Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

May 12, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: Licensee Event Report 2015-002-00, Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No.: 50-293
License No.: DPR-35

LETTER NUMBER: 2.15.031

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2015-002-00, Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation, is submitted in accordance with 10 CFR 50.73.

Based on the complexity of the issue, the conditions were not determined to be immediately reportable upon discovering the concerns about on-demand performance. Initial conclusions were that the valves were operable based on dynamic test results. Subsequently, after considerable review and engineering evaluation of operational data and physical inspection results, the valves were declared inoperable. Accordingly, the time of discovery was established as March 12, 2015 when two valves were determined to have been inoperable.

This letter contains no commitments.

Please do not hesitate to contact Mr. Everett P. Perkins, Jr. (508) 830-8323, if there are any questions regarding this submittal.

Sincerely,

DE Noyes

David E. Noyes
Director, Regulatory and Performance Improvement

Attachment 1: Licensee Event Report 2015-002-00, Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation (6 pages)

IE22
NRR

cc:

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Attachment 1

Letter Number 2.15.031

Licensee Event Report 2015-002-00

Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation

(6 Pages)

NRC FORM 366 (02-2014)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

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4. TITLE Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation

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
03	12	2015	2015	002	00	05	12	2015	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OEPRATING MODE: N

10. POWER LEVEL: 100

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

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☒ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Mr. Everett P. Perkins, Jr. – Regulatory Assurance Manager	508-830-8323

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		06	30	2015

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

On March 12, 2015, after further evaluation of system performance of SRV-3A and SRV-3C, along with results of valve internal conditions identified during physical inspection, the valves were determined to have been inoperable for an indeterminate period during the last operating cycle. Specifically, SRV-3C was determined to be inoperable based on its on-demand performance at low reactor pressures, as well as the visual conditions that were identified during the inspection process. SRV-3A was considered inoperable based on it having similar internal indications as SRV-C when it was disassembled and inspected. SRV-3A was installed in May 2011 and SRV-3C was installed in October 2013.

Additionally, during an extent of condition review of historical SRV performance, the review identified on March 13, 2015 that SRV-3A had failed to open in response to three manual actuation demands on February 9, 2013.

At the time the valves were declared inoperable the reactor was at 100% power. The valves had been replaced in February 2015 during the forced outage relating to winter storm Juno. This event posed no threat to public health and safety.

NARRATIVE

BACKGROUND

On January 27, 2015, during winter storm Juno, Pilgrim Nuclear Power Station (PNPS) experienced a generator load reject and automatic reactor scram. During the pressure vessel cool-down period, a Main Steam Safety Relief Valve (SRV) appeared to have not fully opened when manually operated to control reactor pressure. Reactor vessel pressure did not lower as expected, reactor water level did not increase (swell) as expected, and there was minimal change in tailpipe temperature, which was not consistent with changes observed when other SRVs were opened. Operations maintained control of reactor pressure by alternate openings of other SRVs during plant cool-down.

Specifically, at 1015 hours, the first opening of SRV-3C was initiated when reactor pressure was 220 psig. When the operator placed the hand switch in the Open position there was no significant change in plant operating parameters. The operator initiated the second opening of SRV-3C at 1032 hours when reactor pressure was 262 psig, and again, there was no significant change in plant operating parameters, but a small torus water temperature increase was observed near the SRV-3C tailpipe outlet in the containment suppression pool. After the second attempt, Operations declared the valve SRV-3C, Serial Number (SN) 9, inoperable.

The SRVs are dual function Target Rock Corporation Model 0867F valves that are designed to operate in both safety mode and relief mode. The safety mode is automatically actuated at 1155 psig and involves successive opening of a first stage pilot valve, second stage pilot valve, and the main stage. The relief mode can be automatically actuated by the Alternate Depressurization System (ADS) which opens all four valves. Relief mode can also be initiated manually by the operator using any of the four SRVs individually or together. The relief mode of operation requires Direct Current power to energize a solenoid valve mounted locally on each valve. When the solenoid is energized, locally stored nitrogen is admitted to an air operator mounted on the valve. Nitrogen provides the motive force to open the second stage pilot valve and cause the SRV main stage to open.

