Exelon Generation Company, LLC

James A. FitzPatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315-349-6024 Fax 315-349-6480

Timothy C. Peter
Plant Manager – JAF

JAFP-18-0033
March 30, 2018

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

Subject: LER: 2017-004-01, Safety Relief Valves Out of Tolerance

James A. FitzPatrick Nuclear Power Plant
NRC Docket No. 50-333
Renewed Facility Operating License No. DPR-59

Dear Sir or Madam:

This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(D) as an "Operation or Condition Prohibited by Technical Specifications" and "Event or Condition that Could Have Prevented Fulfillment of a Safety Function," respectively.

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,

Timothy C. Peter
Plant Manager

TCP/WD/ds

Enclosure: LER: 2017-004-01, Safety Relief Valves Out of Tolerance

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

NRC FORM 366 (04-2017)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block) (See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)		Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Safety Relief Valves Out of Tolerance

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
11	21	2017	2017	- 004	- 01	03	30	2018	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 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)
100	☐ 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	SB	RV	T020	Y					

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)*

The As-Found test results for the eleven Safety/Relief Valve (S/RV) pilot assemblies removed and replaced during the 2017 Refueling Outage at James A. FitzPatrick Nuclear Power Plant (JAF) identified ten (10) S/RV pilot assemblies that lifted outside of the allowable tolerance required by Technical Specification Surveillance Requirement 3.4.3.1. Nine (9) two-stage S/RV's were found out of tolerance high, and one three-stage was found out of tolerance low. The ten S/RV pilot assemblies are assumed to have been inoperable at some point in the operating cycle that preceded the 2017 Refueling Outage resulting in a condition reportable pursuant to 10 CFR 50.73(a)(2)(i)(B).

The S/RV design features an electric actuation capability that provides a diversified means of opening the S/RV's despite the out of tolerance condition. However, the electric lift function is considered a backup to the mechanical S/RV's and is not credited in the accident analysis. Therefore, the TS inoperability of the ten (10) S/RV's also resulted in a condition reportable pursuant to 10 CFR50.73(a)(2)(v)(D).

The cause of the two-stage failures has been identified as corrosion bonding; the cause of the three-stage failure is attributed to calibration and subsequent setpoint drift. The safety consequences associated with this event are considered low due to the electric actuation capability.

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

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1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
James A. FitzPatrick Nuclear Power Plant	05000 – 333	2017	– 004	– 01

NARRATIVE

Background

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the Reactor Coolant Pressure Boundary (RCPB) will not exceed the ASME Code limits.

The James A. FitzPatrick Nuclear Power Plant (JAF) used ten (10) two-stage and one (1) three-stage Target Rock Safety/Relief Valves (S/RV) [EIIS Identifier: SB] for emergency pressure relief during operating Cycle 22. These valves are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In safety mode (or spring mode of operation), the spring-loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the code requirement.

Each S/RV can be opened manually in the relief mode from the control room by its associated two-position switch. If one of these switches is placed in the open position the logic output will energize the associated S/RV solenoid control valve directing the pneumatic supply to open the valve. Seven of the installed S/RV solenoid control valves can also be energized by the relay logic associated with the Automatic Depressurization System (ADS).

During each refueling outage all eleven of the pilot assemblies are removed and replaced with vendor tested and certified components. The pilots that are removed are sent to a vendor facility for testing, refurbishment, and certification. The test results for pilot assemblies removed in 2017, during Refueling Outage 22, identified ten (10) S/RV pilot assemblies that were out of allowable tolerance. Nine (9) of the pilots (all two-stage) lifted at greater than the allowable setpoint range, and one (three-stage) lifted at less than the allowable setpoint range.

In order to address the concerns with corrosion bonding, JAF will commence replacement of two-stage with three-stage Target Rock S/RVs in the next Refueling Outage (RO). Industry experience has shown that the three stage S/RVs are less susceptible to corrosion bonding. The design of the three-stage S/RVs produces a greater mechanical force on opening, resulting in a greater likelihood of overcoming any potential effects of corrosion bonding that might occur.

Event Description

As-Found testing was performed on all eleven main S/RV pilot assemblies removed in 2017, during RO22. The testing was conducted by NWS Technologies. The TS setpoint for each S/RV is 1145 +/- 34.3. During the initial lift test, ten of the eleven pilot assemblies failed to open within the allowable range (1110.7 to 1179.3). Nine of the ten two-stage and the three-stage S/RV pilot failed high and low outside the allowable range, respectively. As-Found failed test results are tabulated below.

