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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.121 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By letter dated January 6, 1989, Omaha Public Power District (OPPD) submitted an application for an amendment to Facility Operating License No. DPR-40 that would modify the Fort Calhoun Station, Unit No. 1, Technical Specifications (TS) to (1) change the containment spray system surveillance testing requirements to provide a quantitative value to define the minimum acceptance criteria, (2) change the Basis of the containment spray system surveillance requirements by providing the minimum spray flow requirements determined from analysis, (3) reduce the maximum power level permitted on Figure 2-7, Limiting Condition for Operation for Departure from Nucleate Boiling Monitoring, (4) revise Figure 2-3, Predicted Radiation Induced NDTT Shift, based on calculations using US NRC Regulatory Guide 1.99, Revision 2, (5) correct the neutron fluence value stated as occurring at 14 Effective Full Power Years (EFPY) at the inner surface of the reactor vessel wall at the critical weld location from 1.4×10^{19} n/cm² to 1.21×10^{19} n/cm², and (6) change the references in TS 3.6 from "FSAR" to "USAR" and adding an additional reference to USAR Section 14.16.

2.0 DISCUSSION

2.1 Containment Spray System Surveillance Requirements (Items 1 and 2)

The Fort Calhoun Station Containment Spray (CS) System consists of three containment spray pumps which supply flow, via a common header, to two independently isolable spray headers. Each spray header contains 274 nozzles. The present Technical Specification surveillance requirement for the spray system states, in part, that the system test will be considered satisfactory if visual observations indicate all components have operated satisfactory. However, 10 nozzles on each spray header are blocked by ventilation ductwork and piping, and one nozzle on one of the headers is missing, thus affecting the operability of these nozzles.

The licensee provided an analysis, "Reexamination of Containment Pressure Response for the DBA LOCA and DBA MSLB Events at Fort Calhoun Station", dated December 1988, which was conducted to determine the response during the postulated Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) events, using the revised information on the spray nozzles. Additionally, the analysis included the revised containment spray pumps start logic which delays the time that water is discharged from the spray headers. The pumps now start on the same signal which opens the header isolation valves (containment spray actuation signal) rather than on the safety injection actuation signal (SAIS).

The pressure response analysis was conducted using the CONTRANS containment code, which was the same as used during the previous analysis in USAR Section 14.16.5, and with the current plant configuration. The single header atomized spray flow was reduced from 3400 gpm to 3155 gpm to account for the blocked and missing nozzles. The computer run conducted with a containment spray delay time and fan cooler actuation time of 1200 and 60 seconds, respectively, produced a peak containment pressure of 56.3 psig at 60.3 seconds into the event. The results of this run provided the basis for the LOCA case analysis.

For the MSLB case analysis, the present analysis used the same computer code, the combined SGNIII/CONTRANS code, as utilized in previous analysis. The analysis produced a peak containment pressure of 58.7 psig at 71.7 seconds for the benchmark MSLB response. As noted, the peak occurs prior to any actuation of the sprays.

The summary of the LOCA analysis results show that peak containment pressure is reached at the time of activation of the containment fan coolers. For purposes of single failure, only one containment fan cooler and filtering unit and two cooling units were assumed to operate. The analysis also showed that, if no active heat removal was available, the containment design pressure of 60 psig would be reached at 176 seconds. With the spray pumps now starting on the CSAS, spray flow from the nozzles would occur after approximately 90 seconds, due to the delays in time for the pumps to come up to speed and to fill the lines and headers after the CSAS. The analysis also shows that peak containment pressure occurs nearly concurrent with the initiation of active heat removal from the sprays; therefore, the containment design pressure would not be exceeded. Thus, the change to the containment spray pump start logic is adequate since spray initiation remains at less than 176 seconds and containment spray is redundant to the fan coolers.