PNPS has four, three-stage SRVs installed on the Main Steam lines. Each three-stage SRV contains a pilot (also called the first stage), a second stage, a main stage, and an air-operator. The pilot has main steam constantly applied to a bellows spring via a pressure sensing tube extending through the valve body. As the set pressure is reached, the bellows expands, opening the pilot disc and allowing steam to pass to the second stage. Steam pressure behind the second stage piston pushes the second stage disc open allowing steam to vent from behind the main stage piston to the containment suppression pool. Main steam pressure is present in front of the main stage piston, therefore, venting behind the piston creates a large differential pressure across the main piston causing it to stroke; pulling open the main stage disc to discharge steam and relieve system pressure. The air-operator is used to manually operate (open) the SRV below its setpoint pressure. When the air operator is pressurized, the operator plunger pushes directly against the second stage piston, opening the disc.

Subsequent to the plant reaching cold shutdown, SRV-3C, and another valve, SRV-3A, SN 4, were removed from the Main Steam system for testing, disassembly, inspection, and refurbishment. The valves met the Technical

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Specification required lift set-point acceptance criterion during testing. Based on the testing having demonstrated acceptable results within the Technical Specification acceptance criterion for valve opening and initial inspection results, an operability evaluation for each valve determined that the valves were operable and able to fulfill their intended safety function. However, after disassembly, during the inspection process, internal damage in the main stage piston section was observed that required further investigation.

EVENT DESCRIPTION

On March 12, 2015, after further engineering evaluation of performance of the valves and internal conditions identified during inspection, SRV-3A and SRV-3C were determined to have been inoperable for an indeterminate period during the last operating cycle. SRV-3C was determined to be inoperable based on its on-demand performance at low reactor pressures (first attempt at 220 psig; second attempt at 262 psig;), as well as the visual conditions that were identified during the inspection process. SRV-3A was considered inoperable based on it having similar internal indications as SRV-3C when it was disassembled and inspected. SRV-3A was installed in May 2011 and SRV-3C was installed in October 2013.

Additionally, during an extent of condition review of historical SRV performance, the review identified on March 13, 2015 that SRV-3A had failed to open in response to three manual actuation demands on February 9, 2013 with reactor pressures of 114, 101, and 98 psig.

The condition of the SRVs did not cause adverse results during the plant cool-downs, since the other installed SRVs operated as expected to control reactor pressure. In both cases, the reactor was placed safely in a cold shutdown condition.

Also, all the SRV's responded properly when called upon to function at higher reactor pressures (approximately 1000 psig or pressures close to that). In addition, following high pressure operation, the SRV's functioned over their entire range of operations.

CAUSE OF THE EVENT

The degradation mechanism is believed to be fretting wear (repeated cyclical rubbing) between the main stage piston and liner, increasing the friction in the stroke of the valve. Fretting is a time-dependent wear mechanism which develops while the valves are in-service in the plant. The fretting occurs because the piston-to-disk threaded connection loosens and the main steam line flow vibration drives the piston rings against the guide liner.

It is believed valve certification testing on a limited steam-flow test stand creates the conditions internal to the main body which allows the valve to develop a fretting wear condition while in-service. The gagged-valve test stand operations on a limited steam capacity test stand subjects the valve main stage to high opening force and high impact load. The high impact load increases potential loosening of the threaded joint between the main stage piston and the main disc stem (as-manufactured condition). When the valve is installed in the plant, normal system operation (steam flow) can cause the loosened piston to move (continuous, long-term, low amplitude vibration) relative to its liner. This movement may cause the piston rings to rub (fret) against the liner. Continued fretting may cause the rings to wear a groove into the liner; increasing potential binding friction against the piston when the valve strokes open. If sufficient binding friction has developed then the

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SRV opening stroke may not exhibit the typical rapid popping action when the valve opens at low reactor pressure where less opening force is available.

Target Rock Corporation issued an interim 10 *Code of Federal Regulations* (CFR) Part 21 report to the U.S. Nuclear Regulatory Commission concerning a potential test induced defect in the SRVs on March 16, 2015 (NRC Event # 50900) to provide notification that a multi-faceted investigation is ongoing to identify the cause of internal damage that could go undetected during production of new valves and refurbishment of valves that have been in-service. Although a root cause has not been determined at this time, sufficient facts have been established to warrant investigation of changes to current testing practices. This 10 CFR Part 21 notification was issued as a result of the PNPS SRV failures.

