UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 159 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated June 13, 1994, as supplemented by letters dated August 15, 1994, and March 23, April 18, July 21, and September 22, 1995, Duke Power Company (the licensee) submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the TS to increase the initial fuel enrichment limit from a nominal value of 4.0 to 5.0 weight percent Uranium-235, and establish new loading patterns for new and irradiated fuel in the spent fuel pool to accommodate this increase.

The March 23, 1995, supplement provided additional information that modified the June 13, 1994, application's no significant hazards consideration determination, and revised the TS to (1) change the surveillance requirement for boron concentration in the spent fuel pool (SFP), (2) remove the option to use alternate storage configurations in the SFP and replace it with footnotes, (3) add information contained in the Bases to the footnotes, and (4) change the Bases to discuss the option to use specific analyses on alternate fuel. The April 18, July 21, and September 22, 1995, letters provided additional clarifying information that did not change the scope of the June 13, 1994, application and the initial proposed no significant hazards consideration determination.

In the two later submittals, dated July 21 and September 22, 1995, the licensee provided further detailed information relating to cooling and heat transfer in the spent fuel pool.

The staff's evaluation of the proposed changes follows.

2.0 EVALUATION

2.1 Criticality Aspects

The fresh fuel storage racks are used for temporary storage of unirradiated reload fuel and are built on 21-inch centers. The spent fuel pool consists of two regions. Region 1 is designed for storage of fresh or partially irradiated fuel. The stainless steel cells are spaced on a 10.4-inch center-to-center distance and utilize the neutron absorbing material Boraflex with a nominal 0.02 gm/cm^2 boron-10 loading attached to each exterior cell wall.

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Region 1 has a storage capacity of 286 cells. The stainless steel cells in Region 2 are assembled in a checkerboard pattern, producing a honeycomb structure of cell and non-cell locations. The cell center-to-center pitch in Region 2 is 9.125 inches and these cells also utilize Boraflex having a lower boron-10 areal density (0.006 gm/cm^2) than that used in Region 1. Region 2 has a nominal capacity of 1177 cells.

The analysis of the reactivity effects of fuel storage in the new and spent fuel storage racks was performed with the SCALE system of computer codes with the three-dimensional multi-group Monte Carlo computer code, KENO Va. Neutron cross sections were generated by the NITAWL and BONAMI codes using the 123 Group GMTH library. Since the KENO Va code package does not have depletion capability, burnup analyses were performed with the CASMO-3 /SIMULATE-3 methodology. CASMO-3 is an integral transport theory code and SIMULATE-3 is a nodal diffusion theory code. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the McGuire fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber reactivity worth. The intercomparison between two independent methods of analysis (KENO Va and CASMO-3/SIMULATE-3) also provides an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO Va reactivity calculations, a minimum of 90,000 neutron histories were accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO Va reactivity calculations. Based on the above, the staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the McGuire storage racks with a high degree of confidence.

The fresh fuel storage racks are normally maintained in a dry condition, i.e., the new fuel is stored in air. However, the NRC criteria for new fuel storage require that the effective multiplication factor, k_{eff}, of the storage rack be no greater than 0.95 if accidentally flooded by pure water and no greater than 0.98 if accidentally moderated by low density hydrogenous material (optimum moderation). The new fuel storage racks were analyzed for 4.75 w/o U-235 enriched fuel for the full density flooding scenario and for the optimum moderation scenario. The calculated worst-case k_{eff} for a full rack of the Mark BW fuel design, which is the most reactive of the three fuel types which exist at McGuire, as a function of moderator density was 0.9499. Appropriate biases and uncertainties due to the calculational method and material tolerances were included at the 95/95 probability/confidence level. This meets the staff acceptance criteria of 0.95 for full density water flooding and 0.98 for optimum moderation conditions and is, therefore, acceptable.

For spent fuel storage, the staff's acceptance criterion is that k_{eff} of the storage racks be no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level, when fully flooded by unborated water. The licensee has used the acceptable methodology discussed above to demonstrate that fuel assemblies with nominal enrichments up to 4.19 w/o U-235 can be stored in every cell of the Region 1 spent fuel storage racks. To enable the storage of depleted fuel assemblies initially enriched to greater than 4.19 w/o U-235, the concept of burnup credit reactivity equivalencing was used. This is predicated upon the reactivity decrease associated with fuel depletion and has been previously accepted by the staff for spent fuel storage analysis. For burnup credit, a series of reactivity calculations are performed to generate a set of initial enrichment-fuel assembly discharge burnup ordered pairs which all yield an equivalent k_{eff} less than 0.95 when stored in the spent fuel storage racks. This is shown in Table 3.9-1 in which a fresh 4.19 w/o enriched fuel assembly yields the same rack reactivity as an initially enriched 4.75 w/o assembly depleted to 3.4 GWD/MTU. The curve shown in the Table includes biases due to methodology, Boraflex width shrinkage and B_4C self-shielding, as well as an uncertainty due to Boraflex axial shrinkage, a 95/95 methodology uncertainty, and a mechanical uncertainty due to manufacturing tolerances. In addition, a bias and uncertainty associated with fuel burnup was also included. The staff has reviewed the assumptions made in determining these biases and uncertainties, including the results obtained from blackness testing performed on representative Boraflex panels at McGuire in 1991, and concludes that they are appropriately conservative.

