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JPN-92-036

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, D.C. 20555

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Reactor Pressure Vessel Head Welds
Flaw Indication Inspections and Evaluation Analysis

References:
1. NYPA letter, J. C. Brons to NRC (JPN-90-040) dated May 25, 1990, "Reactor Pressure Vessel Head Flaw Indication Inspections and Evaluation Analysis."
2. NRC letter, D. E. LaBarge to J. C. Brons (TAC 76861) dated June 13, 1990, "Evaluation of Reactor Vessel Head Flaw Indication Inspection and Evaluation Submittal - J. A. FitzPatrick Nuclear Power Plant."
3. NYPA letter, H. P. Salmon, Jr. to J. P. Durr (JAFP-92-0360) dated April 30, 1992, "NRC Inspection Report 50-333/92-05."

Dear Sir:

Indications of possible flaws in a reactor pressure vessel head weld were found during routine in-service refueling outage inspections in 1990. Evaluations of the indications were performed in accordance with the ASME code. Reference 1 transmitted these evaluations to the NRC. These evaluations confirmed the existence of subsurface flaws due to original welding imperfections. They concluded that reactor operation with the existing weld flaws did not constitute a safety concern.

In order for the reactor vessel head to be accepted for continued service, the ASME code requires a reexamination of the weld during the next three refueling outages. The NRC requested in Reference 2 that the results of the reexaminations be incorporated into an analytic evaluation to justify operation. Reference 2 stated that the evaluations should be submitted to the NRC for staff review prior to resumption of reactor operation from each of the three subsequent operating cycles.

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The attachment to this letter provides the results of the reactor vessel head weld inspections conducted during the 1992 refueling outage. The inspection data was submitted to the NRC in Reference 3. Inconsistencies between 1990 and 1992 examination data have been resolved as described in the attachment.

Based on the results of these inspections, the reactor vessel head weld flaws do not constitute a safety concern.

If you have any questions, please contact Mr. J. A. Gray, Jr.

Very truly yours,

Ralph E. Beedle
Executive Vice President
Nuclear Generation

cc: Regional Administrator
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Attachment to JPN-92-036

James A. FitzPatrick Nuclear Power Plant

Reactor Vessel Head Weld Flaw Indications
1992 In-service Inspection Results

Introduction

As part of the routine in-service inspection (ISI) program, selected reactor pressure vessel (RPV) head welds were inspected during the 1990 refueling outage. Inspectors used ultrasonic testing (UT) techniques to detect and size flaws in reactor pressure vessel head welds.

Ultrasonic Inspections - 1990 Refuel Outage Results

UT inspections of RPV head weld number VC-TH-1-2 showed several recordable indications. The largest indication was observed along approximately five inches of the circumferential weld between the upper dome plate (dollar plate) and the vertical dome segments. These indications were the subject of NRC Information Notice 90-32 and General Electric Company Rapid Information Communication Services Information Letter (RICSIL) 051. Both documents are dated May 3, 1990.

As a result of these findings, additional examinations were performed in accordance with the requirements of ASME Section XI, paragraph IWB-2430 as stated in Reference 1.

Other inspections, beyond those required by ASME Section XI, were conducted on weld VC-TH-1-2 to clarify the nature and extent of the flaws. These supplemental inspections included visual (VT), radiographic (RT), dye penetrant (PT), and magnetic particle (MT) examinations on the reactor side (underside) of the vessel head. Additional UT exams were performed from both the outside and inside of the head. Construction radiographs and those taken during the 1990 refueling outage were computer enhanced to better quantify the weld characteristic.

Some of the UT exams used in sizing these flaws were hampered by the existence of numerous small reflectors located about mid-wall in the plate. These reflectors are believed to be metallic inclusions (also known as plate segregates), probably manganese sulfides. These inclusions are part of the steel making process and are considered acceptable by the manufacturing specification for ASME SA-533 Grade B steel. They were also observed during pre-service UT inspections.

