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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

TAC NO. M82258

1.0 INTRODUCTION

By letter dated December 5, 1991 (with enclosure GENE-508-014-1191, GENE-523-133-1191 and GENE-523-134-1191 Rev 1), and supplemented by letter dated December 20, 1991 (with enclosure GENE-523-133-1191 Rev 1), the licensee submitted for staff review and approval a fracture mechanics evaluation of flaw indications found during the Fall 1991 Outage through Ultrasonic (UT) examination in Cooper Nuclear Station's (CNS) feedwater nozzle-to-vessel welds. The intent of this submittal is to demonstrate that the feedwater nozzle welds of CNS, although containing flaw indications exceeding the prescribed acceptance criteria for ASME Code Category B-D welds, is suitable for continued operation without repair for 40 years or 120 startup and shutdown cycles coupled with 600 thermal/pressure cycles.

2.0 EVALUATION

During the fall 1991 in-service inspection of CNS feedwater nozzles, UT examination of welds revealed a total of seventeen flaw indications that exceeded Table IWB-3510-1 (ASME Code, Section XI) requirements; among them, nine were for Weld N4A, two for Weld N4C, and six for Weld N4D. The indications, most of them located close to the weld mid-plane, exhibited the characteristics of subsurface cracking, which could be attributed mostly to hot cracking during fabrication of the weld. Another mechanism due to thermal fatigue growth caused by feedwater leakage through the thermal sleeve of the feedwater nozzle may be a secondary one because no indications were found in the nozzle bore or inner radius zones.

The licensee performed a fracture mechanics analysis consistent with the procedures outlined in Section XI, ASME Code, 1989 Edition. The RT_{NDT} of the weld material needed for toughness determination was established to be 18°F following the guidelines in NRC Branch Technical Position MTEB 5-2. Due to licensee's citing of wrong paragraph from MTEB 5-2 and absence of existing Charpy V-notch (CVN) test data in a telephone conversation on

December 12, 1991, the staff demanded the fracture mechanics analysis be revised to reflect weld RT_{NDT} of 30°F. After reviewing the supplemental submittal, GENE-523-133-1191 Rev 1 with CVN test data, the staff now accepts the RT_{NDT} of 18°F.

The licensee considered all thermal and pressure transients occurring in the vessel and nozzle and selected the worst case - the hydrotest transient for the fracture mechanics analysis. Results from the analysis showed that the applied stress intensity factor K for code indications (100% signal amplitude level per ASME Code Criteria) was 28 $ksi(in)^{1/2}$ and the applied K for non-code indications (50%/20% signal amplitude levels per Reg Guide 1.150) was 47 $ksi(in)^{1/2}$. Since both are smaller than the available toughness of 63 $ksi(in)^{1/2}$, and meet the criteria of IWB-3612, Section XI, ASME Code, they are acceptable.

According to the criteria in IWB-3600 of ASME Code Section XI, 1989, the reactor pressure vessel is acceptable for service without excavation and repair of the flaw if the fracture mechanics analysis indicates the flaw will not exceed 0.6t (t = thickness of the nozzle wall). The licensee performed a fracture mechanics growth analysis of the limiting flaw (# 18) in the N4D nozzle weld. The analysis indicates that the flaw will grow from a depth of .22t to a depth of .225t for 40 years or 120 startup and shutdown cycles coupled with 600 thermal/pressure cycles. Similar negligible end of life crack growth of 0.02 inch for non-code indications is documented in GENE-523-134-1191, Rev 1. Both reports meet the criteria and are acceptable to the staff.

Thermal fatigue caused by feedwater nozzle bypass leakage was not considered in the fracture mechanics analysis because: (a) no indications were found in the nozzle bore or inner radius zones during automated Ultrasonic Testing of the four feedwater nozzles in the recent fall outage; (b) a new flow leakage measuring system was installed in this fall outage which will give on-line indication about the severity of the thermal fatigue due to the leakage of cold feedwater into the vessel during startup and shutdown and during hot standby conditions. Since the staff's approval of this submittal is based on the assumption of no bypass leakage, the licensee should report to the staff about any nozzle leakage exceeding 0.3 gpm (Ref 6).

The licensee plans to reinspect the feedwater nozzle indications in the spring 1996 refueling outage, about four years from now. This is only one more year than the normal inspection interval and is acceptable to the staff provided the newly installed flow leakage measuring system functions properly and with no readings exceeding 0.3 gpm during operation.

3.0 CONCLUSION

The staff conclude that the submittal is acceptable based on the following reasons: (a) the licensee considered all thermal and pressure transients occurring in the vessel and nozzle and selected the worst case - the hydrotest transient for the fracture mechanics analysis; (b) the available toughness of 63 $ksi(in)^{1/2}$ is larger than the applied K of 28 $ksi(in)^{1/2}$ for code indications and 47 $ksi(in)^{1/2}$ for non-code indications; (c) the fatigue crack growth for both indications was predicted to be less than 0.02 inch at the end of plant life; (d) a new flow leakage measuring system was installed in this fall outage which will give on-line indication about the severity of the thermal fatigue due to the leakage of cold feedwater into the vessel during startup and shutdown and during hot standby conditions.

Since the staff's approval of this submittal is based on the assumption of no bypass leakage, the licensee should report to the staff about any nozzle leakage exceeding 0.3 gpm during operation.

4.0 REFERENCES

1.0 GENE-508-014-1191, "Analysis of UT Indications in Feedwater Nozzle-to Vessel Welds N4A, N4C, and N4D" GE Nuclear Energy Division, November 26, 1991 (proprietary).

2.0 GENE-523-133-1191, "Fracture Mechanics Evaluation of UT Indications Found in the Cooper Feedwater Nozzle to Shell Weld" GE Nuclear Energy Division, November 11, 1991 (proprietary).

3.0 GENE-523-134-1191, Rev 1, "Fracture Mechanics Evaluation of UT Indications Found per Reg. Guide 1.150 in the Cooper Feedwater Nozzle to Shell Weld" GE Nuclear Energy Division, November 20, 1991 (proprietary).

4.0 Letter from G R. Horn (NPPD) to NRC, "Responses to NRC's Request for Additional Information Made in a Telephone Conversation on December 12, 1991" December 20, 1991.

5.0 GENE-523-133-1191, Rev 1, "Fracture Mechanics Evaluation of UT Indications Found in the Cooper Feedwater Nozzle to Shell Weld" GE Nuclear Energy Division, December 13, 1991 (proprietary).

6.0 NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" November 1980.

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