New or irradiated assemblies with initial enrichments up to 4.75 w/o U-235 which do not meet the requirements for unrestricted storage in Region 1, but which require temporary placement in Region 1 for operational requirements, must be placed in a restricted loading pattern. Reactivity analyses for these assemblies, arranged in a three-out-of-four storage configuration, were performed using the previously discussed methods. Acceptable fuel assemblies which qualify for storage in the fourth storage location of each three-out-of-four pattern are shown in Table 3.9-2 and are referred to as filler assemblies. These filler assemblies were also determined from minimum burnup versus initial enrichment calculations as described above.

Region 2 of the McGuire spent fuel pools has similarly been analyzed for storage of fuel initially enriched to a maximum 4.75 w/o U-235. Table 3.9-3 shows the minimum burnup required for unrestricted storage in this region, ranging up to 45.10 GWD/MTU for an assembly initially enriched to 4.75 w/o. Table 3.9-4 shows the minimum burnup requirements for restricted storage, i.e., a two-out-of-four configuration, with the remaining two locations either vacant or containing filler assemblies. The minimum qualifying burnup versus initial enrichment for Region 2 filler assemblies are given in Table 3.9-5. Fuel assemblies which do not meet any of these burnup requirements may be placed in a checkerboard configuration in Region 2, but each adjacent cell must remain empty.

These configurations have all been analyzed using the acceptable reactivity methods described previously and meet the NRC acceptance criterion of k_{eff} no greater than 0.95, including all appropriate uncertainties at the 95/95 probability/confidence level. The results are, therefore, acceptable.

Tables 3.9-1 through 3.9-5 contain a footnote which would allow for specific criticality analyses for fuel which differs from those designs used to determine the requirements for storage defined in these tables. This would allow storage of fuel from another facility or storage of individual fuel rods as a result of fuel assembly reconstitution. A similar specification was previously approved for the Oconee Nuclear Station. These analyses would require using the NRC approved methodology described above to ensure that k_{eff} does not exceed 0.95 at a 95/95 probability / confidence level and fuel storage would still be limited to the configurations defined in TS 3.9-13. At the staff's request, the Bases for TS 3.9-13 were revised to include additional discussion which reflects the intended use of this provision.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the spent fuel racks. However, it is possible to postulate events, such as the misloading of an assembly with a burnup and enrichment combination outside of the acceptable requirement, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of boron in the pool water required during storage of fuel by TS 3.9.12 since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. Therefore, the staff criterion of k_{eff} no greater than 0.95 for any postulated accident is met.

The following Technical Specification changes have been proposed as a result of the requested enrichment increase. The staff finds these changes, and the associated Bases changes, acceptable.

(1) TS 3/4.9.12 is being replaced by a new TS 3/4.9.12, which relocates the required minimum spent fuel pool boron concentration in the Core Operating Limits Report (COLR), and by TS 3/4.9.13, which specifies the new required fuel storage requirements given in Tables 3.9-1 through 3.9-5 and Figures 3.9-1 through 3.9-3 based on the reactivity analyses evaluated and approved above. The relocation of the minimum spent fuel pool boron concentration to the COLR has previously been approved by the NRC in a separate licensing action. Based on the NRC staff's recommendation, the licensee has also reduced the soluble boron surveillance interval from 31 days to 7 days, added a statement to Tables 3.9-1 through 3.9-5 indicating that specific analyses may be performed to qualify fuel assemblies for storage using NRC approved methodology, and added additional discussion in the Bases to allow for specific criticality analyses for special situations without requiring additional TS changes, as discussed above.

(2) TS 5.6.1 is being changed to reflect the NRC criticality acceptance criteria for both the new fuel storage racks and the spent fuel storage racks.

(3) TS 5.6.3 is being changed to eliminate reference to a maximum initial fuel enrichment limit since this limit is being relocated to the Tables associated with TS 3.9.13.

2.2 Spent Fuel Pool Cooling and Transfer Aspects

In addition to the initial submittal dated June 13, 1994, the licensee provided their responses to questions raised by the staff relating to cooling and heat transfer in the spent fuel pool.

The spent fuel pool cooling system (SFPCS) consists of two incompletely separated trains. Each train consists of a pump, a heat exchanger (HX) and associated piping and valves. The trains are separated from the pump suction line in the SFP to some distance downstream of the HX in each train, at which point they combine into a pipeline common to both trains, for discharge in to the SFP. Cooling water for each HX supplied by the Component Cooling System (CCS).

