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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated September 19, 1994, as supplemented April 26 and June 19, 1995, Duke Power Company, et al. (the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would allow an increased limit for fuel enrichment. The current new (fresh) and spent fuel storage rack maximum nominal enrichment is 4.00 weight percent (w/o) U-235. As-built manufacturing variations of up to 0.05 w/o U-235 are accounted for in the nominal enrichment value. The proposed changes would allow for the storage of fuel with an enrichment not to exceed a nominal 5.00 w/o U-235 in the Catawba new and spent fuel storage racks. The June 19, 1995, letter provided clarifying information that did not change the scope of the September 19, 1994, application and initial proposed no significant hazards consideration determination.

2.0 EVALUATION - CRITICALITY ASPECTS

The fresh fuel storage racks are used for temporary storage of unirradiated reload fuel and are built on 21-inch centers. Both of the two independent spent fuel pools are designed for storage of both fresh and irradiated fuel. The stainless steel cells for each Unit are spaced on a 13.5-inch center-to-center distance and each has a storage capacity of 1418 fuel assemblies. 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 Catawba fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment and assembly spacing. 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

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of KENO Va reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Catawba 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 5.00 w/o U-235 nominally 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 Westinghouse Optimized Fuel Assembly (OFA) design, which is the most reactive fresh fuel of all fuel types which exist at Catawba, was 0.9302 for full density flooding and 0.9586 for optimum moderation conditions. 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.

Fuel assemblies with nominal enrichments up to 4.00 w/o U-235 can be stored in every cell of the Catawba spent fuel storage racks. To enable the storage of depleted fuel assemblies initially enriched to greater than 4.00 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 4.05 w/o enriched fuel assembly yields the same rack reactivity as an initially enriched 5.00 w/o assembly depleted to 5.67 GWD/MTU. The curve shown in the Table includes biases due to methodology, 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 and concludes that they are appropriately conservative.

New or irradiated assemblies with initial enrichments up to 5.00 w/o U-235 which do not meet the requirements for unrestricted storage 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. These special configurations have 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 and 3.9-2 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 the licensee's other facilities, pursuant to provision 2.B(7) of the Catawba Facility Operating Licenses, or storage of individual fuel rods as a result of fuel assembly reconstitution. 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. At the staff's request, the Bases for TS 3.9-13 was 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 added to establish a required minimum spent fuel pool boron concentration in the Core Operating Limits Report (COLR). The relocation of the minimum spent fuel pool boron concentration to the COLR has previously been approved by the NRC in other licensing actions. Based on the NRC staff's recommendation, the licensee has also reduced the soluble boron surveillance interval from 31 days to 7 days.

(2) TS 3/4.9.13 is being added to specify the new fuel storage requirements given in Tables 3.9-1 and 3.9-2 and Figure 3.9-1 based on the reactivity analyses evaluated and approved above.

In response to the NRC staff's concern, the licensee has added a statement to Tables 3.9-1 and 3.9-2 indicating that specific analyses may be performed to qualify fuel assemblies for storage using NRC-approved methodology and has added additional discussion in the Bases to allow for specific criticality analyses for special situations without requiring additional TS changes, as discussed above.

(3) 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.

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the Catawba new and spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the Catawba 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.

3.0 EVALUATION - SPENT FUEL POOL COOLING & HEAT TRANSFER ASPECTS

In addition to the initial submittal dated September 19, 1994, the licensee provided a response, dated June 19, 1995, to a series of questions raised by the staff, relating to cooling and heat transfer in the spent fuel pool (SFP). 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 into the SFP.

The cleanup portion of the SFPCS consists of a pre-filter (used to remove particulates suspended in the coolant), a deionizer (to remove soluble material), and a post-filter (to remove particulate material).

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

3.1 New Fuel Storage

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

3.2 Spent Fuel Storage

3.2.1 Decay Heat Generation

The licensee used a computer code, "Panther," to calculate the decay heat generated for two cases. One is for the "normal" case (normal reload); the other "maximum" case (full core offload). It is assumed, in the normal case, that the spent fuel pool (SFP) is filled with 1216 assemblies, leaving room for slightly more than one full core to be offloaded, while for the maximum case the SFP is filled with 1409 spent fuel assemblies. The values of decay heat the licensee calculated for these cases are: 18.5E6 BTU/HR for the normal case, 47.0E6 BTU/HR in the maximum case. For the normal case, the licensee could have added one more normal offload since the allowed number of fuel assemblies licensed for the SFP is 1418 in number, 202 more than that accounted for in the analysis and in excess of that contained in the core. Nevertheless, the addition would change the calculated decay heat generation only slightly. Furthermore, the licensee noted that, when the decay heat generation was used to calculate the SFP coolant temperatures (see Section 2.2.3, below) the licensee calculated the decay heat assuming all of the cells were filled with fuel assemblies. Therefore, the licensee's calculation of decay heat generation for both the normal and maximum cases is found to be acceptable.

