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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Joseph M. Farley Nuclear Plant - Units 1 and 2
Proposed Steam Generator Tube Plugging Limit and
Heat Flux Hot Channel Factor Technical Specification Changes

By letter dated June 2, 1987, Alabama Power Company submitted proposed changes to the Technical Specifications which would increase the steam generator tube plugging limit and F_Q coefficient. NRC questions concerning this submittal were provided verbally to Alabama Power Company. The response to these questions is enclosed.

Should you have any further questions, please advise.

Respectfully submitted,

R. P. McDonald

RPM/BHW:dst-D-T.S.7

Enclosure

cc: Mr. L. B. Long
Dr. J. N. Grace
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ENCLOSURE

(1) NRC Question

With the plugging of up to 10% S.G. tubes, the flow area will be decreased and the resistance will go up. This will tend to decrease the RCS flow. Please give information on the following for Farley Nuclear Plant - Units 1 and 2:

A. What is the Thermal Design Flow (TDF) value and where is it documented in the FSAR?

B. What are the latest measured RCS flow values?

C. What is the flow measurement uncertainty values?

D. What is the calculated RCS flow value for 10% S.G. tube plugging?

APCo Response

A. The TDF for Farley Units 1 and 2 is 88,500 gpm per loop which corresponds to a total RCS flow of 265,500 gpm. TDF is described in FSAR Section 5.1, Summary Description, and the value is listed in FSAR Table 5.1-1, System Design and Operating Parameters.

B. The latest measured total RCS flow values were 283,963.3 gpm for Unit 1 and 285,767.6 gpm for Unit 2. These values are documented in the Unit 1-Cycle 8 and Unit 2-Cycle 5 Startup Reports which were submitted to the NRC in Alabama Power Company letters dated March 13, 1987 and August 28, 1986, respectively.

C. Farley TDF values for Units 1 and 2 are 265,500 gpm. Since initial plant startup, flow measurement values have consistently been in excess of 15,000 gpm above the TDF. An evaluation of the plant calorimetric procedure and instrumentation results in a bounding estimate for the RCS flow measurement uncertainty of 2.5%. The measured flow values are sufficiently above the TDF to account for RCS flow measurement uncertainties even if the generic NRC flow measurement uncertainty of 3.5% for Westinghouse plants is applied.

D. With 10% of the steam generator tubes plugged, the calculated RCS flow is 94,500 gpm per loop or 283,500 gpm total for Unit 1 and 93,800 gpm per loop or 281,400 gpm total for Unit 2. These values are based on best estimate flow calculations.

(2) NRC Question

It appears that there may be some changes to the heat balance on the steam generator as the surface area for heat transfer will be reduced and the flow rate will be reduced due to the increase in resistance. If the power is kept constant with a reduction in flow, the temperature difference will change. What effect will this have on the core average temperature and will any Technical Specification trip settings need to be changed?

APCo Response

With 10% tube plugging the TDF will not change. The core average temperature will not change with the same power level and the same vessel average temperature. TDF and RCS average temperature are inputs to the non-LOCA safety analyses. Raising the tube plugging level to 10% has not changed these values. Therefore, no changes to Technical Specification trip setpoints are required.

(3) NRC Question

You state that for low steam generator plugging levels (up to 20%), Small Break LOCA transients would not be affected by the tube plugging. What is the bases for saying that 20% tube plugging is small and would have no affect?

APCo Response

The intent of the 20% tube plugging statement was to demonstrate margin above the 10% plugging level. The conclusion that 20% tube plugging would have no affect in the Small Break LOCA analysis resulted from an evaluation performed in 1985 based on a three-loop Westinghouse PWR. Three relevant phenomena described in that evaluation are: 1) Only a small portion of the steam generator tube heat transfer area is sufficient in a small break transient to provide effective heat sink to the primary side,[1] 2) Operating temperature differences as a result of plugging disappear right after the break because the secondary side pressure reaches steam generator safety valve setpoints almost immediately, 3) The Counter Current Flow Limit (CCFL) characteristics would be such that the CCFL would still be dominant and limiting in the inclined pipe connecting the steam generator inlet plenum to the hot leg for steam generator tube plugging levels up to 20%. For steam generator tube plugging levels beyond 20%, a CCFL calculation in steam generator tube locations would increase. This would reduce the dominance of the CCFL in the inclined pipe, in which case, the plugging level would exert an influence.[2]

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[1]Ciana, S., Patti, B. and Lee, N., "Simulation of Small Break Type behavior of PUN and SPES using the NOTRUMP Code," Proceedings of the Specialists Meeting on Small Break LOCA Analyses in LWRs, Pisa, Italy, June (1985).

[2]Lee, N., "Limiting Counter Current Flow Phenomenon in Small Break LOCA Transients, "Proceedings of the Specialists Meeting on Small Break LOCA Analyses in LWRs, Pisa, Italy, June (1985).

Farley's current small break analysis is based on the WFLASH code. The reference evaluation of 1985 further goes on to conclude the effect of steam generator tube plugging would not be seen in WFLASH analyses because the CCFL phenomenon is not taken credit for in WFLASH.

When the new NOTRUMP code is factored into the evaluation, there is compensation for any Peak Clad Temperature (PCT) effect for steam generator tube pluging due to the fact that PCTs are significantly lower when analyzed with NOTRUMP. The NOTRUMP code will calculate steam generator plugging effects which will become significant in PCT at levels above 15% because of CCFL modeling improvements. NOTRUMP calculations have been performed and reported that have corroborated this position, showing a small PCT effect between cases assuming 15.3% and 23.4% plugging levels, though based on a non-Westinghouse plant.

It has been concluded that the reference evaluation is directly applicable to the Farley plants. Based on these considerations, it is concluded that for a 10% steam generator tube plugging level there would be no adverse effect on the WFLASH small break analysis of record. By projecting effects if analyzed with NOTRUMP, minimal steam generator tube plugging effects in PCT would be expected to be observable at a level near 15% to as high as 20%. But, again, this would be insignificant compared to the PCT improvement that would be expected by applying the NOTRUMP Evaluation Model. The Farley analysis of record results would continue to be bounding.