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Brian R. Sullivan
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JAFP-17-0024
March 22, 2017

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: LER: 2017-002, Residual Heat Removal to Reactor Water Recirculation Loop A Weld Flaw Indication

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

Dear Sir or Madam:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(A), as a condition of the nuclear plant, including its principle safety barriers, being seriously degraded.

There are no new regulatory commitments contained in this report.

Questions concerning this report may be addressed to Mr. William Drews, Regulatory Assurance Manager, at (315) 349-6562.

Sincerely,

Brian R. Sullivan
Site Vice President

BRS/WD/dc

Enclosure: LER: 2017-002, Residual Heat Removal to Reactor Water Recirculation Loop A Weld Flaw Indication

cc: USNRC, Region I Administrator
USNRC, Project Manager
USNRC, Resident Inspector
INPO Records Center (ICES)

NRC FORM 366 (06-2016) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Residual Heat Removal to Reactor Water Recirculation Loop A Weld Flaw Indication

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	N/A
1	22	2017	2017	002	00	3	22	2017	N/A	N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
5	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☑ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
0	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Mr. William Drews, Regulatory Assurance Manager	315-349-6562

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	AD			N					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☑ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

During James A FitzPatrick Nuclear Power Plant (JAF) Refueling Outage R22, while in Mode 5, on January 22, 2017, phased array Ultrasonic Test (UT) examination identified a defect in weld 24-10-130. This dissimilar metal weld is located between the ‘A’ Residual Heat Removal (RHR) Low Pressure Coolant Injection (LPCI) Loop valve 10RHR-81A and the reducing Tee to the Reactor Water Recirculation (RWR) System Loop A. The flaw is consistent with Inter-Granular Stress Corrosion Cracking (IGSCC). The defect did not meet IWB-3514-2 allowable standards and is reportable pursuant to 10 CFR 50.73(a)(2)(ii)(A) as a condition of the nuclear plant, including its principle safety barriers, being seriously degraded.

JAF performed a full structural weld overlay based upon ASME Code Cases N-504-4 and N-638-4. The alternative repair method was submitted to the NRC as ISI Relief Request RR-21.

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NARRATIVE

Background

The Reactor Water Recirculation (RWR) System [EIIS identifier: AD] is designed to provide forced coolant flow through the core to remove heat from the fuel. The RWR System consists of two recirculation pump loops external to the reactor vessel, inside the Drywell. These recirculation loops are part of the Reactor Coolant Pressure Boundary (RCPB) and provide the piping path for the driving flow of reactor water to the reactor vessel jet pumps.

The Residual Heat Removal (RHR) system [BO] provides the means to remove decay and residual heat from the Reactor Coolant System (RCS) for refueling operations; supplement the spent Fuel Pool Cooling system, cool the suppression chamber water, and provide low pressure coolant injection to the Reactor Pressure Vessel (RPV). The Emergency Core Cooling System (ECCS) safety function of the RHR system is to restore and maintain the coolant inventory in the RPV so that the core is adequately cooled after a Loss of Coolant Accident (LOCA). The LPCI is designed for automatic operation following a break in one of the reactor recirculation loops.

The A Loop of RHR system connects to the RWR Loop A at the dissimilar metal weld between the 24” nominal pipe size locked open carbon steel (CS) manual isolation valve 10RHR-81A and the 24” side of the stainless steel (SS) 28” x 28” x 24” Tee of the RCS A Loop. The dissimilar metal weld joins the carbon steel valve and the stainless steel Tee using Inconel 82/182 as the weld materials. See Figure 1.

