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JAFP-17-0016
March 13, 2017

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

Subject: LER: 2017-001, Vent Line Socket Weld Failure

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), 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,

(Acting for Sullivan)

Brian R. Sullivan
Site Vice President

BRS/WD/ds

Enclosure: LER: 2017-001, Vent Line Socket Weld Failure

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
James A. FitzPatrick Nuclear Power Plant	05000333	1 OF 5

4. TITLE
Vent Line Socket Weld Failure

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: N/A	DOCKET NUMBER: N/A
1	14	2017	2017	- 001	- 00	3	13	2017	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE: 2

10. POWER LEVEL: 0

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

[ ] 20.2201(b)	[ ] 20.2203(a)(3)(i)	[x] 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)
[ ] 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)
[ ] 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)* [x] NO				

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

Refueling Outage 22 commenced on January 14, 2017 at James A. FitzPatrick Nuclear Power Plant (JAF). With the plant in Mode 2 at 0613, the initial Drywell inspection identified a through wall leak on the ¾ inch vent line off of the bonnet of the motor operated gate valve on the suction side of Reactor Water Recirculation Pump ‘A’. This condition was determined to constitute Reactor Coolant Pressure Boundary (RCPB) leakage, which is prohibited by Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.4. The average reactor coolant temperature decreased to less than 212 degrees F and the plant was in Mode 4 at 1530 of the same day, which is within the applicable TS LCO 3.4.4 required completion time.

The condition of a through wall leak on the RCPB is reportable pursuant to 10 CFR 50.73(a)(2)(ii), as a condition of the nuclear plant, including its principle safety barriers, being seriously degraded.

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 forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Water Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, driven by a motor generator (MG) set to control pump speed, and associated piping, jet pumps, valves, and instrumentation.

Motor operated gate valves on the suction and discharge sides of each recirculation pump serve as isolation valves for the pumps. A ¾” stainless steel line coming from the bonnet section of each valve provides a vent line for the valves. The recirculation loops are part of the Reactor Coolant Pressure Boundary (RCPB) and are located inside the drywell structure.

The Reactor Coolant System (RCS) includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the RCPB. The joints of the RCPB components are primarily welded or bolted.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. Limits on RCS operational leakage are specified in Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.4, and are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. TS LCO 3.4.4 is applicable in Modes 1, 2, and 3, and allows for no RCPB leakage. JAF is required to be in Mode 4 within thirty-six (36) hours if RCPB leakage exists.

Event Description

Drywell inspections are conducted at the discretion of the Operations Manager during shutdown to identify sources of Drywell leakage. During the initial Refueling Outage 22 (R22) Drywell inspection, personnel observed a three to four foot steam plume emanating from the vent line coming from the bonnet section of the motor operated gate valve on the suction side of RWR Pump ‘A’ (02-2MOV-43A). The steam plume was observed on the vertical run of 02-2-3/4”-WH-1504-35A where the line leaves the insulation.

Investigation revealed that the leak originated from a weld on the downstream side of a 45 degree elbow which is the first fitting on the line coming from the bonnet of 02-2MOV-43A. Further analysis by the Welding Engineer determined that the crack originated at the toe of the socket weld on the 45 degree elbow, and then propagated out away from the point of origin and into the pipe wall.

The RCPB leakage was discovered during the initial drywell inspection with the plant in Mode 2 at 0613 on January 14, 2017. The average reactor coolant temperature decreased to less than 212 degrees F and the plant was in Mode 4 at 1530 of the same day, which is within the applicable TS LCO 3.4.4 required completion time.

Event Analysis

The allowable RCS operational leakage limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence from experiments suggests that, for leakage even greater than the specified unidentified leakage limits, the probability is small that the imperfection or crack associated with such leakage would grow rapidly.

The unidentified leakage flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs show that leakage rates much greater than 5 gpm precede crack instability.

Sources of identified and unidentified leakage in the drywell are classified by the drain sump to which leakage is directed. Should the time to pumpdown exceed preset limits an abnormal amount of leakage from one of the sump services is indicated. Increased amounts of identified or unidentified leakage will be noted by pump out alarms in the control room. In addition, the drywell sump monitoring system for the floor drain sump and equipment drain sump uses flow integrators to monitor the leakage. The two flow integrators, one for the equipment drain sump and the other for the floor drain sump, comprise the basic instrument system for quantifying leakage inside the drywell. The control room operator has the capability of trending the daily volumes pumped from the drywell sumps by means of flow integrators located in the Control Room. The unidentified leakage trends were reviewed for Cycle 22. The review confirmed that the TS LCO 3.4.4 RCS operational leakage limits were met for the entire Operating Cycle 22.

Cause

An apparent cause evaluation was conducted in accordance with the JAF Corrective Action Program. The Apparent Cause of the event was determined to be high cycle fatigue (HCF) due to vibration stress. The design of the vent piping and resonance frequency of the RWR pumps at low flow contributed to the vibration stress.

