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

FLAW INDICATIONS IN A REACTOR PRESSURE VESSEL (TAC 67960)

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT - UNIT 1

DOCKET NO. 50-348

1.0 INTRODUCTION

During the 1988 refueling outage, Alabama Power Company (the licensee) performed an inservice inspection of several welds in the reactor pressure vessel in the Joseph M. Farley Nuclear Plant, Unit 1. As a result of conventional ultrasonic testing of the welds, four flaw indications were reported to exceed the allowable size criteria in Article IWB-3500 of ASME Code, Section XI. Article IWB-3600 establishes rules for evaluating flaw indications that exceed the acceptance criteria in Article IWB-3500. Paragraph IWB-3610 states that the evaluation procedures shall be the responsibility of the owner and shall be subject to approval by the regulatory authority having jurisdiction at the plant site. In letters dated April 28 and May 5, 1988, the licensee submitted, for staff review a technical evaluation of the four flaws. In addition, the licensee's letters included a description of a fifth indication that was found during a previous inservice inspection and was determined to be acceptable during the 1988 ultrasonic examination. The staff also considered the licensee's letter dated October 26, 1983, submitted in response to Generic Letter 83-15, which implemented Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

2.0 EVALUATION

A. Inspection Interval

During the 1988 refueling outage, the licensee conducted scheduled first and second interval inservice inspections (ISI) of certain reactor vessel welds. First interval ISI was performed on the longitudinal and circumferential shell welds, inlet nozzle-to-shell welds, flange-to-shell weld and inlet nozzle safe-end welds and was based on the requirements in ASME Code, Section XI, 1974 Edition, including addenda through Summer 1975 (74E75S). Second interval ISI was performed on the outlet nozzle-to-shell welds, outlet nozzle safe-end welds, flange-to-shell weld and the flange ligaments and was based on the requirements in ASME Code, Section XI, 1983 Edition, including addenda through Summer 1983 (83E83S).

Paragraph IWA-2400(a) of these ASME Code editions and addenda permit extension of the inspection interval such that examination can be performed concurrent with the plant outage. The licensee has exercised this option for the initial inspection interval.

B. Nondestructive Examination

An NRC Inspector from Region II was at the plant site during the reactor vessel inspection to evaluate the nondestructive examination techniques and to observe inspections in progress. Conclusions will be documented in Inspection Report 88-14. Region II personnel also reviewed the inspection results contained in the licensee's letters.

After detection, the flaw indications that exceed the recording criteria established by ASME Code, Section XI, are evaluated further to determine the location, size, and characteristics of the reflectors. The conventional ultrasonic examination identified two indications in the lower shell longitudinal seam. The apparent depth of these flaw indications, from the inside diameter surface, extended beyond the clad-to-base metal interface. To provide supplemental information the licensee use a computer-based system known as UDRPS (Ultrasonic Data Recording and Processing System). The data from the UDRPS indicates that the two flaws are volumetric in nature and are located entirely within the clad.

Indications 3A, 4A, and 22A in nozzle-to-shell weld No. 21 were detected at or near the weld/nozzle forging fusion line using the Westinghouse 40-month array plate. The transducer is a 2.25 MHz, 1

1/2 inch diameter, 0° longitudinal wave unit. This configuration was also used with the UDRPS for characterization and sizing of the flaw indications. The three reflectors are oriented around the circumference of the weld in essentially the same plane.

The scanning and dimensioning examinations with both ultrasonic testing systems provided consistent and similar data. The straight beam techniques produced relatively strong reflections. Satellite pulses were observed with the supplemental characterization technique suggesting that the flaw indications originate from volumetric type defects, such as slag or porosity. Even with small flaws, the estimated size is more consistent with the beam size of the transducer rather than the size of the flaw (for beam sizes greater than the size of the flaw).