When performing sizing exams with refracted longitudinal wave transducers, shear and longitudinal sound waves are generated. The UT inspectors initially confused the segregate response from shear waves with a flaw response from longitudinal waves. The shear waves reflected off the segregates generated a response near the center of the plate on the time display. This resembled a response from the longitudinal waves which was interpreted as a flaw. As a result, inspectors overestimated the flaw depth to be 2 inches. The length of the flaw was similarly overestimated.

Flaw Evaluation

Two flaws were rejectable under the guidelines of NRC Regulatory Guide 1.150. These flaws were conservatively estimated to be 0.5 inch deep by 5 inches long, and 0.55 inch deep by 2.3 inches long. For the purposes of the fracture mechanics evaluation, these flaws were assumed to be open to the vessel interior (i.e. cracks), although inspection data indicated the contrary. The assumption that a crack exists is conservative since it presupposes flaw growth.

This information, and the original structural and detailed fracture mechanics evaluations, were provided to the NRC as Attachments I and II to Reference 2.

This weld was re-inspected during the 1992 refueling outage as required by the NRC and ASME Section XI.

Ultrasonic Inspections - 1992 Refuel Outage Results

The ISI inspections performed during the 1992 refueling outage included weld VC-TH-1-2. The inspections were performed by Ebasco Services Inc., the ISI contractor, with additional inspections and final data review conducted by two Authority quality assurance (QA) level III inspectors. Although not required, all inspections were conducted by personnel certified by the BWROG-EPRI IGSCC program.

The inspection techniques and equipment used during the 1992 reexaminations were comparable to those employed during the 1990 inspections. When the initial 1990 examinations were performed, no permanent references existed to ensure repeatability. To make sure the reexaminations captured indications identified in 1990, inspections performed in 1992 included an area larger than the locations reported in the early examinations. This also enabled Authority personnel to develop permanent reference marks for repeatability when performing future examinations. The examination performed in 1992 on RPV head weld VC-TH-1-2 confirmed the two indications previously reported as unacceptable in 1990.

Inconsistencies Between 1990 and 1992 Data

The evaluation identified some differences between the recorded data of 1990 and 1992. The 1992 recorded data included shorter length measurements and smaller through wall dimensions. These differences prompted supplemental examinations by Authority QA level III personnel and a complete reevaluation of all 1990 and 1992 inspection data to determine final disposition of these indications. The examinations by the Authority, reevaluation of all data, and subsequent discussions with GE and Ebasco personnel, resolved the differences noted in inspection data.

The differences between 1990 and 1992 data are attributed to differences in the evaluation techniques used during the inspections. The longer lengths and greater through wall dimensions reported in 1990 are from using composite data (consolidating inspection results of various examination angles and combining automated with manual inspections).

Determining the dimensions of indications using the 1990 evaluation technique is an extremely conservative methodology exceeding the sizing criteria outlined in ASME Section XI and NRC Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Pre-service and Inservice Examinations," Rev. 1. Applying this type of conservative sizing in the structural evaluation assures a large safety margin between the evaluated flaw size and the actual flaw size. Duplication of this type of evaluation when applying the sizing criteria outlined in the governing codes and documents is not possible due to the amount of conservatism built into the bounding rectangle. The inspection and evaluation, performed during the 1992 examinations, used criteria outlined in ASME section XI and NRC Regulatory Guide 1.150, Rev. 1. When data from the 1990 and 1992 inspections are compared after analysis, no measurable change in length or through wall dimension is discernable.

Conclusion

Based upon the inspections, and manufacturing records for the RPV head, the Authority determined that the flaws are due to original manufacturing imperfections. The flaws are not cracks because they do not reach the surface.

Based upon the results of evaluations performed in accordance with the Technical Specifications, the ISI program, and ASME section XI, continued operation with the existing reactor vessel head indications do not constitute a safety concern.

The Authority will re-inspect the weld during the next refueling outage in accordance with ASME Section XI (1980 edition through winter 1981 addenda), IWB-2420, and will report any changes to the NRC in the inspection results.