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Table 1 – As-Found Test Results

In-service Location	Pilot Serial Number	First Test (psig)	Acceptance Range (1110.7 – 1179.3 psig)
02RV-71A	1088	1184	Unsat – High
02RV-71B	1080	1254	Unsat – High
02RV-71C[(a)]	51	1103	Unsat – Low
02RV-71E	1235	1245	Unsat – High
02RV-71F	1195	1183	Unsat – High
02RV-71G	1194	1202	Unsat – High
02RV-71H	1111	1228	Unsat – High
02RV-71J	1192	1242	Unsat – High
02RV-71K	1193	1239	Unsat – High
02RV-71L	1056	1214	Unsat - High

[(a)]Three-stage

Cause

JAF has extensive internal Operating Experience with the S/RVs failing higher than the allowable setpoint. Causal evaluations identified corrosion bonding as the cause for the upward setpoint drift on the two-stage S/RVs. The As-Found test results shown above in conjunction with the successful second lift of all two-stage valves support this conclusion. Corrosion bonding is a crevice corrosion phenomenon that occurs between highly polished metals in a wetted solution in close proximity to each other. This close proximity (usually a gap of between 0.1 and 100 μm) creates a crevice-like condition between the two wetted surfaces setting up the conditions for crevice corrosion to occur. An oxygen rich environment is created by the accumulation of oxygen in the area of the pilot disc due to the breakdown of water into hydrogen and oxygen. Susceptible material in the right geometry with exposure to oxygen and high temperatures are the conditions which cause corrosion bonding in JAF S/RVs. There is extensive industry experience with corrosion bonding in the Target Rock two-stage S/RVs pilot assemblies.

As stated above, the three-stage S/RV pilot valve failed the As-Found testing low. Disassembly and testing was performed by NWS Technologies to determine cause. NWS concluded that the three-stage S/RV pilot was originally calibrated within the lower half of the acceptance range in the OEM specification. Calibration within the lower half of the acceptance range resulted in a greater setpoint drift, which ultimately resulted in the S/RV pilot being outside the allowable range.

Similar Events

Internal

- JAF LER-15-002 “Safety Relief Valve Upward Setpoint Drift,” June 1, 2015
- JAF LER-11-003 “Safety Relief Valve Setpoints Outside of Allowable Tolerances,” August 8, 2011
- JAF LER-09-005 “Safety Relief Valve Setpoints Outside of Allowable Tolerances,” June 22, 2009
- JAF LER-07-001 “Safety Relief Valve Setpoints Outside of Allowable Tolerances,” August 6, 2007

External

- Edwin I. Hatch Nuclear Plant, Unit 1: LER-16-004 “Safety Relief Valves As Found Settings Resulted in Not Meeting Tech Spec Surveillance Criteria,” May 26, 2016
- Edwin I. Hatch Nuclear Plant, Unit 2: LER-08-004 “Safety Relief Valves Allowable Test Range Exceeded Due to Setpoint Drift,” August 12, 2008

Corrective Actions

Future Corrective Actions

- Commence replacement of S/RVs with redesigned three-stage (RO23)

Previous Corrective Actions

- Installed Stellite 21 discs in all eleven S/RV pilot assemblies during refurbishment at the vendor facility
- Installed the S/RV Electric Lift System recommended by the Boiling Water Reactor Owner’s Group
- Installed enhanced insulation on the S/RVs

Safety Significance

Nuclear Safety

Actual Consequences

There were no actual consequences to the general safety of the public, nuclear safety, industrial safety, or radiological safety associated with this event.

Potential Consequences

The potential consequences of this event are associated with the over-pressurization of the Reactor Coolant Pressure Boundary. The S/RVs provide overpressure protection for the Reactor Coolant Pressure Boundary as required by the ASME Boiler and Pressure Vessel Code. Events similar to the one reported herein may be significant if design limits are challenged. The potential consequences of this event are considered low based on the operation and availability of the Electric Lift System.

Radiological Safety

There was no radiological safety impact associated with this event.

Industrial Safety

There was no industrial safety impact associated with this event.

References

- Issue Report No. 04077124, R22 SRV As-Found Testing Failures
- Issue Report No. 04082823, R22 3-Stage SRV As-Found Testing Failure
- JAF Technical Specifications