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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE FLAW EVALUATION OF THE CORE SPRAY INTERNAL DOWNCOMER PIPING

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-237

1.0 INTRODUCTION

During the current Dresden, Unit 2, refueling outage (D2R14), crack like indications were visually observed at three components of the core spray internal downcomer piping. The three flawed components are a "B" loop lower sparger inlet elbow, and an upper ("A" loop) and a lower ("B" loop) sparger inlet thermal sleeve collars. All indications were located in the heat affected zones (HAZ) of welds. The flawed piping components 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 elbow is 6 inches in diameter. Each end of the elbow was welded to the thermal sleeve and the downcomer piping, respectively. The thermal sleeve collar was attached to the outside surface of the core shroud at one end and on the outside surface of the thermal sleeve at the other end. The length of these indications as measured by ultrasonic examination varied from 2 inches to 5.5 inches. The crack indications were reported to be very tight and showed characteristics of jagging and branching. The locations and appearance of these crack indications are typical of intergranular stress corrosion cracking (IGSCC).

By a letter dated September 12, 1995, the licensee submitted flaw evaluation reports of the core spray internal piping for NRC review and approval. The revised flaw evaluation reports were submitted to NRC on September 25, 1995. The revised evaluation reports did not change the conclusions of the previous reports. The results of the licensee's evaluations concluded that sufficient margins exist to operate for one cycle with the identified flaws. The staff's evaluation and conclusion are provided below.

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. Because 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 determine the end points of the detected flaws. The technique was qualified on the mockups of the subject flawed piping components and was independently reviewed by EPRI and the licensee. For the thermal sleeve collars, the UT examination covered 360 degrees of the circumference. The flaw at the upper thermal sleeve collar in loop A was reported to be 2 inches in length. Two flaws were found at the lower thermal sleeve collar in loop B. One of the flaws was not visually observable because it was not connected to the outside surface of the collar. The lengths of the two flaws were reported to be 3 inches and 5.5 inches, respectively. The flaw at the lower sparger inlet elbow in loop B was estimated to be 3.5 inches in length. Due to access limitation, a portion of the elbow circumference (about 4.8 inches) 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 elbow weld was performed using the gas tungsten arc welding (GTAW) process and that the thermal sleeve collar welds were fabricated with the shielded metal arc welding (SMAW) process.

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) was implemented at Dresden, Unit 2, since 1983 to mitigate the IGSCC. The licensee also stated that the neutron fluence in the area of the core spray is less than $6.0x10^{+18}$ 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 conservative.

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 21 months with a 90 percent availability factor (13,608 hours). The final crack length was derived by adding 0.68 inches to each end of the detected flaw.

To develop the loads acting on the thermal sleeve collar flaws, the licensee performed a three dimensional finite element analysis (FEA) by using the ADINA program to model and analyze the core spray thermal sleeve shroud penetration assembly. The results of the FEA (stiffness of the penetration assembly and the load distribution) were used in the PIPSYS program 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 the 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 assessed the load design margins and the allowable months of operation 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 Code, Section XI, safety factors. The bounding case beyond the design basis was determined to be a simultaneous occurrence of a seismic SSE event and a reactor recirculation line break (RRLB) LOCA. The licensee has determined that the loads generated by the RRLB LOCA event are bounded by the main steam line break (MSLB) LOCA event for this piping.

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 38 and 28, respectively.

The licensee also performed simplified elastic-plastic evaluation for the SMAW welds in accordance with ASME Code, Section XI, Appendix C. The welds at the thermal sleeve collars were fabricated by the SMAW process. In this evaluation, a reduction factor (Z) and the secondary stresses were included in the limit load formulation. At the staff's request, this evaluation was also performed for the elbow weld. In addition, the elbow areas (4.8 inches) that were inaccessible to ultrasonic examination were assumed to be flawed through-wall in this evaluation. The results of the licensee's evaluation showed that the flawed elbow for the condition beyond the design basis represented the bounding case. For the bounding case, the load margin factor was reported to be 1.8. The staff has reviewed the licensee's flaw evaluation and concludes that the licensee's method of evaluation is conservative and complies with the ASME Code requirements and, therefore, the evaluation results are acceptable.

The licensee performed a leak rate calculation for the flawed elbow by using the PICEP program. The thermal sleeve collars are not part of the core spray system pressure boundary and, therefore, are not considered in the core spray system leakage evaluation. The PICEP program was developed by EPRI for leak-before-break applications. The leak rate was calculated for several piping conditions. For the bounding condition of a 64 psig line pressure in the core spray piping with the reactor vessel pressure at a zero psig, the leak rate was calculated to be no more than 1.38 gpm at the end of next fuel cycle and 82.84 gpm at the end of the plant life. The leakage was considered lost in this evaluation as a reactor recirculation suction line break was assumed. The licensee stated that with a concurrent loss of the low pressure coolant injection (LPCI) system, the leakage may impact the peak cladding temperature (PCT). For a core spray leakage of 300 gpm, the licensee's preliminary estimate of the PCT increase is 36 degrees Fahrenheit. 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 detected cracks were reported to be very tight, the staff expects the leakage flow resulting from the flawed elbow to be small during the next fuel cycle with no significant impact on the PCT. Therefore, the licensee's conclusion 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 components. The postulated loose parts consisted of a separated stainless steel elbow and its debris. The safety evaluation considered its potential impact for 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 the control rod operation. Although extensive IGSCC may lead to the separation of pieces of various sizes from the flawed components, in the short term, the staff does not anticipate any loose parts to occur; especially the separation of the elbow. However, to ensure safe plant operation in the longer term, the staff recommends that the licensee submit an evaluation prior to the end of the next refueling outage to address the plant capabilities in the detection of the loose parts during operation and the 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 subject flawed core spray components 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 ASME Code allowable values. Therefore, Dresden, Unit 2, can be safely operated for the next fuel cycle without repairing the subject flawed core spray components. However, continued plant operation beyond the next fuel cycle will depend on the satisfactory evaluation of the re-inspection results or by implementing acceptable repairs during the next refueling outage.

Principle Contributor: Bill Koo

Date: February 23, 1996

4.0 REFERENCES

Letter, Peter L. Piet, Commonwealth Edison Company, to U.S. Nuclear Regulatory Commission, September 12, 1995.

Letter, Peter L. Piet, Commonwealth Edison Company, to U.S. Nuclear Regulatory Commission, September 25, 1995.

the next fuel cycle should be supported by the results of re-inspection and reevaluation of the subject flaw indications. In addition, to ensure safe plant operation in the long-term, please provide an evaluation to address the plant capabilities in the detection of loose parts during power operation and the program for removing loose parts from the reactor vessel. This evaluation should be provided for staff review prior to restart of the unit from the next scheduled refueling outage.

This completes the NRC staff review of the subject evaluation and closes TAC No. M93590. If you have any questions regarding this issue, please contact me at (301) 415-1345.

Sincerely,

/s/

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-237

Enclosure: Safety Evaluation

cc w/encl: see next page

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