ADDITIONAL CONDITIONS

The SRVs also exhibit a spring "shortening" (or relaxation) phenomenon. GE SIL-196, Supplement 17 determined that Main Spring relaxation was caused by "*extreme dynamics encountered during limited flow testing....Valve dynamics under full flow conditions (i.e., discharge not gagged) are much less severe than those under limited flow conditions.*

The shortened spring is directly related to the overload condition created on the test stand that is potentially contributing to the loosened main stage piston connections. It is not unusual for a valve on the test stand to not fully close after a test stroke. Based on evaluations to date, a shortened main stage spring does not impact the valve over-pressure set-point, automatic actuation, or manual operation. Thus, this phenomenon does not directly impact the functionality of the valves

CORRECTIVE ACTIONS

Prior to restart from the forced outage related to winter storm Juno, SRV-3A and 3C were replaced with certified spare valves.

All SRV body/bases were removed from the system during the current refueling outage. In place of the four SRV's removed from the plant during the current refueling outage, PNPS has installed 2-stage SRV's. These will be used for Cycle 21.

Corrective actions will be captured in the PNPS corrective action program in Condition Report CR-PNP-2015-0561 and appropriate engineering documents.

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SAFETY CONSEQUENCES

The function of the safety relief valves is to limit peak vessel pressure during overpressure transients to satisfy the American Society of Mechanical Engineers Boiler and Pressure Vessel Code requirements for overpressure protection.

The Automatic Depressurization System (ADS) provides a means to rapidly depressurize the primary system to a pressure where low-pressure systems can provide makeup for core cooling. In the event of a small or medium break Loss of Coolant Accident, the ADS function would be required if the High Pressure Coolant Injection (HPCI) system is unable to maintain reactor water level. The postulated transients that require SRV actuation are described in Chapter 14 and Appendices R and Q of the Final Safety Analysis Report (FSAR). In accordance with plant Technical Specification 3.5.E.1 Limiting Condition for Operation, the ADS is required to be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a cold condition. In accordance with FSAR Section 4.4 Nuclear System Pressure Relief System sub-section 4.4.5 Description, "For depressurization operation, each relief valve is provided with a power actuated device capable of opening the valve at any steam pressure above 100 psig, and capable of holding the valve open until the steam pressure decreases to about 50 psig." Additionally, FSAR Table 6.3-1 Core Standby Cooling Systems Equipment Design Data Summary lists ADS valves as having a pressure range of 1,120 to 50 psig which spans from above normal operating pressure at rated core thermal power to below the pressure interlock for entry into Residual Heat Removal Shutdown Cooling.

During both cool-downs when SRV-A (February 2013) and SRV-C (January 2015) did not perform as expected, other SRVs were available to perform the necessary function of pressure control. During the event, both HPCI and the Reactor Core Isolation Cooling systems were used when needed to provide the functions of supplying makeup water to the vessel, providing adequate core cooling, and heat removal. Therefore, there was no adverse impact on the public health or safety.

REPORTABILITY

This report is submitted in accordance with:

- 10 CFR 50.73(a)(2)(v)(B) and 10 CFR 50.73(a)(2)(v)(D) – Event or Condition that Could Have Prevented Fulfillment of a Safety Function.
- 10 CFR 50.73(a)(2)(i)(B) – Operation or Condition Prohibited by Technical Specifications

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Since ADS is a single train system designed to fulfill the safety functions of removing residual heat and mitigating the consequences of an accident, the inoperability is reported in accordance with 10 CFR 50.73(a)(2)(v)(B) and 10 CFR 50.73(a)(2)(v)(D). Based on both valves having likely been unable to open at low pressure during plant operations the Limiting Condition for Operation would not have been met and the actions required by the Action Statement were not taken, so this condition is reported in accordance with 10 CFR 50.73(a)(2)(i)(B).

PREVIOUS EVENTS

Fretting is a newly identified phenomenon with the 3-stage Target Rock valves. No previous events have been reported by PNPS relating to this phenomenon. The only other events related to the 3-stage SRVs at PNPS were reported in the following LERs concerning leakage of the first stage pilot and setpoint drift:

LER 2011-007-00, Safety Relief Valve Declared Inoperable Due to Leakage

LER 2013-002-00 and -01, SRV-3B Safety Relief Valve Declared Inoperable Due to Leakage and Setpoint Drift

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

COMPONENTS	CODES
Relief Valve	RV
SYSTEMS	
Main Steam System	SB

REFERENCES

Condition Report CR-PNP-2015-0561 – SRV-3C appears to have not opened fully during manual operation.

Condition Report CR-PNP-2015-1520 – Testing, disassembly and inspection of SRV serial number (SN) 4 has been performed.

Condition Report CR-PNP-2015-1983 – A low pressure SRV opening problem has been identified by an extent review that is part of the ongoing investigation of potential SRV main stage fretting.