulated "worst case" accidents assume the complete loss of spent fuel pool cooling as a result of a safe shutdown earthquake (SSE). As long as the reactor is shutdown, the RHR system could be used immediately as a backup for the spent fuel cooling system. The maximum emergency measures needed for the annual refueling will be less stringent than those for the full core offload. By the time the reactor is restarted, the heat load in the spent fuel pool will have decayed by a substantial amount. However, if for the purpose of defining maximum emergency measures, we assume that the reactor is running at the time of the completion of an annual refueling offload and the postulated SSE causes the loss of both spent fuel cooling pumps, we calculate that the maximum heat up rate for the fuel pool water will be 5.5 °F/hr. Thus, assuming the maximum 138°F fuel pool outlet water temperature and a shutdown reactor, the operator would have more than two hours to align one RHR system to the spent fuel pool before the outlet water temperature would increase to 150°F. For the full core offload case, if we assume that the postulated SSE causes the complete loss of spent fuel cooling just after a full core offload, the heat up rate of the water in the fuel pool will be about 13°F/hr. Since at this time the temperature of the outlet water from the fuel pool will be less than 150°F, it will take about 5 hours to heat the water at the surface of the spent fuel pool to 212°F when bulk boiling can commence. After bulk boiling commences, the maximum evaporation rate would be 45 gpm.

2.5 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation (SE). There will be no change in the waste treatment systems or in the conclusions of the evaluation of these systems as described in Section 11.0 of the SE because of the proposed modification.

2.6 Structural and Mechanical

The new racks are seismic category I structures. Supporting arrangements for the racks including their restraints; design, fabrication, installation procedures; structural analysis for all loads including seismic and impact loadings; load combinations; structural acceptance criteria; quality assurance requirements for design, fabrication and installation; applicable industry codes; were all reviewed in accordance with the criteria described in Sections 3.7 and 3.8 of the Standard Review Plan. The licensee used seismic input in the form of floor response spectra as approved for the plant FSAR.

The seismic analysis of the fuel rack assemblies was performed by modeling the upper grid and rack base/sub-base structures as two separate three dimensional finite element structures with the masses corresponding to the individual fuel cell weights lumped at the appropriate upper grid and base models. The rack upper grid and base structural analysis frequencies were combined with the fuel can and support frequencies to obtain overall system frequencies. Response spectrum accelerations based on the adjusted first mode frequencies were applied to the model analysis of the upper grid and base structural models. The licensee has performed a confirmatory analysis to prove the validity of the method of superposition of independent structural analyses. The confirmatory analysis consisted of a coupled model of the upper and lower grid structures along with the fuel can assemblies. The analysis showed that the fuel can accelerations were higher than the superposition analysis predicted, however, the stresses were within acceptable limits. The modal responses for the upper grid and base structural analysis were combined in accordance with Regulatory Guide 1.92, "Combination of Modes and Spatial Components in Seismic Response Analysis," December, 1974. In addition, the effect of the fuel assembly impacting the storage cell wall was considered in the seismic analysis.

The racks were analyzed for a fuel assembly drop from a height of 24 inches above the top of the rack. The local and gross effects of the fuel assembly drop on the rack structure were analyzed using energy balance methods.

The effects of the additional loads on the existing pool structure due to the high density storage racks have been examined in accordance with the appropriate portions of Sections 3.7 and 3.8 of the NRC Standard Review Plan.

The use of 6061-T6 aluminum and 300 series of stainless steel materials for the fabrication of the spent fuel racks, and its performance requirements during the service life, were reviewed for consistency with the requirements identified in Section 9.1.2 of the Standard Review Plan. We have considered the possibility of swelling due to gas generation from the radiation environment. We have concluded that no such problem exists for Cooper based on extensive industrial experience with Boral in sealed aluminum cans.

The analysis, design, fabrication and the installation of the proposed fuel rack storage system are in accordance with accepted criteria. The analysis of the structural loads imposed by dynamic, static, seismic and thermal loadings, and the acceptance criteria for the appropriate loading conditions are in accordance with the appropriate portions of Sections 3.7 and 3.8 of the NRC Standard Review Plan.

Since the possibility of long term storage of spent fuel exists, the effects of the pool environment on the racks, fuel cladding and pool liner are being investigated. Based upon our preliminary review and previous operating experience, we have concluded that at the pool temperature and the quality of the demineralized water, and taking no credit for inservice inspection, there is reasonable assurance that no significant corrosion of the racks, the fuel cladding or the pool liner will occur over the lifetime of the plant. However, if the results of the current generic review indicate that additional protective measures are warranted to protect the racks, the fuel cladding and the liner from the effects of corrosion, the necessary steps and/or inspection programs will be determined to assure that an acceptable level of safety is maintained.

We find that the subject modification proposed by the licensee is acceptable and satisfies the applicable requirements of the General Design Criteria 2, 4, and 61 of 10 CFR Part 50, Appendix A.

3.0 SUMMARY

Our evaluation supports the conclusion that the proposed modification to the Cooper SFP is acceptable because:

1. The physical design of the new storage racks will preclude criticality for any moderating condition with the limits imposed.

2. The SFP cooling system has adequate cooling capacity.

3. The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

4. The installation and use of the new fuel racks can be accomplished safely.

5. The restriction on carrying heavy loads over spent fuel which is being incorporated in the Technical Specifications by this amendment will preclude the likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool.

6. The structural design and the materials of construction are adequate and meet the applicable design criteria.

7. The installation and use of the new fuel racks does not alter the consequences of the design basis accident for the SFP, i.e., the rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory within the gap.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: SEPTEMBER 2 9 1978

ENVIRONMENTAL IMPACT APPRAISAL

BY

OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO INCREASE IN

STORAGE CAPACITY FOR SPENT FUEL

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

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