The staff has reviewed the examination results and concludes that the licensee's dimensions of the flaw indications are conservative. The staff does not believe that weld No. 21 contains slag indications which have a depth of 1 1/2". The length and depth dimensions obtained with the conventional and supplemental techniques are similar for all three indications. The examination data suggests that the reported dimensions are a function of the characteristics of the transducer rather than a measurement of the size of the flaw indication.

C. Fracture Mechanics Evaluation

The most conservative dimensions obtained during the ultrasonic examinations were used in the fracture mechanics evaluation of the nozzle-to-shell weld. The licensee has provided a flaw evaluation chart for the B loop reactor vessel outlet nozzle-to-shell weld. The method and criteria used in the fracture mechanics analyses are documented in Reference 1. The portions of this document that were related to the flaw evaluation were documented in Enclosure 3 of licensee's April 28, 1988 submittal. The fracture mechanics analyses that were performed to develop the flaw evaluation chart were in accordance with the methodology and criteria specified in Article

IWB-3600 and Appendix A of the ASME Code, Section XI, except that stresses were not linearized and stress intensity factors were not calculated in accordance with the recommendations in Appendix A. In lieu of linearizing the stress, the method used represented the actual stress profile by a third order polynomial. Stress intensity factors were calculated using the expressions of Reference 2. These stress intensity factor expressions have been shown to be applicable to vessels in Reference 3. These stress profiles and stress intensity factor expressions provide a more accurate determination of the critical flaw size, and are particularly important during the evaluation of emergency and faulted conditions where the stress profile is generally nonlinear and often very steep.

Important parameters in a fracture mechanics analyses are the materials' brittle fracture resistance and the projected flaw growth rate during operation of the component. The standard measurement of the brittle fracture resistance for the reactor vessel material are their crack initiation and arrest fracture toughness. These values of fracture toughness are used to determine a critical flaw size. Westinghouse indicates that the critical flaw size calculation used the crack initiation and arrest fracture toughness for vessel materials that are recommended in Appendix A of the ASME Code, Section XI. The critical flaw size for the outlet nozzle-to-shell weld location was determined using a reference temperature, RT_{NDT}, of 60°F and an upper shelf toughness of $200 ksi(in)^{1/2}$. These values are acceptable for this location in the reactor vessel because the materials in this location are not subject to significant amounts of neutron irradiation and the RT_{NDT} values were estimated using the method recommended by the staff in NUREG-0800, "USNRC Standard Review Plan," Branch Technical Position MTEB 5-2.

The amount of projected flaw growth was determined to be negligible. The calculation was performed for the reactor vessel design transients that are listed in Table 2-1 in Enclosure 3 of the April 28, 1988 submittal. The rate of fatigue growth was calculated using the ASME reference curve for air environment. Since the flaws under evaluation are embedded, this method of calculating the flaw growth rate is acceptable. The flaw evaluation chart was constructed from fracture mechanics analyses of all reactor vessel design transients, that are listed in Table 2-1 and pressurized thermal shock (PTS) transients that are listed in Table 4-1. Fracture mechanics analyses were performed for only limiting PTS transients. PTS transients were considered limiting because their conditional probability of failure was greater than 1×10^{-2}, and all other PTS transients had conditional probability of failure much less than 1×10^{-2}. The PTS probabilistic failure assessment was based on probabilistic fracture mechanics and considered 8,000 events, which were categorized in Reference 4. The limiting transient for the outlet nozzle location was determined to be the turbine roll transient. The flaw evaluation chart indicates that the reported flaw sizes meet the fracture mechanics criteria in Article IWB-3600 for the 40 years of service life of the plant.

In addition to the reactor vessel design transients, which are listed in Tables 2-1 and 4-1, the licensee evaluated a postulated low temperature overpressure (LTOP) event, which was not mitigated by the LTOP protection system. The licensee's detailed probabilistic risk assessment indicated that this event should be classified as a faulted condition. The licensee's analysis of the postulated LTOP event indicates that the LTOP event is not a governing transient because it is much less severe than the other faulted conditions.

