ATTACHMENT 1 TO AEP:NRC:1110

REASONS AND 10 CFR 50.92 ANALYSIS

FOR CHANGES TO THE

DONALD C. COOK NUCLEAR PLANT UNIT 1

TECHNICAL SPECIFICATIONS

This letter proposes to modify Unit 1 Technical Specification Table 3.3-2 (Reactor Trip System Instrumentation Response Times) Items 12 and 13. These items are the response times for the single loop and two-loop loss-of-flow reactor trips, respectively. We are proposing to change the required response time for both of these trips from 0.6 seconds to 1.0 seconds.

Description of Trips

These are two components of the loss-of-flow reactor trips. Above the P-8 permissive (approximately 31% of rated thermal power), degradation of flow in a single reactor coolant loop to 90% of the loop design flow will cause a trip. Below the P-8 permissive but above the P-7 permissive (approximately 11% of rated thermal power), degradation of flow to 90% of design in two loops is necessary to cause a trip. Below the P-7 permissive, no reactor trip on loss of flow is active.

Analysis Considerations

The current accident analyses involving the loss-of-flow reactor trips were performed by Westinghouse Electric Corporation (Westinghouse), as documented in WCAP 11902. These analyses were submitted to the NRC in our letter AEP:NRC:1067, dated October 14, 1988. The analyses were accepted by the NRC via Amendment 126 to our Unit 1 license, on June 9, 1989.

Reactor trip on loss of flow was assumed in two analyses in WCAP 11902. These analyses were 1) coastdown of one pump with four coolant loops in operation, and 2) locked rotor accident. Both of these analyses assumed the reactor was at full power as a starting condition. The full power condition was analyzed because it was determined that this was conservative with respect to the acceptance criteria, which include DNBR and reactor coolant system overpressurization. The analyses assumed a 1.0-second reactor trip system response time, rather than the 0.6-second response time in the current T/S.

Although the analyses were done only at full power (corresponding to the loss of flow in a single loop above P-8), the change in response time from 0.6 seconds to 1.0 seconds is also justified for the reactor trip on loss of flow in two loops above P-7 but below P-8, since, as stated above, the full power cases are limiting with regard to the DNB ratio and reactor coolant system overpressurization. (Between P-7 and P-8, a one-pump coastdown would not necessarily result in a direct reactor trip. The reactor would reach a new steady state consistent with the flow provided by the remaining three pumps. Power may increase due to moderator temperature feedback, but would be limited by the P-8 setpoint. After the P-8 reactor power level was exceeded, loss of flow in a single loop would automatically become sufficient to cause the reactor trip. Thus, the case of initial power between P-7 and P-8 is bounded by the case of initial power above P-8, and it can reasonably be concluded that the change in response time can be applied to both of the loss-of-flow reactor trips.)

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

1) Involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

3) Involve a significant reduction in a margin of safety.

Criterion 1

The change from a response time of 0.6 seconds to 1.0 seconds for the loss of flow reactor trips is consistent with the assumptions of the current accident analyses, as approved by the NRC via Amendment 126 to the Unit 1 T/Ss. The accident analyses demonstrated acceptable DNBR and reactor coolant system pressurization results for the applicable accidents. It can be reasonably concluded, therefore, that the change will not involve a significant increase in the consequences of a previously analyzed accident. The proposed T/S change will not require any physical modifications to the plant nor any changes in plant operating configuration. The required time response of the loss-of-flow trips is a factor in the results of the appropriate accident analyses, but is not an initiating event. Therefore, we believe the change will not involve a significant increase in the probability of a previously analyzed accident.

Criterion 2

As discussed in Criterion 1, above, the proposed T/S change involves no physical changes to the plant nor any changes in plant operating configuration. Therefore, we believe the change will not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

The proposed change is consistent with the assumptions of the current accident analyses. (These analyses were approved via Amendment 126 to our Unit 1 license and will be incorporated into our next annual FSAR update.) These analyses demonstrated acceptable DNBR and reactor coolant system pressurization results, and were approved by the NRC in support of Amendment 126 to the Unit 1 T/Ss. Therefore, we believe the change does not involve a significant reduction in a margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth example refers to changes that may result in some increase to the probability or consequences of a previously analyzed accident, but the results of which are within established acceptance limits. Since the proposed change is supported by analyses performed by Westinghouse that have been previously accepted by the NRC, we conclude that the example cited is relevant and that the change should not involve significant hazards consideration.

ATTACHMENT 2 TO AEP:NRC:1110

PROPOSED REVISED TECHNICAL SPECIFICATION CHANGES

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13. Loss of Flow - Two loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14. Steam Generator Water Level--Low-Low	$\leq$ 1.5 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE