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SUBJECT: Forwards info as followup to 890616 telcon request for discretionary enforcement relief from Tech Spec 4.8.2.

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June 28, 1989

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Oconee Nuclear Station
Document Numbers 50-269, -270, -287
Discretionary Enforcement Relief
From Technical Specification 4.8.2

Dear Sir:

This letter constitutes written follow-up of a request for temporary waiver of compliance from the requirements specified by Oconee Nuclear Station Technical Specification 4.8.2 via a telecon between members of my staff and members of the NRC Staff on June 16, 1989. This temporary waiver from compliance with the Technical Specification Surveillance Requirements was requested to avoid performing an unnecessary test of the main steam stop valves during each refueling outage for all three units. The requested waiver will be required until a proposed Technical Specification Amendment can be submitted and approved by the NRC staff. I currently anticipate being able to submit a Technical Specification Amendment request for NRC review and approval by no later than July 14, 1989.

The proposed waiver of compliance request was the result of a review of the procedure used to implement the surveillance requirement of Technical Specification 4.8.2. Technical Specification 4.8.2 requires that during each refueling outage, the Main Steam Stop Valves (MSSV's) be leak tested at 59 psig. The specified allowable leak rate for a MSSV is limited to less than 25 cubic ft. per hr. The acceptance criteria used in the performance test procedure only measured water leakage and did not correct the measured leakage for an equivalent steam/air leak rate. When test results are appropriately corrected, the leak rate would exceed 25 cubic ft. per hr. limit. A review of the past results for Units 1 and 3 indicated that the equivalent steam/air leak rate exceeded the allowable leak rate limit, thus failed its test. An operability evaluation of the MSSV was performed to determine the impact of this failure. The conclusion reached by the evaluation was that the failure of the Main Steam Stop Valves to satisfy the leak rate criteria of Technical Specification 4.8.2 has no impact on the capability of the MSSV's to satisfy their intended safety functions, thus the MSSV's are considered to be operable.

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Duke Engineers have performed a detailed review and evaluation of the design basis for the MSSV and the technical basis for the surveillance requirement. Based on this review, it was concluded that the performance of the leak rate test for the MSSV was unnecessary. Accordingly, I have agreed with my staffs recommendation to request Discretionary Enforcement Relief from the requirements of Technical Specification 4.8.2, so that this test will not have to be performed during the current Unit 2 refueling outage as well as all future outages for all three units. Attachment 1 to this letter provides the Technical Justification in support of this request and substantiates our conclusion that not performing this surveillance test will not pose an undue risk to the health and safety of the public. In addition, I have instructed my staff to prepare and submit, with appropriate technical justification, a technical specification amendment request to delete this surveillance requirement.

Should there be any questions or if additional information is required, please advise.

Very truly yours,

J. W. Hampton
for

H. B. Tucker

PFG.00/lcs

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Oconee Nuclear Station

Attachment

Duke Power Company
Oconee Nuclear Station

Discretionary Enforcement Relief
Technical Specification 4.8.2

DISCUSSION

Technical Specification 4.8.2 specified leakage testing of the Main Steam Stop Valves (MSSVs) at 59 psig during each refueling outage. The specification limits leakage to 25 cubic ft. per hr. The apparent bases for the leak rate is for the use of the MSSVs as containment isolation valves in the unlikely event of a simultaneous break of the reactor coolant line and a steam generator feedwater header. The technical bases for the allowable leakage of 25 cubic ft. per hr. is 25% of the total allowable containment leakage from all penetrations and isolation valves. The acceptance criteria used in performance test PT/O/A/0270/32 measures water leakage thru the valve seat. This leak rate is not corrected for an equivalent steam/air specific volume.

Through a detailed review and engineering evaluation, leak testing of the MSSV has been determined to be unnecessary in order to determine the operability of these valves and that the bases for the leak rate testing requirement specified by Specification 4.8.2 is inappropriate. This effort involved the following activities:

1) A review of past Licensing documents associated with the MSSV component, which included the following documents; Oconee Technical Specification and Amendments thereto; Oconee FSAR (original, supplements, revisions and annual updates); 10 CFR Appendix J; AEC Safety Evaluation Report for Oconee Unit 1 and for Units 2 and 3 (original).

2) Evaluation of the impact of the failure of the leak rate test on Design Basis events in order to determine the capability of the MSSVs to perform their intended safety function.

3) Evaluation of the impact on mechanical systems necessary for post-accident operations and on the Radiological release assumptions used in the Design Basis events for the plant.

The following discussion provides a brief chronology of the information found during the review of past Licensing documents:

A) May 25, 1989 FSAR Revision 5, Section 10.4 Pages 10-6 through 10-7

The safety function of the main steam stop valves is identified to be isolation of the unaffected steam generator in the event of a main steam line break, in order to prevent blowdown of both generators.

B) September 14, 1970 FSAR Revision 12, Supplement 8

Request 12, Pages 8-22 Through 8-23

In response to a request by the Division of Reactor Licensing, Duke provided the results of an analysis which shows that primary pipe whip will not cause failure of the secondary system. Duke performed a two part evaluation which first determined credible break locations and second evaluated the effects of each break. The feedwater header was

not explicitly identified, however the OTSG was evaluated with acceptable results. From this analysis, it can be assumed that the feedwater header would not be affected by any credible primary break location.

Request 7, Pages 8-12 Through 8-14

In response to a request by the Division of reactor Licensing, Duke provided the results of a reactor building pressure analysis for the rupture of one feedwater header simultaneous with a worst case LOCA. Although the hot leg is not free to whip, the analysis assumed it moved into feedwater piping. Peak pressure was 59.9 psig.

