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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 55 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, 1988, and supplemented December 28, 1988, and March 6, 1989, Duke Power Company, et al., (the licensee) requested changes to the Technical Specifications (TSs) and Bases associated with the relocation of steam generator (SG) lower instrument taps for Catawba Unit 2. These changes, which revise the setpoints for SG level trips, are applicable to Unit 2 only. Unit 1 is included administratively because the TSs are combined in one document for both units. The proposed changes for Unit 2 would revise:

(1) Table 2.2-1, Item 13.b
(2) Table 3.3-4, Items 5.b.2., 6.c.2., and 8.c.2)
(3) the Bases for Steam Generator Water Level, page B 2-7

The March 6, 1989, submittal clarified certain aspects of the request. Therefore, the substance of the changes noticed in the Federal Register and the proposed no significant hazards determination were not affected.

2.0 EVALUATION

Catawba Unit 2 is equipped with Westinghouse Model D5 SGs while Unit 1 has Model D3. A major difference between those two models is the design of the moisture separator section. Due to this design difference, the narrow range span (NRS) currently used for control and protection functions is different. Specifically, the lower instrument tap for the narrow range level instrumentation was located above the transition cone and lower deck plate on the D5 SG at Catawba Unit 2 as opposed to below the transition cone in the downcomer in the D3 SG on Unit 1. Due to the location of the lower tap in the D5 SG, the shrink and swell characteristics are more pronounced than in the D3. This makes plant control more difficult and more susceptible to trips. The proposed modification will relocate instrument tap on the D5 SG to a location similar to the D3.

In order to determine the potential gain in operational control characteristics of the D5 SG if the lower instrument tap were relocated to the equivalent location as the D3, the licensee and Westinghouse installed pilot instrumentation on the Catawba 2C SG to determine the potential gain in operational control characteristics due to the proposed modification. Transient data indicated that the modified D5 level instrumentation will perform similarly to the D3 in terms of post-trip response. Also, the current Unit 2 transmitters will be replaced by Barton 764 transmitters, of the same type used in Unit 1, which are environmentally qualified for post-accident conditions.

By relocating the lower tap, the high level trip, lower level trip and operating level trip setpoints will be reduced. With this proposed arrangement, the margin between the operating level setpoint and low level trip setpoint will be increased from a current 42" to 58". This will make Unit 2 more tolerant to feedwater system malfunctions at power, thus reducing unnecessary reactor trips and challenges to safety systems.

Relocating the narrow range instrumentation lower sensing tap on the Westinghouse model D5 SG to the same elevation as the model D3 SG would provide the following safety enhancements:

(1) The effects of level shrink and swell at low power levels will be greatly reduced, thus reducing the potential for reactor trips.

(2) The time necessary to recover indicated level following a reactor trip will be greatly reduced, thus reducing the potential for an overcooling event due to excessive auxiliary feedwater.

(3) The margin to low level trip will be increased thus reducing the potential for reactor trips at power.

Relocation of the level sensing tap to the downcomer region requires that the velocity induced error be accounted for in the determination of trip and operating level setpoints. This can be accomplished without reducing any current margin to trip. The proposed values for trip setpoints are based on the latest component error data and the same approved Westinghouse setpoint methodology is used.

The licensee performed a detailed evaluation to assess the effect of this design modification on the Catawba Final Safety Analysis Report (FSAR) transients and accident analyses. Most of the analyses remain bounded except the Steam Generator Tube Rupture (SGTR) event and a few feedwater related transients. As a result, the feedwater related transients were reanalyzed and it was concluded that the auxiliary feedwater system is adequate to provide sufficient heat removal from the reactor coolant system following reactor trip.

The licensee has previously reanalyzed the SGTR event (Ref. 1) to comply with the appropriate license condition for Units 1 and 2. This analysis is currently undergoing staff review. It has been determined that the design modification would have a minor impact on the dose calculations for a SGTR event by increasing the time the steam generator tubes may be uncovered from 12 to 13 minutes. The licensee stated that a 15-minute tube uncovery time was used for the dose calculations and therefore bounds the impact of relocating the lower instrument taps. The licensee's dose calculations showed compliance with 10 CFR 100 requirements.

Although the staff's review of the licensee's SGTR analysis is not yet complete, we concur with the licensee's assessment that relocation of the level taps will have only a minor impact on the resultant dose consequences for a SGTR event. Furthermore, we find that the licensee's analysis provides a reasonable assurance that the 10 CFR 100 requirements can be met.

Based on its review, the staff concludes that implementation of the design modification, while the staff completes its review of the SGTR analysis, has no adverse impact on safety and does not pose an undue risk to the public health and safety, and is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendments involve 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 exposure. The NRC staff has made a determination that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 6191) on February 8, 1989. The Commission consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCE

1. Letter from Hal B. Tucker (Duke Power Company) to USNRC, dated August 24, 1988, regarding License Conditions 16 (Unit 1) and 10 (Unit 2) on Steam Generator Tube Rupture Analysis and Technical Specification Amendments.

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