The subject vent piping was redesigned in 2002 to address cracking in the fittings caused by thermal movement (reference Similar Events section). A pipe support was modified by changing a U-bolt to a lateral restraint. This solution resolved the thermal movement issue, but created a new cantilever branch line.

Cantilever branch lines are typically two inch diameter or less, and fixed to a larger pipe on one end with the opposite end unsupported. The connection to the larger pipe is usually to a ‘sockolet’ or half coupling fitting. All loads applied to a cantilever branch line are resisted by the connection of the branch to the larger pipe. The applied loads (e.g., deadweight, thermal, seismic) create bending moments in the pipe that increase to a maximum value at the connection to the larger pipe. If all welds are of equal quality, size, and profile, the socket weld at the connection to the sockolet is the most prone to failure since it resists the greatest load. For this reason, the welds at the sockolet are typically a more robust weld profile. The vent line did not have the enhanced 2:1 weld profile on the socket weld connections, which is desirable when the line may be subject to vibration. In addition, the smaller (standard) weld profiles have less resistance to vibration stress and HCF. Note that vibration is not a design basis load, and is not accounted for in the plant’s piping design. This piping was properly configured and supported such that it satisfied all applicable design loading conditions (ie. deadweight, pressure, thermal, seismic).

JAF operated the RWR pumps at lower flow for an extended period of time during the previous operating cycle. This condition was determined to be a contributing factor because the vane pass frequency of the RWR pump coincides with the vent line resonance frequency with the RWR pump operating at lower pump speeds. Based on this condition, the 02-2MOV-43A vent line likely experienced an increased number of vibration cycles with vibration stress in the range of the piping endurance limit thus leading to high cycle fatigue failure of the socket weld.

Similar Events

Internal

A condition report initiated on October 30, 2000 documented cracks in fittings on the 02-2MOV-43A vent piping. The cause was determined to be excessive thermal stress in the piping due to pipe support binding. A walkdown of common lines showed no bent pipes or support damage. Since the configuration of the common lines allowed for significantly more flexibility than the line with the cracked fittings, the absorbed thermal growth resulted in less stress in these lines. Corrective actions included replacement of a U-bolt with a lateral restraint pipe support.

External

Hope Creek Generating Station, Unit 1: LER 2013-003-1, Through-wall flaw discovered on Residual Heat Removal Shutdown Cooling Return Vent Line

Three Mile Island Nuclear Station, Unit 1: LER 2014-002-0, Through-wall Leak on High Pressure Injection “A” Train Root Valve MU-V-1034 Socket Weld

FAILED COMPONENT IDENTIFICATION:

Manufacturer:	N/A
Manufacturer Model Number:	N/A
NPRDS Manufacturer Code:	N/A
NPRDS Component Code:	N/A
FitzPatrick Component ID:	N/A

Note that the leak originated from a failed socket weld, which is not assigned component identifiers.

Corrective Actions

Completed Actions and Extent of Condition (EOC)

- Perform Code Case N-666 weld overlay repair on 02-2MOV-43A vent line leak location (45 degree elbow).
- Install new Pre-fabricated Vent Line section (downstream of the repaired 45 elbow) with 2:1 weld reinforcements on 02-2MOV-43A.
- Add new Tie-Back Support to Vent Line.
- Perform 2:1 Socket Weld Buildups on 02-2MOV-43B, -53A, -53B Vent Lines, and -53A Drain Line (EOC).
- Perform VT1 Line inspection of pipes/socket welds

Future Actions

- Investigate cost/benefit of a modification for eliminating the 02-2MOV-43A/B & -53A/B vent lines.

Safety Significance

Actual Consequences

There were no actual consequences of this event relative to nuclear, industrial, or radiological safety.

Potential Consequences

Accidents that could result in the release of radioactive material directly into the primary containment are the result of postulated Reactor Coolant System pipe breaks inside the drywell. All possibilities for pipe break sizes and locations have been investigated including the severance of small lines, the main steam lines upstream and downstream of the flow restrictors, and the recirculation loop lines. The most severe Reactor Coolant System effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop lines. The design basis accident analyses that have been performed for JAF demonstrate that all accident scenarios remain within the 10 CFR 100 dose limits.

The potential safety significance of this event is considered minor. The safety significance of RCS leakage from the RCPB varies widely depending on the source, rate, and duration. The event discussed herein resulted in TS LCO 3.4.4 not being met due to RCPB leakage; however, the unidentified leakage limits were not exceeded during the previous operating cycle. This ensures that all accident analyses radiation release assumptions remain bounding for this event. In addition, quantitative information is available in the control room to permit operators to take corrective action should the leak have worsened to the point of being detrimental to the safety of the facility or public.

References

- Condition Report: CR-JAF-2000-05520
- Condition Report: CR-JAF-2017-00245 Equipment Apparent Cause Evaluation
- JAF Technical Specification and Bases