C) December 14, 1970 FSAR Revision 14, Section 15.4.8.2, Page 15-118

This revision provides Technical Specifications including main steam stop valve leak testing. This is the first record of this Technical Specification. No document identifying the technical basis for including this specification has been identified. This specification is virtually identical to the current specification 4.8.

D) December 29, 1970 AEC SER for Unit 1, Section 6.2, Page 35

In the discussion of reactor building structural design, the AEC determines that although the "peak pressure resulting from the combined blowdown of primary and secondary systems slightly exceeds the design pressure, it is acceptable because reactor coolant piping layout and restraints act to minimize the kinds and location of reactor coolant pipe breaks that could cause such a loss of the feedwater ring."

E) November 30, 1976; October 24, 1980; September 3, 1981 Duke Letters

Duke submits an amendment request for 10 CFR50, Appendix J Technical Specifications. Technical Specifications proposed for Table 4.4-1 include a note regarding leak testing of main steam stop valves, however Type C testing of penetrations 26 and 28 (main steam lines) is not required.

F) November 6, 1981; NRC letter

Amendment Nos. 104/104/101 are issued to incorporate the containment penetration testing requirements of Appendix J. Within the TER referenced in the NRC, SER, Section 3.1.6, (Penetrations 26 & 28) the NRC concurs with Duke's analysis that Appendix j does not require testing of the main steam lines. The main steam lines are part of a closed system which does not communicate with the reactor coolant pressure boundary or the containment atmosphere and is not liable to rupture as a result of the LOCA.

In summary, the review of the Licensing documents indicate that the safety function of the main steam stop valves is to close following a main steam line break and thus prevent blowdown of the unaffected steam generator. The scenario provided in the bases of Specification 4.8.2 (i.e. feedwater header rupture concurrent with LOCA) is beyond the design basis for Oconee. As such, leak

testing of the main steam stop valves is not necessary to meet the requirements of 10CFR50, Appendix J. Further, leak testing of the main steam stop valves is not required to demonstrate that operation of Oconee remains within the bounds of the accident analyses. Accordingly, the basis for the Technical Specification Surveillance requirement to leak test the MSSV is inappropriate and thus the test is unnecessary.

As discussed above, the event described in the bases for specification 4.8.2, (ie. concurrent breaks in the RCS and in the SG feedwater line), is beyond the standard Design Basis events and therefore should not be considered. The potential for an RCS line break to cause a SG feedwater line break was considered and evaluated during the initial licensing process for Oconee. The evaluation indicated that this accident scenario is not credible. The original environmental analysis does not address this scenario for the LOCA event. The feedwater lines and feedrings, as well as, the RCS piping, are seismically designed. The current FSAR environmental analysis does not consider this event nor its leakage for the LOCA event. In addition, the leakage criteria for the MSSVs to limit containment leakage for this event will provide little or no secondary side leak tightness due to the number of smaller drains and vents in the line up to the MSSVs. None of these smaller valves are tested for leakage to maintain containment leak tightness post-LOCA-feedwater line break. As such the bases for this specification is beyond design basis.

Main Steam Line Break accident event includes the use of the MSSVs to isolate the faulted steam line from the unfaulted steam line. In this way, over cooling of the RCS is minimized. Leakage of the MSSVs is inconsequential to this isolation safety function. The environmental analysis assumes the release of 1% equilibrium RCS inventory at 1 gal./min. for 3 hours. Therefore, MSSV leakage does not impact offsite dose.

Steam generator tube rupture accident event also includes the use of the MSSVs to isolate the faulted steam generator. This isolation function assures that RCS cooldown will be accomplished through the use of the unfaulted steam generator. Any leakage thru the MSSVs would be to the condenser and would result in negligible releases compared to activity released during the initial blowdown and long term cooldown.

Current design basis analysis bounds any leakage through the MSSVs that may occur. One of the assumptions identified in the analysis (See FSAR 15.13) is that the steam line break occurs between the reactor building and the MSSVs. This assumption is applicable to both the steam line break case and the steam line break concurrent with a steam generator tube rupture event case. The analysis assumes that the activity contained in the available inventory is released directly to the atmosphere through the break. As such leakage past the MSSVs is of no consequence. The analysis bounds the case of any leakage through the MSSVs that could possibly occur, whether the break is upstream or downstream of the MSSVs, provided that the valves stroke closed when required. This function (stroke closed) is periodically verified per specification 4.8.1.

Following a reactor trip, proper control of steam generator pressure and level is necessary to limit the primary system cooldown. The cooldown associated with an uncontrolled decrease in steam generator pressure could lead to a loss of shutdown margin. Normally, a reactor trip will trip the turbine which closes the MSSVs. After the stop valves close, steam generator pressure increases until the main steam relief valves open. The relief capacity of these valves is large enough to prevent the peak pressure from exceeding the steam generator/main steam

design pressure. Eventually, the steam load is reduced to a capacity that can be handled by the turbine bypass valves control steam generator pressure. The Chapter 15 accident analyses that result in a reactor trip take credit for the above described steam generator pressure response. The post-trip primary system temperature stabilizes at approximately the saturation temperature of the steam generators. Leakage past the MSSVs would have no effect on the minimum post-trip primary system temperature. Thus, this leakage would not cause a loss of shutdown margin due to an excessive cooldown.

All Chapter 15 accidents events have been reviewed to determine impact of additional MSSV leakage on the Design Basis results. No impact has been found for mechanical systems and equipment and no impact is found on any effluent release analysis.

CONCLUSION

The failure of the Main Steam Stop Valves to satisfy Technical Specification 4.8.2 leakage criteria has no impact on the capability of the MSSVs to satisfy the intended safety functions for Design Basis Events. Therefore, the MSSVs are considered operable, and further leak rate testing as described in Technical Specification 4.8.2 is unnecessary.