The NRC required licensees to install LTOP protection systems in 1979. Since the industry installed LTOP protection systems, there has been only one event in which the LTOP system did not mitigate the event. This event occurred on November 28, 1981 at Turkey Point Unit

4 (Reference 5). In this event, the pressure rose to 1100 psi, at a temperature of 110°F. Pressurized Water Reactors (PWRs) have accumulated approximately 400 years of plant operation since installation of LTOP protection systems. Since only one event has occurred in 400 years of accumulated PWR plant operation, the event would not be expected to occur during the 40 year life of a PWR nuclear power plant. Hence, according to 10 CFR Part 50, Appendix A, the event is not an anticipated operational occurrence and may be considered an emergency/faulted condition.

To conservatively bound LTOP events for the Farley Unit 1 reactor vessel, the staff has performed a fracture mechanics analysis for the reactor vessel in which the postulated event occurred at 110°F and pressurized the vessel at 1500 psi. The analysis was performed using the methodology described in Appendix A of ASME Code, Section XI. The staff's evaluation indicates that for the postulated event, the flaws in the nozzle-to-shell weld will meet the acceptance criteria in Article IWB-3600 for emergency/faulted conditions.

3.0 CONCLUSIONS

A. Based on the UDRPS data, the flaws found during examination of Weld No. 6 are entirely within the clad and will not affect integrity of the Farley Unit 1 reactor vessel. In the event that future examination of the reactor vessel welds detect flaw indications that extend past the clad-to-base metal interface and a fracture mechanics evaluation is required by the ASME Code, the stress resulting from differential contraction of the steel and clad must be included in the evaluation of PTS transients.

B. Based on the licensee's and the staff's independent evaluation of a postulated LTOP event, the flaws in the Loop B, outlet nozzle-to-shell weld satisfy the analytical evaluation criteria in Article IWB-3600. Based on these analyses, the flaws in the weld will not grow to a size that will affect the integrity of the reactor vessel during the life of the plant . Therefore, the reactor vessel is acceptable for the 40 years of service life of the plant.

C. However, the flaws in the outlet nozzle-to-shell weld are conditionally acceptable. Pursuant to ASME Code, Section XI, paragraphs IWB-3122.4(b) and IWB-2420(b), weld No. 21 will be reexamined during the next three inspection periods. The staff concludes that the licensee should evaluate the use of an additional transducer with a narrower beam spread for the reexaminations. A comparison of the results with the transducer used during the 1988 examination and another with optimum characteristics at the location of the flaw should provide a better definition of the dimensions of the reflector.

D. Paragraph 10 CFR 50.55a(g)(4)(i) establishes the ASME Code requirements for the initial 10-year inspection interval. The requirements for successive 10-year inspection intervals are defined by paragraph 10 CFR 50.55a(g)(4)(ii). Based on these provisions of the regulation, the staff has determined that inservice inspections performed during the four inspection intervals can not coincide. Therefore, all inservice inspections required by the regulation and the Technical Specifications for the initial inspection interval must be completed before any valid second interval inspections start. The ASME Code requirements for the testing of pumps and valves are addressed separately in the regulation and the inspection intervals may be different than for ISI.

ASME Code (83E83S), paragraph IWB-2420(a), states that the sequence of component examinations established during the first inspection interval shall be repeated during each successive inspection interval, to the extent practical. Table IWB-2412-1 defines the minimum and maximum credit for examinations for each inspection period during a 40-years plant service. Based on these provisions, that staff has made the interpretation that the ASME Council did not intend that a plant owner perform examinations of reactor vessel pressure boundary welds for two inspection intervals during the same refueling outage.

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References

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5. W.D. Lanning, "Low Temperature Overpressure Event at Turkey Point Unit 4," Case Study Report by Office for Analysis and Evaluation of Operational Data, NRC, March 1984.