A prefilter, deionizer and post-filter serves as part of the SFPCS to remove corrosion and fission products from the water in the spent fuel pool. A portion of the SFP water being moved by a SFPCS pump may be diverted through the prefilter, deionizer and post-filter for the corrosion/fission products removal process. The volume of water corresponding, approximately, to one full SFP can be circulated thought the corrosion/fission product removal process. The volume of water corresponding, approximately, to one full SFP can be circulated through the corrosion/fission product removal process each day. A skimmer loop is also part of the SFPCS. This loop is used to remove debris on the surface of the SFP water.

The staff has reviewed both the licensee's initial submittal and responses to the questions raised and found the licensee's proposal to be acceptable as discussed below.

2.3 New Fuel Storage

The staff found no issues involved in storage of new fuel with increased initial enrichment. Therefore, storage of new fuel is found to be acceptable.

2.4 Spent Fuel Storage

2.4.1 Decay Heat Generation

The licensee calculated the decay heat load for two different cases:

1. The normal heat load, i.e., the heat load generated by a pool filled with 1463 spent fuel assemblies, assuming a normal offload of 76 assemblies is used as the final addition to the SFP, and

2. The maximum heat load, i.e., the pool filled (1463 fuel assemblies) with the final addition assumed to be a full core of 193 assemblies.

In each case, the licensee calculated the decay heat generation by using the methods specified in both ANSI 5.1 and BTP 9-2 of Standard Review Plan (NUREG-0800), Section 9.2.5, "Ultimate Heat Sink," with the following results:

Method	Normal (BTU/Hr.)	Maximum (BTU/Hr.)
ANSI	19.5E6	39.6E6
BTP 9-2	20.8E6	42.2E6

In order to be conservative, the licensee used the higher values found; those found when employing BTP 9-2. A check of some of these values was conducted by the staff. The staff concluded that the method employed by the licensee was conservative, met the criteria of the Standard Review Plan (SRP), and was found acceptable.

2.4.2 SFP Heat Exchanger (HX) Heat Transfer Coefficient

The licensee reported that tests had been conducted with heat exchanger 2B to determine the experimental value of heat transfer coefficient (U) in the equation: $Q = UAF\,\Delta T$

- Q = heat transferred, BTU/Hr.
- U = heat transfer coefficient BTU/HrFt$^{2\circ}$ F
- A = heat transfer area
- F = correction factor for HX
- ΔT = temperature difference

The licensee noted the value of U found for the 2B heat exchanger was 460 BTU/HrFt$^{2\circ}$ F while the value assumed in the design analyses is 321. Use of the lower value of U in the calculation would result, conservatively, in a higher value of SFP coolant temperature than would occur in actuality. Therefore, the use of the lower heat transfer coefficient is acceptable.

2.5. SFP Coolant Temperatures

2.5.1 Normal Case

The licensee reported the results of the analysis for the normal case, using the calculated decay heat generation value of 20.8E6 BTU/Hr. The calculated SFP coolant temperature was reported to be 136° F when using one train. This result is acceptable since it is lower than the SRP guideline of 140° F (SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System").

2.5.2 Maximum Case

The licensee reported that the analysis of coolant temperature in this case was determined to be 137° F when using two SPFC's trains and 180° F when using one train. These results are acceptable since they are lower than the guideline value of less than 212° F as noted in SRP Section 9.1.3.

2.6 Protection of Demineralizer

The licensee noted that the greatest potential for damaging the demineralizer resins would occur on loss of component cooling water (CCW) to the operating heat exchanger when using the demineralizer. In that case, it would take about 2-3 hours to raise the coolant temperature from its normal operating temperature, 90 to 100° F, to 140° F, the temperature at which the demineralizer resins would start to degrade. Annunciator alarms at the chemical and volume control (CVCS) heat exchangers and at upper and lower RCS pump bearing coolers, as well as computer indications at these and other locations served by CCW would indicate low CCW flow. These would serve to indicate both to the senior reactor operator and other operators that failure of the demineralizer could follow; therefore, there is sufficient time for action to be taken to protect the demineralizer. The staff finds this to be acceptable.

3.0 STAFF CONCLUSION

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the McGuire new and spent fuel pool storage racks are acceptable. All normal and accident conditions have been shown to result in a subcritical configuration (k_{eff} less than unity) and thus meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the McGuire TS have been modified to specify the above-mentioned fuel as acceptable for storage in the fresh or spent fuel racks, evaluations of reload core designs (using any enrichment) will, of course, be performed on a cycle-by-cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TS to ensure that reactor operation is acceptable. The staff finds the higher enrichment aspect for the new and spent fuel storage acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on August 24, 1995 (60 FR 44087).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality or the human environment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Kopp
N. Wagner

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