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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

VERMONT YANKEE NUCLEAR POWER STATION

JET PUMP RISER INSPECTION RESULTS AND THE FLAW EVALUATION

VERMONT YANKEE NUCLEAR POWER CORPORATION

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1.0 INTRODUCTION

By letter dated May 4, 1998, Vermont Yankee Nuclear Power Corporation, (the licensee) submitted for NRC review, its jet pump riser (JPR) circumferential weld inspection results and the associated flaw evaluation for the detected flaws. An ultrasonic (UT) inspection of 26 of the 30 circumferential welds was conducted during the 1998 refueling outage in accordance with the guidelines of report BWRVIP-41. A visual (VT-1) examination of the remaining four welds, which could not be inspected by UT due to interferences, was also performed. The inspection results indicate that four flaws exist in the circumferential weld connecting thermal sleeve to riser elbow (RS-1 weld). The maximum flaw size is 2.82 inches. The licensee intended to demonstrate through an analytical flaw evaluation that the unit could be operated without repair for one fuel cycle.

2.0 EVALUATION

2.1 Licensee

Based on the maximum flaw size of 2.82 inches for the detected flaws in the RS-1 welds and a flaw measurement uncertainty of 0.191 per flaw end, the licensee calculated the initial flaw size to be 3.20 inches. The licensee then assumed that the flaw was through-wall and performed a flaw evaluation using the limit load analysis consistent with the latest Appendix C (1996 Addenda) of Section XI of the American Society of Mechanical Engineers (ASME) Code (the Code). The limit load analysis used a safety factor of 2.77 for Normal and Upset and 1.39 for Emergency and Faulted conditions, a Z-factor for submerged arc welds (SAW), and a bounding intergranular stress corrosion cracking (IGSCC) crack growth rate of $5X10^{-5}$ inch/hour. The normal load includes dead weight, hydraulic loads, flow induced vibration (FIV), and thermal loads. The faulted load includes the normal load plus the safe shutdown earthquake inertia load. The fatigue crack growth due to FIV under the normal condition has also been considered and determined to be insignificant.

The licensee added the crack growth corresponding to one fuel cycle, or 12,000 hours of operation (0.60 inch per crack end per cycle), to the initial crack size (3.20 inches), and obtained the final crack size (4.40 inches). On the other hand, the licensee calculated the allowable crack size using limit load analysis and found its value to be 16.41 inches. Since the

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predicted crack size at the end of one fuel cycle is less than the allowable crack size with an adequate margin, the licensee concluded that JPR integrity and performance will not be compromised for the next fuel cycle.

2.2 NRC Staff

2.2.1 Limit Load Analysis of the 1996 Addenda

The licensee performed a flaw evaluation for the detected flaw at the RS-1 weld of JPRs. The flaw evaluation was based on the limit load analysis of the latest Appendix C (1996 Addenda) of Section XI of the ASME Code. Previously, the Code required the user to set the pipe diameter in the Z-factor equations to be 24 inches for any pipe having a diameter less than 24 inches. The 1996 Addenda has removed this additional conservatism based on recent results from extensive pipe fracture experiments on austenitic SAW and SMAWs. The staff accepts the limit load analysis of the 1996 Addenda because it is supported by numerous test data documented in NUREG/CR-4878. Further, since the 1996 Addenda only involves a minor modification in Z-factor calculation while keeping the main body of the limit load analysis intact, the staff considers the limit load analysis of the1996 Addenda to be essentially the same as that in the current Code and is acceptable in this application.

2.2.2 Flaw Evaluation

The staff evaluated the licensee's allowable crack size evaluation and determined that the limit load analysis meets the rules of the ASME Code (1996 Addenda), and therefore is acceptable. As to the predicted flaw size estimation, the staff determined that the use of the bounding IGSCC growth rate is conservative, and the use of FIV for the fatigue crack growth calculation is adequate. The UT measurement uncertainty of 0.191 inch for each crack end was determined in accordance with BWRVIP-03, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examinations Guidelines." This report was approved by the NRC on June 8, 1998. This UT technique was partially demonstrated at the EPRI NDE Center in 1997. The demonstration was completed at Peach Bottom Atomic Power Station in the same year. The staff examined the margin (Margin 1) between the allowable flaw size (16.41 inches) and the predicted flaw size (4.40 inches) and the margin (Margin 2) between the allowable flaw size for FIV (5.4 inches) and the predicted flaw size (4.40 inches), and found both margins are acceptable. Margin 1 is used to justify the continued operation of the unit for one fuel cycle with the detected flaws in the JPR welds. Margin 2 is used to support that FIV contributes to no fatigue growth during the intended fuel cycle. It should be noted that "the allowable flaw size for FIV" is not a universally accepted terminology. It is defined in the submittal as the flaw size beyond which the applied stress intensity factor difference (ΔK) would be large enough to contribute to fatigue crack growth.

3.0 CONCLUSIONS

The staff has determined that the flaw evaluation meets the rules of the ASME Code and the assumed crack growth rate is appropriate for this application. Since the predicted final flaw size at the end of one cycle (4.40 inches) is far less than the allowable flaw size (16.41 inches) from the limit load analysis, the staff determines that continued operation for Vermont Yankee without repair is acceptable for one fuel cycle. The licensee plans to reinspect these flaws during the next refueling outage and to further evaluate the flaws at that time.

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