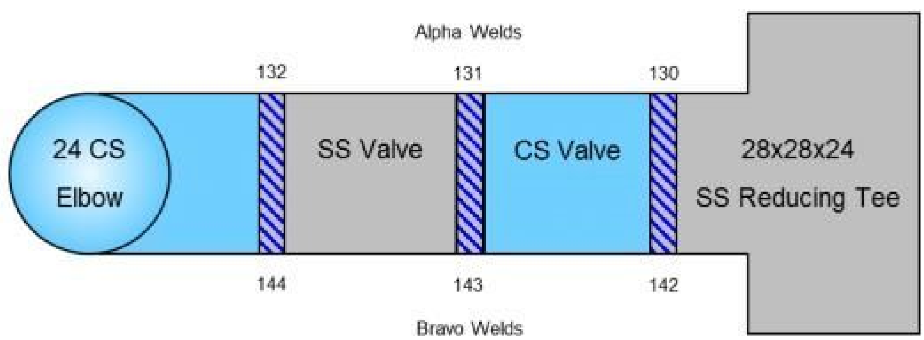

Figure 1

Inter-granular stress corrosion cracking (IGSCC) in Boiling Water Reactor (BWR) piping is an industry wide concern. Engineering studies and industry studies have shown that Inconel 82/182 weld filler material is susceptible to IGSCC. JAF’s program for addressing IGSCC is based on the EPRI Technical Report BWRVIP-75A which defines inspection schedules for stainless steel piping welds in BWR. The report defines Category ‘D’ welds as those made of susceptible materials which have been found free of defects. JAF mitigates the potential for IGSCC using hydrogen (in a Hydrogen Water Chemistry- HWC- program) and noble metal chemical addition (in an on-line Noble Metal Chemical Addition – NMCA- program).

JAF scheduled and completed the examination of all six Category ‘D’ welds in the RHR system during Refueling Outage R22. The examination of weld 24-10-130 is the only one that showed an indication of a flaw. Weld 24-10-130 was previously examined in 2010 with no indications identified.

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Event Description

A planned inspection of the A RHR LPCI to A RWR loop dissimilar metal weld 24-10-130 took place during R22 in January 2017. The inspection consisted of an ultrasonic testing (UT) examination of the dissimilar metal weld between the 24” nominal pipe size carbon steel valve 10RHR-81A and the 28” x 28” x 24” stainless steel Tee of the A RWR Loop. The examination showed an inner diameter axial flaw indication (oriented parallel to the axis of the piping run) with a length approximately 0.95” long with a maximum wall depth of 0.81” with a remaining ligament (sound metal above the flaw tip) of 0.34”. See Figure 2.

