400 Chestnut Street Tower II

May 28, 1981

SQRD-50-328/81-15

Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

SEQUOYAH NUCLEAR PLANT UNIT 2 - REACTOR VESSEL FIELD WELD INDICATIONS - SQRD-50-328/81-15 - FINAL REPORT

The subject deficiency was initially reported to NRC-OIE Inspector M. Thomas on January 20, 1981 in accordance with 10 CFR 50.55(e) as NCR's 2511 and 2547. Interim reports were submitted on February 19, April 6, and May 19, 1981. Enclosed is our final report.

If you have any questions, please get in touch with D. L. Lambert at FTS 857-2581.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure
cc: Mr. Victor Stello, Director (Enclosure) ✓
Office of Inspection and Enforcement
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ENCLOSURE
SEQUOYAH NUCLEAR PLANT UNIT 2
REACTOR VESSEL FIELD WELD INDICATIONS
SQRD-50-328/81-15
10 CFR 50.55(e)
FINAL REPORT

Description of Deficiency

Baseline liquid penetrant inspection (PT) of the Sequoyah unit 2 reactor vessel revealed several unacceptable linear indications in two of the eight reactor vessel nozzle safe end welds. In-situ metallography was performed to characterize the nature of the indications. These indications were reduced to a size acceptable to ASME Section XI. Because of the nature of these indications, a commitment was made to reexamine the areas following hot functional testing.

Following hot functional testing the reactor vessel nozzle safe end weld indications were reexamined by liquid penetrant testing (PT). Also, additional metallography was performed. The PT inspection revealed that several of the indications which had previously been reduced to an acceptable size before hot functional testing to meet the acceptance requirements of ASME Section XI (Summer 1978 Addenda), had increased in size but still met ASME Section XI criteria. During this examination, several additional indications were discovered that were not revealed when the liquid penetrant examination was performed before hot functional testing. Metallography performed at this time indicated no essential change in microstructure. At this time the decision was made to liquid penetrant examine the remaining six nozzle safe end welds which had been examined before hot functional testing and were free of reportable indications. All six nozzle safe end welds had some indications that were not present before hot functional testing. The indications discovered on the six nozzle safe end welds were similar to those discovered on 2RC-9 and 2RC-17, however, none had cracking as extensive as that found on 2RC-17.

Approximately 75 percent of the linear indications were parallel to the direction of coolant flow and were located in a section circumferentially around the outside surface of the nozzles. The remaining 25 percent of the linear indications were randomly oriented and lay in the same circumferential section. All were located in the 309 stainless steel weld metal "buttering" which was deposited on the end of the low alloy steel nozzles to facilitate field welding.

TVA elected to inspect all eight unit 1 reactor vessel nozzle safe end welds during the outage concurrent with this inspection. Of the eight nozzle safe end welds inspected, six welds have indications which meet ASME Section III and Section XI acceptance criteria. Two nozzle safe end welds have indications which meet ASME Section XI preservice and inservice examination acceptance criteria, i.e., one indication 1/8 inch long in each nozzle. The same pattern of extensive axial indications as was seen on nozzle 2RC-17 was not found on the unit 1 nozzle safe end welds.

Metallurgical evaluation of the indications at various times during inspection and field work on 2RC-17 and 2RC-9 indicates that the cause of cracking is intergranular stress-assisted corrosion cracking. Tight cracks, of a size undetectable by liquid penetrant testing opened up under the imposed stresses of hot functional testing to a size detectable by PT as the buttering adjusted to the stress field.

The extensive cracking observed on nozzle 2RC-17 can be attributed to an incident which occurred during a clad repair operation performed at the site on this nozzle. During the low temperature heat treatment (400 degrees F to 500 degrees F) on 2RC-17, 2RC-1, and 2RC-24 during the clad repair, Griffolyn type 75 and duct tape, which was attached to the outside of the nozzles, was burned and melted onto the weld buttering resulting in extensive halogen contamination of the buttering. Records exist to show that all nozzles except 2RC-17 were penetrant examined during the construction phase to verify integrity. The nozzle safe end at 2RC-17 was not PT examined at this time due to the field weld being in progress. The entire area was to be PT examined upon completion of the field weld. No documentation can be located to show that the final ANSI B31.7 liquid penetrant examination was performed on the outside diameter of nozzle 2RC-17.

