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

RELATED TO THE FLAW EVALUATION OF THE CORE SPRAY INTERNAL PIPING

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES, UNIT 1

DOCKET NO. 50-254

1.0 INTRODUCTION

During the current Quad Cities, Unit 1, refueling outage (Q1R14), crack-like indications were visually observed at two components of the core spray internal downcomer piping. The two flawed components are a lower sparger inlet elbow in "B" loop, and an upper sparger inlet elbow in "A" loop. All indications were located in the heat affected zones (HAZ) adjacent to the thermal sleeve to elbow welds. The flawed elbows were made of Type 304 stainless steel and were located inside the vessel annulus between the inside wall of the reactor pressure vessel and the outside wall of the core shroud. The subject elbows are six inches in diameter. Each end of the elbow was welded to the thermal sleeve and the downcomer piping, respectively. The length of these indications, as measured by the visual and ultrasonic examinations, varied from 5.6 inches to 7.0 inches. The locations and appearance of these crack indications are typical of intergranular stress corrosion cracking (IGSCC).

By a letter dated April 2, 1996, the licensee submitted a flaw evaluation report of the core spray internal piping for NRC review and approval. The staff held several conference calls with the licensee (April 1, 4 and 9, 1996) to discuss the licensee's flaw evaluation. The staff requested that the licensee provide the results of an additional flaw evaluation. The results of the licensee's evaluations demonstrated that sufficient margins exist to operate for one additional cycle with the identified flaws. The staff's evaluation and conclusion are provided below.

ENCLOSURE

2.0 EVALUATION

Because IGSCC is known to be initiated from the piping inside surface, visual examination can only find flaws that are through-wall. To ensure all flaws, whether they are through-wall or not, are found and properly sized, the licensee performed ultrasonic examination of each of the flawed core spray components. Since the pipe wall is relatively thin, it is not practical to determine the depth of the flaws and, therefore, only the length of each flaw was ultrasonically determined. Thus, in the licensee's flaw evaluation, each flaw was assumed to be through-wall. The ultrasonic technique used in the examination was developed by General Electric Company (GE) to detect and measure the length of the flaws. The technique was previously qualified on mockups of flawed piping components at Dresden Nuclear Power Station during the D2R14 outage, and was independently reviewed by EPRI and the licensee. A "Smart 2000" ultrasonic data acquisition system was used in the ultrasonic examination. The length of the flaw at the lower sparger inlet elbow in loop B and the upper sparger inlet elbow in loop A was reported to be 5.6 inches and seven inches, respectively. The reported flaw length corresponds to the maximum length determined by the visual and ultrasonic examinations. Due to the access limitation, a portion of the circumference (about 3.5 inches) in the upper sparger inlet elbow was not ultrasonically examined. However, visual examination did not find any crack indication in this area.

The licensee reported that, based on the fabrication records, the subject thermal sleeve-to-elbow welds were performed using the gas tungsten arc welding (GTAW) process. The sparger inlet elbows are seam welded elbows (six inches in diameter) and were made from solution heat treated Type 304 austenitic stainless steel (schedule 40).

In the crack growth calculation, the licensee used the bounding crack growth rate of $5.0X10^{-5}$ inches/hour. The licensee stated that hydrogen water chemistry (HWC) has been implemented at Quad Cities, Unit 1, since 1990 to mitigate the IGSCC. The licensee also stated that the neutron fluence in the area of the core spray internal piping is less than $6.0x^{+18}$ n/cm^2. Because the neutron fluence is less than the threshold level of $5.0x10^{+20}$ n/cm^2, irradiation assisted stress corrosion cracking (IASCC) is not expected to occur at the subject core spray piping. Based on the consideration discussed above, the staff concludes that the crack growth rate used by the licensee in the crack growth calculation is consistent with the staff's guidelines and is, therefore, acceptable.

By using the bounding crack growth rate, the licensee calculated the final crack length at the end of the next fuel cycle for a period of 24 months (17,280 hours). The final crack length was determined by adding 0.68 inches to each end of the detected flaw.

The piping models used in the piping stress analyses are based on the design basis drawings. Two piping models representing the affected portions of the upper and lower core spray sparger system, respectively, were constructed for the piping analysis. The stiffnesses of the penetration assembly were derived from a finite element analysis. The PIPSYS program was used to calculate the loads and stresses in the piping system. The loads used for the elbow flaw evaluation were taken directly from the piping analysis.

The licensee performed the flaw evaluation by using the limit load methodology in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix C. The ASME Code allows the limit load approach for welds fabricated by the GTAW process. The loads used in the evaluation were obtained from the piping analysis. The following loads were included in the evaluation: weight, thermal, seismic, operating drag and loss-of-coolant accident (LOCA). The design basis load combinations were evaluated and the worst case of normal/upset and emergency/faulted condition load combinations were used in the evaluations. Additionally, the licensee performed evaluations of cases beyond the design basis faulted condition. The licensee calculated the load design margins and the allowable months of operation to reach the critical flaw length for each of these cases. The load design margin is defined as the ratio of the maximum permitted stress to the applied stress. The ratio represents the margin with respect to the applied load above the ASME Section XI safety factors. The bounding case beyond the design basis was determined to be a simultaneous occurrence of a seismic safe-shutdown earthquake (SSE) event and a reactor recirculation line break (RRLB) LOCA. The licensee has determined that the loads generated by the RRLB LOCA event bounded that by the main steam line break (MSLB) LOCA event for this piping.

