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SUBJECT: Responds to NRC 890626 RAI re rept of evaluation of effects of steam line break on standby shutdown facil capabilities.

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July 24, 1989

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Request for Additional Information Concerning Oconee Nuclear Station - Report of Evaluation of Effects of Steam Line Break on Standby Shutdown Facility Capabilities (TACS 68348/68349/68350)

Gentlemen:

By a NRC letter dated June 26, 1989 a request for additional information on report, document 86-1106952-00, dated February 18, 1980, "Report of Evaluation of Effects of Steam Line Break on Standby Shutdown Facility Capabilities" for Oconee Nuclear Station, Units 1, 2, and 3 was received from Leonard A. Wiens, Project Manager, Project Directorate II-3 Division of Reactor Projects I/II, Office of Nuclear Reactor Regulation. I am submitting a written response to this request.

Very truly yours,

Hal B. Tucker

RRE/59/td
Attachment
cc: Mr. S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, GA 30323

Mr. P. H. Skinner
NRC Resident Inspector
Oconee Nuclear Station

Mr. L. A. Wiens
Office of Nuclear Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

A001

ATTACHMENT

Draft Responses To 6/26/89 NRC Letter

ONS SSF Steam Line Break Mitigation

Question 1:

It is our understanding that the individual steam line rupture area is 6.3 square feet. This is about 20 percent larger than was used in the analyzed event. Please confirm that the conclusions of the analyzed event will not change with the increased area, and provide a basis for this conclusion.

Response:

The rupture area used in the steam line break analysis is 5.29 ft^2 per steam line. The maximum possible rupture area is 6.28 ft^2, which corresponds to the steam generator steam outlet nozzles. Since this steam line break analysis is unmitigated in terms of pumped high pressure injection of borated water, there is little significance to the size of the rupture. Rupture size only impacts the cooldown rate for this scenario. The rate is not important since there is no corresponding rate of boron addition. What is important for this scenario is the mass of secondary coolant delivered to the steam generator, which determines the magnitude of the primary overcooling. This mass has been very conservatively modeled in this analysis in order to bound the potential consequences of any steam line break requiring mitigation by the SSF. The dominant conservative assumption is that main feedwater continues for 60 seconds following the steam line break. This is impossible considering that offsite power is immediately lost, and this results in a loss of hotwell pumps and condenstate booster pumps, and a consequent loss of main feedwater pumps. Steam supply to the main feedwater pump turbines is also rapidly lost as a result of the steam line break. Another conservative modeling feature is that main feedwater flow increases in response to the decrease in steam generator pressure, and approaches a runout flowrate. A flowrate of this magnitude cannot be accommodated by the condensate/feedwater train. The combination of these modeling assumptions, which produce a very conservative secondary mass addition, result in a conservative simulation of the response to a double steam line break. There is no impact resulting from the rupture area being somewhat less than the maximum.

Question 2:

For the analyses, the RCS pumps are assumed to trip at the time the control rods drop. Please confirm that the conclusions of the analyzed event will not change if offsite power remains available to drive the RCS pumps. If the conclusions of the analyzed event change, please justify your assumption that a loss of off-site power occurs in conjunction with the steam line ruptures.

Response:

The assumption of a loss of offsite power at the time of reactor trip is consistent with the design basis of the SSF, which assumes a loss of offsite power and a loss of all onsite non-SSF power. This assumption is also consistent with the assumptions used by NRC and Los Alamos National Laboratory when performing vital area modeling analyses. This is in itself a sufficient basis for the assumption.