If a liquid penetrant examination had been performed upon completion of field weld 2RC-17, the probability of indications being detected at that time would have been very good. Due to the fact that a liquid penetrant examination was not performed, it cannot be conclusively determined that the nozzle was cleaned; consequently, the contaminants on the nozzle would have remained for a long period of time. It may be surmised that the extensive cracking seen on 2RC-17 in relation to cracking on the other nozzles is probably attributable to a high concentration of contaminant material and a failure to PT and clean up this nozzle adequately. TVA concludes that the 2RC-17 nozzle is unique in the entire population of nozzles at Sequoyah.

Safety Implications

Nozzle 2RC-17 had numerous indications of a size which would not have compromised the safe operation of the plant. WCAP-9382, "Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March 1978, presents a generic analysis giving the critical flaw size in primary piping under maximum design loads. The most limiting region is the reactor vessel outlet nozzle safe end. The flaw considered was a through-wall flaw oriented in the circumferential direction so as to be normal to the maximum stress. The applied moment in this region was 59.28×10^6 in-lb. The calculations yield a critical through-wall flaw (2.6 inches deep) about 25 inches long. The indications of interest here are significantly smaller (a maximum of 7/32-inch deep by 1/4-inch long) and oriented longitudinally, a more favorable orientation. The maximum applied moment is 41×10^6 in-lb. Crack growth by fatigue is considered negligible. Accordingly, it is concluded that the indications would have had an insignificant effect on structural integrity.

Corrective Action

All linear indications on all eight unit 2 nozzles were removed by grinding. The maximum depth of grinding for removal of indications on 2RC-17 was 7/32 inch. Six of the nozzles had less than 1/16-inch metal removal and 2RC-16 had less than 3/32-inch metal removal. The longest indication on 2RC-17 was 7/16 inch.

The unit 2 outlet nozzle safe end weld (2RC-17) required repair. After removal of the indications, an area from 3 o'clock to 6 o'clock (approximately 25 inches) had numerous areas which violated the design minimum wall thickness calculated by Westinghouse. The depth varied from a maximum of 7/32 inch to a minimum of less than 1/32 inch. The average width of the cavity was approximately 3/8 inch. Typically, 2/3 of the excavation was in the weld buttering side and 1/3 of the excavation was on the nozzle side of the nozzle-butter interface.

The entire area between 3 o'clock and 6 o'clock was repaired to eliminate numerous starts and stops which would make the repair more difficult. A half bead weld repair was performed in accordance with ASME, Section III, paragraph NB-4642, 1974 Edition.

Examination of the completed repair will consist of a liquid penetrant and ultrasonic examination of the repair area at least 48 hours after the weld has cooled to ambient temperature. The liquid penetrant examination will be performed to the requirements of ASME, Section V, and the acceptance standards of ASME, Section III, paragraph NB-5350. A preliminary liquid penetrant examination of the repair was acceptable. The ultrasonic examination will consist of a 70 degrees angle beam examination to the maximum extent possible and supplemented by 0 degrees straight beam examination from the outside diameter to the requirements of ASME, Section V, using the acceptance criteria of ASME, Section III, paragraph NB-5330. A preliminary 70 degrees angle beam and 0 degrees straight beam ultrasonic examination of the area showed acceptable results. These inspections will be completed by May 29, 1981.

In addition to the above examinations, a manual baseline ultrasonic examination from the inside diameter will be performed of the repair area in accordance with ASME, Section XI, before fuel load and the nozzles restored to a clean condition. Also, a hydrostatic test will be performed after fuel loading in accordance with ASME Section XI.

After the remaining indications on 2RC-17 and the indications on the other seven nozzles were removed by grinding, the nozzles had sufficient remaining wall thickness. These cavities were blended to at least a 3-to-1 taper.

All nozzles in units 1 and 2 are in an acceptable condition for safe operation. TVA will reinspect nozzle 2RC-17 and one additional unit 2 nozzle at the first regularly scheduled cold shutdown outage of sufficient duration to perform the inspection. The indications on unit 1 will be monitored at the normal inservice inspection intervals for these dissimilar metal welds.