Assumed Flaw Shape

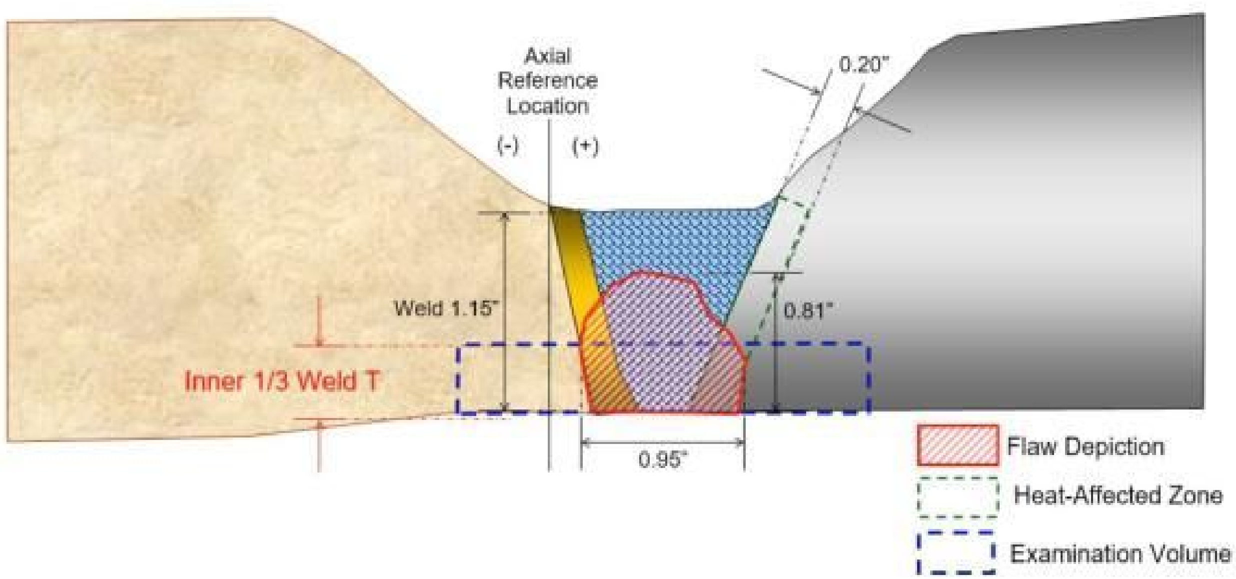

Figure 2

This flaw did not meet IWB-3514-2 allowable standards and was reported to the NRC per 10 CFR 50.72(b)(3)(ii)(A) as ENS 52503 and this License Event Report (LER) is submitted per 10 CFR 50.73(a)(2)(ii)(A) as a condition of the nuclear plant, including its principle safety barriers, being seriously degraded.

Cause

The flaw indication, in the dissimilar metal weld between the A RHR LPCI connection to the A RWR Loop, 24-10-130, is consistent with Inter-granular Stress Corrosion Cracking (IGSCC).

Review of weld record history for weld 24-10-130 identified that in 1973 the original weldment underwent a repair from the inner surface of the weld root to address a rejectable linear indication. The indication was successfully repaired based upon later weld acceptance. The 1973 repaired area and this 2017 flaw indication are located at the same azimuthal location. A repair from the inside diameter can generate high tensile residual stresses associated with the repair process. A tensile stress condition provides a driving force for IGSCC.

In recent years prior to R21 in 2014, when JAF replaced Condenser tubing in both Main Condensers, reactor water chemistry was challenged by frequent Condenser tube leaks due to wall thinning. Water chemistry could be part of the cause of crack growth after the previous weld inspection in 2010.

Similar Events

A similar condition was identified in R18 in 2008 when an axial flaw was identified during an ISI inspection of dissimilar metal weld in RPV Reactor Recirculation inlet nozzle N2-C. The flaw indication, in the dissimilar metal weld area of N2-C nozzle to safe end weld, was consistent with IGSCC.

Corrective Actions

Completed Actions

The weld flaw at 24-10-130 was corrected by implementation of a full structural weld overlay using ASME Code Cases N-504-4 and N-638-4. This repair method was submitted as ISI Relief Request RR-21 and authorized by the NRC. The weld overlay repair involved the deposition of Alloy 52M weld metal on the outside surface of the dissimilar metal weld and adjacent base material. The repair method reduces the potential for IGSCC by the use of weld filler materials that are resistant to this mechanism.

- Performed UT of final weld overlay at 24-10-130. Detected indications associated with the final weld overlay were evaluated using the acceptance standards of ASME Section XI IWB-3514-2 and determined to be acceptable.
- Performed UT examinations of six RWR to RPV nozzle to safe end N2 dissimilar metal welds. Results of these examinations were acceptable.

Future Actions

- Perform UT examination of the weld overlay at 24-10-130 in R23 or R24.

Safety Significance

There was no actual radiological or nuclear safety consequence during this event.

The safety significance of this event is considered minimal. The axially – oriented weld flaw in the valve to tee dissimilar metal weld was approximately 0.95” long with a maximum wall depth of 0.81” with a remaining ligament (sound metal above the flaw tip) of 0.34”.

If the axial weld flaw had extended 100% through wall during power operation and resulted in RCS leakage, Control Room operators monitoring unidentified and total Drywell leakage would have identified the condition. If the unidentified leakage rate increased to more than 2 gpm within a 24 hour period or greater than 5 gpm total, a plant shutdown would have been performed in accordance with plant Technical Specification 3.4.4.

JAF is designed to mitigate the consequences of major pipe breaks. The plant’s design criteria ensures that the public is protected in accordance with 10 CFR 100 guidelines for pipe break events. Therefore, the identified flaw in the RHR Loop A to A RWR Loop dissimilar metal weld was fully enveloped by JAF design analyses.

References

- Condition Report: CR-JAF-2017-00706, Axial Weld Flaw in A RHR LPCI Injection Loop to A RWR Loop