3.2.2 SFP Heat Exchanger (HX)

The SFP HX's were tested under conditions of high (maximum) and low (normal) heat load with the following results: UAF (heat transfer coefficient x heat transfer area x correction factor) equal to 1.36E6 BTU/HR °F in the maximum case, 1.17E6 in the normal case. The higher value was obtained with a component cooling system (CCS) water flow rate of 3500 gpm, the lower with a CCS flow rate of 2450 gpm. Note that water from the CCS system is used to cool the SFP HXs. To be conservative, the licensee assumed a UAF value of 1E6 BTU/HR °F for the one HX used in the calculation for the normal case and for each HX used in the calculation for the maximum case. In addition, the licensee assumed a CCS flow rate of 3000 gpm to the HX in the normal case and 3000 gpm split between the HX's (1500 gpm to each) used in the two train analysis for the maximum case. The licensee noted that test data showed that one component cooling water pump is capable of delivering 3500 gpm to one HX. The licensee assumed an SFP coolant flow rate of 2300 gpm for each SFP coolant system pump while each is designed for a flow rate of 2840 gpm. The heat exchanger (UAF) and coolant flow rates used in the analysis, including the values for flow for the component cooling water and SFP coolant, are conservative and, thus, found to be acceptable.

3.2.3 SFP Coolant Temperatures

For the normal case, the licensee reported the results of the analysis using the Panther calculated decay heat generation value. The calculated SFP coolant temperature was reported to be 128° F when using one train. This result is acceptable since it is lower than the Standard Review Plan (SRP) guideline value of 140° F.

For the maximum case, the licensee reported that the analysis of coolant temperature in this case was determined to be 145° F. This result is acceptable since it is lower than the value of 150° F noted in the Catawba Safety Evaluation Report, NUREG-0 and the guideline of 212° F noted in SRP, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

Therefore, the analyzed values of SFP coolant temperatures are found acceptable for both the normal and maximum cases.

3.3 Protection of Deionizer

The resins in the deionizer in the cleanup portion of the SFPCS have an operating limit of 140° F. There is a temperature alarm, set to operate at a temperature of 135° F so as to permit an operator to shut off that portion of the system (when in operation) before damaging the resins. Therefore, the method by which the resins in the deionizer are protected from excessive temperature is found to be acceptable.

3.4 Standby Shutdown Facility (SSF) Event

The standby shutdown system (SSS), which is part of the SSF is designed to mitigate the consequences of a fire at the Catawba Nuclear Station. The basis of the design of the SSF is to allow maintenance of a hot standby condition for 72 hours. As part of this process, the standby makeup pump in each unit is designed to pump water from the SFP into the reactor coolant system (RCS) via the RCS pump seals. The makeup pump is capable of pumping at least 26 gpm of makeup water from the SFP into the RCS; 14 gpm for seal leakage and 12 gpm for RCS makeup. The licensee reported that the calculation of water loss from the SFP included RCS makeup, boiloff, with the assumption that the analysis was initialized with the minimum amount of water in the SFP. The licensee concluded that "--boiloff to the top of the fuel assemblies will not occur until well after 72 hours". The actual calculation showed that boiling was attained in 26 hours after initiation of the standby makeup pump (starting at an initial temperature of 135° F). The calculated time for the initiation of fuel uncovery would occur 92 hours later, for a total time of 118 hours after initiation of the standby makeup pump.

This is found to be acceptable since the initiation of uncovery of fuel in the spent fuel pool does not occur within 72 hours, in compliance with the design basis of the SSF.

3.5 SUMMARY

The staff has concluded that the licensee's submittal is acceptable in the areas of spent fuel pool cooling and heat transfer.

An issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," October 7, 1993, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the licensee under separate correspondence.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South 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 have been prepared and published in the Federal Register on August 28, 1995 (60 FR 44513). Accordingly, based upon the evironmental assessment, the Commission has determined that the issuance this amendment will not have a significant impact on the quality of 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.

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Date: August 31, 1995