The flaw in the upper sparger inlet elbow for the beyond design basis case represents the bounding case in the flaw evaluation. The results of the licensee's limit load analysis have shown that the bounding final flaw length at the end of the next fuel cycle would not exceed the critical flaw length, and that the load margin factor for the bounding design basis condition and the beyond design basis condition is at least 17 and 11, respectively. By using a bounding crack growth rate of 5×10^{-5} inch/hour, the licensee's flaw evaluation showed that it would take at least 114 months of hot operation to reach the critical flaw length for the bounding case.

The licensee also performed simplified elastic-plastic evaluation for the flawed elbows in accordance with ASME, Section XI, Appendix C. In this evaluation, a reduction factor, Z, was assumed to be unity, and the secondary stresses were included in the limit load formulation. The results of the evaluation showed that for the bounding case, it would take at least 71 months of hot operation to reach the critical flaw length with a load margin factor of 2.35.

At the staff's request, the licensee also performed a bounding flaw evaluation by assuming the thermal sleeve-to-elbow welds to be SMAW welds and, in addition, the areas (3.5 inches) in the upper sparger inlet elbow that were inaccessible to ultrasonic examination were assumed to be flawed through-wall. The licensee informed the staff in the conference calls that in the bounding case, it would take at least 39 months of hot operation to reach the critical flaw length with a load margin factor of 2.33.

The staff has reviewed the licensee's flaw evaluation and concludes that the licensee's method of evaluation is technically sound and complies with the ASME Code requirements. Therefore, the flaw evaluation results are acceptable.

The licensee performed leak rate calculations for the flawed elbows by using the PICEP program. The PICEP program was developed by EPRI for Leak-Before-Break applications. The leak rate was calculated for several core spray piping conditions. The bounding condition with respect to the peak cladding temperature (PCT) is a core spray internal piping differential pressure of 47 psid with a design basis rated flow of 4600 gpm. The leak rate for this bounding condition was calculated to be no more than 5 gpm at the end of the next fuel cycle and 62 gpm at the end of plant life. The calculated leakage was assumed to be not available for core cooling in this evaluation for postulated reactor recirculation suction line break. For a core spray leakage of 400 gpm, the licensee's estimate of the PCT increase was 40 degrees Fahrenheit. The licensee also calculated the leakage from the core spray T-Box weld flaw repair, core spray line weld flaws in the T-Box, core spray slip fit thermal sleeve-to-nozzle safe end, vent hole in core spray line T-Box, and purge hole in the nozzle thermal sleeve. The combined leakage from these locations was estimated to be less than 237 gpm in the design basis LOCA scenario. Adding on another 5 gpm from the elbow flaws, the total leakage is still well within the 400 gpm. Since the existing PCT is only 1775 degrees Fahrenheit, even with the postulated 40 degrees Fahrenheit increase, the resulting PCT is still well within the regulatory limit of 2200 degrees Fahrenheit per 10 CFR 50.46. Therefore, the licensee concluded that the calculated leakage at the end of the next fuel cycle is well within the design basis margin and its impact on the PCT is insignificant. Since the IGSCC cracks were generally very tight, the staff expects the leakage flow from the flawed elbows to be small during the next fuel cycle with no significant impact on the PCT. Therefore, the staff has determined that the licensee's conclusion regarding the impact of the potential leakage is acceptable for the short-term operation of the next fuel cycle.

The licensee performed a safety evaluation of the loose parts which may result from the flawed core spray elbows. The postulated loose parts consisted of a separated stainless steel elbow and its debris. The safety evaluation assessed its potential impact upon the fuel bundle flow blockage and consequent fuel damage, fretting wear of the fuel cladding, interference with control rod operation and corrosion or chemical reaction with other reactor materials. The licensee's evaluation concluded that the postulated loose parts would not result in any safety concern in maintaining the proper fuel cooling and control rod operation. Although extensive IGSCC may lead to the separation of pieces of various sizes from the flawed elbows, in the short term, the staff does not anticipate any loose parts to occur, especially, the separation of the elbow. To ensure safe plant operation in the longer term, the staff recommends that the licensee consider implementing a program to enhance the plant capabilities in the detection of the loose parts during operation and a program for removing the loose parts from the reactor pressure vessel.

3.0 CONCLUSION

Based on the staff's review of the licensee's flaw evaluations, the staff concludes that the structural integrity of the flawed core spray elbows will be maintained during the next fuel cycle on the basis that the final flaw sizes at the end of the next fuel cycle will not exceed the Code allowable values. Therefore, Quad Cities, Unit 1, can be safely operated for the next fuel cycle without repairing the subject flawed core spray elbows. However, continued plant operation beyond the next fuel cycle will depend on the findings from the next reinspection of the core spray piping or by implementing acceptable repairs during the next refueling outage.

4.0 REFERENCE

Letter from E. S. Kraft, Jr., Commonwealth Edison Company, to U.S. Nuclear Regulatory Commission, "Quad Cities Unit 1 Assessment of Identified Core Spray Piping Flaws," dated April 2, 1996.