UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR VESSEL FLAW EVALUATION

VERMONT YANKEE NUCLEAR POWER STATION

MATERIALS AND CHEMICAL ENGINEERING BRANCH

DIVISION OF ENGINEERING

1.0 INTRODUCTION

By letter dated October 9, 1996, the licensee submitted a flaw evaluation report for NRC review and approval. The report contains the licensee's evaluation of a flaw indication in the reactor pressure vessel (RPV) that exceeded the allowable flaw sizes in IWB-3500 of the American Society of Mechanical Engineers (ASME) Code, Section XI. Ultrasonic examinations (UT) of the RPV were performed in accordance with the requirements of the ASME Code, Section XI, 1986 Edition. The UT examinations resulted in one flaw, out of a total of seven, that was not acceptable by IWB-3500 acceptance standards.

Any indications that are found during an inspection must be characterized. The characterized flaw is then compared to the acceptance standards in IWB-3500. A flaw that exceeds the allowable flaw sizes that are defined in IWB-3500 may be evaluated by analytical procedures. Appendix A of Section XI provides methodology that may be used for detailed fracture mechanics evaluation. The flaw is acceptable for continued operation if it meets the criteria of IWB-3600.

2.0 EVALUATION

The flaw was found in Plate 1-15, below the circumferential weld that joins Plates 1-12 and 1-15. However, since the subsurface flaw is located in Plate 1-15 near the circumferential weld, it was evaluated by assuming that its location is within the weld metal as well as within the plate. Plate 1-15 extends into the beltline region, however, the flaw is outside of the core region and is circumferential in orientation. The allowable flaw depths were determined using Appendix A of the ASME Code. The flaw size was limited in all cases to the ASME Code proximity rules for subsurface flaws, where flaws greater than the limiting end-of-license (EOL) flaw would have to be evaluated as a surface flaw.

The licensee reported an initial RT_{NDT} (IRT) value of 30°F for Plate 1-15 (heat C3116-2). The initial RT_{NDT} for the weld was also assumed to be 30°F. In a supplement to the submittal dated October 11, 1996, the licensee stated that the weld was fabricated using the shielded metal arc process by Chicago Bridge and Iron Company (CB&I). In a letter dated September 24, 1993, the licensee reported

ENCLOSURE

a generic value for shield metal arc welds fabricated by CB&I to be -70°F. Since an initial RT_{NDT} of -70°F is less than the value assumed for the weld in the licensee's analysis, the assumed initial RT_{NDT} is conservative. The staff verified that the IRT values are consistent with the reactor vessel integrity database (RVID). Since the location is not in the beltline, there will be no increase in RT_{NDT} resulting from neutron irradiation.

After consideration of the original RPV design basis transients, bounding load cases were selected for the flaw evaluation. Two transients were in the original design specification, and two were added specifically for the flaw evaluation. These transients were determined to be the most severe for the RPV for all operating conditions (normal/upset and emergency/faulted). The load cases considered were: 1) Hydrotest of the RPV at 1100 psig, 2) Heatup to 545°F from 100°F at a rate of 100°F/hr, 3) Improper start of recirculation loop, and 4) Reactor blowdown with a rapid cooldown from operating temperature (545°F) to 370°F in 10 minutes followed by a further reduction in temperature at a rate of 100°F/hr. The hydrotest and the improper start of the recirculation loop were the two transients that were added for the evaluation.

The improper start of the recirculation loop was determined to be the limiting operating condition. Although the blowdown transient has higher thermal stresses, the limiting event is the improper start of the recirculation loop because it has the highest overall calculated stresses (thermal and pressure stresses combined). Seismic loads were not evaluated since they are insignificant in reactor vessel shell regions away from reactor supports. All transients were evaluated to the criteria for normal and upset conditions in IWB-3600, including the appropriate applied safety factor of 3.162.

The licensee calculated the allowable flaw depths and the stress intensity factors using the methodologies provided in Appendix A of the ASME Code, Section XI. The current observed flaw depth (2a) is 0.35 inches. The current allowable flaw depths for the weld and plate are 1.7606 and 1.7622 respectively. The slight difference in value is due to weld residual stresses.

For the consideration of flaw growth, 35 heatups/cooldowns were assumed to occur in every 6 year interval. Since, 16 years remain for the plant's license, the flaw evaluation conservatively considered 105 remaining heatups and 105 rapid cooldowns. In addition, the evaluation considered 105 hydrotests and 105 improper starts of a recirculation system pump. The flaw growth for the weld and plate was calculated to be 0.0034 and 0.0018 respectively. Considering the flaw growth, the predicted EOL flaw depths (2a) for the weld and plate are 0.3534 and 0.3518 respectively. Since the allowable depths at EOL for both the plate and the weld (1.764 inches) are larger than the predicted values, the licensee has demonstrated that the flaws meet the criteria of IWB-3600.

The staff has verified the licensee's methodology and calculation of allowable flaw depths during review of the flaw evaluation. The staff finds the results to be acceptable to support the safe operation of Vermont Yankee through EOL without repair to subject plate.

3.0 CONCLUSION

The licensee's evaluation indicates that the flaw satisfies the ASME Code criteria in IWB-3600. Since the licensee has met the ASME Code criteria, the staff concludes that the structural integrity of the reactor pressure vessel will be maintained, and that the flaw is acceptable for continued operation through end-of-license without repair.

Principal Contributor: A. Lee

Dated: October 11, 1996

REFERENCES

a. ASME Boiler and Pressure Vessel Code, Section XI, 1986 Edition

b. Letter to USNRC Document Control Desk from J. J. Duffy (VYNPC) Subject: "Reactor Pressure Vessel Inspection at Vermont Yankee, NRC Docket No. 50-271, License No. DPR-28" October 9, 1996.

c. Reactor Vessel Integrity Database (RVID), U.S. Nuclear Regulatory Commission

d. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988