For the MSLB cases, the steam generator blowdown energy rates to containment are generally greater than the removal rates from the combined fan capacity and containment wall. Therefore, peak containment pressure occurs after fan cooler actuation and when the generator reaches dryout conditions. The previous MSLB analysis assumed spray delivery at 55 seconds which strongly influenced peak pressure due to the effect of the sprays on the superheated containment atmosphere. However, even without fans and a delay of spray actuation, the peak pressure is 59.3 psig and thus, neither are needed to control peak pressure.

The staff finds the analysis assumptions, input conditions, and computer code utilized to be satisfactory for both the LOCA and MSLB containment response analyses. Also, the analysis results appear to be in agreement with the conditions. Thus, the staff finds the proposed changes to the number of fully operable spray nozzles and the reduced single spray header atomized flow rate are acceptable.

2.2 Maximum Power Level Permitted by LCO for DNB (Item 3)

The Limiting Condition for Operation (LCO) for Departure from Nucleate Boiling (DNB) Monitoring, Technical Specification Figure 2-7, provides the core power level limitation versus the Axial Shape Index, Y_I. This is one of several parameters which are maintained to ensure that the fuel design limits will not be exceeded during a design basis anticipated operational occurrence and the consequences of a DBA will be no more severe than predicted. The present Figure 2-7 defines a core power limit of 100.5% of rated thermal power for Y_I between a value of -0.057 and 0.098. However, the USAR safety analysis were performed with an input assumption of reactor power at 102% (e.g. 100% plus 2% uncertainty). Thus, plant operation up to a limit of 100.5%, assuming the uncertainty, may cause these analysis to not be valid. Additionally, the license conditions do not allow steady state power levels above 1500 MWt, which is 100% of rated thermal power. Since this proposed change is a further restriction to ensure that the authorized power level is not exceeded and no safety analysis are affected, the staff finds the change to be acceptable.

2.3 Revise Predicted Radiation Induced NDTT Shift (Item 4)

Regulatory Guide 1.99, Revision 2, provided equations for predicting the shift in nil ductility transition temperature, RT_{NDT}, due to neutron irradiation at the reactor vessel inner surface and the 1/4t depth from the inner surface. The present Technical Specification Figure 2-3 provides curves of this temperature shift versus the irradiation level based on draft Revision 2 of the regulatory guide. However, Revision 2 issued a different through wall attenuation equation than that in the draft. This has required a change in the Figure 2-3 curves to correspond with the new equation. The value of the RT_{NDT} shift is used in the adjustment of the heatup and cooldown curves so that sufficient margin is maintained. Since the value of the predicted RT_{NDT} shift used in the generation of the heatup and cooldown curves was more conservative than that in the proposed change to Figure 2-3, no further correction of these curves was necessary. Therefore, the staff finds this change to the Figure 2-3 to be acceptable.

2.4 Administrative Changes

a. Neutron fluence (Item 5)

The Basis for Technical Specification 2.1.2, Heatup and Cooldown Rate, states that the predicted neutron fluence at the reactor vessel inner surface for the critical beltline weld at 14 Effective Full Power Years (EFPY) is 1.4×10^{19} n/cm². During a previous amendment, No. 114, which changed the heatup and cooldown curves to correspond to operation through 14 EFPY rather than 15 EFPY, the stated value of neutron fluence was not changed. The proposed change to the TS provides the correct value of 1.21×10^{19} n/cm². This change is administrative in nature since it causes no impact on any analysis and the heatup and cooldown curves had been previously changed for operation through 14 EFPY. Therefore, the staff finds the proposed change to be acceptable.

b. References to USAR (Item 6)

The licensee has proposed changes which provide correct references to the Updated Safety Analysis Report (USAR) rather than the erroneous Final Safety Analysis Report (FSAR) references set forth in Technical Specifications (TS) 3.6. The change also adds an additional reference for USAR Section 14.16 to this TS as a clarification. The staff finds the proposed changes and clarifications to be administrative in nature and to correct the reference information errors. Thus, the proposed changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of a facility, component located within the restricted area defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The NRC staff has concluded, based on the consideration 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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