Entergy

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

May 29, 2012

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

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

Licensee Event Report 2012-001-00

LETTER NUMBER: 2.12.042

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2012-001-00, " Safety Relief Valves' Test Pressure Exceeded Setpoint Limits" is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please do not hesitate to contact Mr. Joseph R. Lynch, (508) 830-8403, if there are any questions regarding this submittal.

Sincerely,

Ralph A. Dodds, III

Ralph A. Dodds, III

FXM
Enclosure: Licensee Event Report 2012-01-00, “Safety Relief Valves' Test Pressure Exceeded Setpoint Limits”

cc:

Mr. William M. Dean
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USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

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NRR

Attachment 1
Letter Number 2.12.042

Licensee Event Report 2012-001-00

"Safety Relief Valves' Test Pressure Exceeded Setpoint Limits"
(5 Pages)

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER)	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 Service Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet 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
Pilgrim Nuclear Power Station	**05000293**	**1 OF 5**

4. TITLE
Safety Relief Valves' Test Pressure Exceeded Setpoint Limits

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 05000
03	**28**	**2012**	**2012**	**001**	**00**	**05**	**29**	**2012**	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING 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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Joseph R. Lynch, Licensing Manager	(508)-830-8403

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)*

On March 28, 2012, Pilgrim Nuclear Power Station (PNPS) was notified that three of four, two-stage Target Rock Safety Relief Valve (SRV) pilot assemblies exceeded the Technical Specification (TS) tolerance limit for routine set point pressure testing performed at the Wyle Laboratories test facility. Certified replacement three-stage SRVs were installed in the plant at the time Pilgrim was notified.

The cause of the as-found initial popping pressures exceeding the TS tolerance limit for the pilot valves was "setpoint variance" and "corrosion bonding."

Corrective action was taken to replace the two-stage Target Rock SRVs with certified tested replacement three-stage Target Rock SRVs during Refueling Outage 18. The corrective action also included revising the PNPS Safety Analysis and Technical Specifications (TS) to reflect a 3% tolerance on set pressure for the new SRVs.

The safety significance of the event was negligible and the condition posed no threat to public health and safety. An evaluation of the as-found set pressures and the potential increase in peak reactor pressure indicates that the increase would have been negligible and reactor vessel system integrity would not have been impaired.

NARRATIVE

BACKGROUND

The PNPS Pressure Relief System (PRS) is designed to prevent over-pressurization of the ASME Boiler and Pressure Vessel Code qualified nuclear steam supply system. The system consists of two spring safety valves (SSVs) and four safety relief valves (SRVs). These valves are installed in the main steam system piping upstream of the main steam isolation valves and are located within the drywell. The SSVs are self-actuating, provide over-pressure protection, and discharge directly to the drywell atmosphere when actuated. The SRVs augment the SSVs and are sized to prevent unnecessary actuation of the SSVs. The SRVs are self-actuating and discharge into the suppression pool through discharge piping connected to the valves. Each two-stage SRV consists of a pilot assembly and a main stage. The SRV pilot assembly provides the pressure sensing function and the main stage provides the pressure relieving function. The SRVs are also part of the Automatic Depressurization System (ADS). As part of the ADS, the SRVs are designed to automatically actuate as a result of a reactor depressurization permissive signal, and can also be manually actuated from the Control Room for depressurization.

Technical Specification (TS) 3.6.D.1, in effect during Cycle 18 (May 2009 to April 2011), identified that the nominal setpoint of the two-stage SRVs shall be selected between 1095 and 1115 psig and that all SRVs shall be set at this nominal setpoint ± 11 psi. The valves' nameplate setpoint was 1115 psig. Based on the tolerance limit of 11 psi (± 1%), a maximum pressure of 1126 psig was allowed. The established TS limit is stricter than the standard allowable relief valve setpoint drift range of ± 3% given in Section XI of the ASME Boiler and Pressure Vessel Code.

During Refueling Outage (RFO) 18, a design change was implemented to address SRV setpoint test pressure performance issues. This design change replaced the two-stage Target Rock SRVs (Model# 7567F) with three-stage Target Rock SRVs (Model# 0867F).

EVENT DESCRIPTION

On March 28, 2012 PNPS was notified that three of the four pilot valve assemblies installed during Cycle 18, had as-found popping pressures that exceeded the Cycle 18 maximum TS tolerance limit of 1126 psig for SRVs. The as-found popping pressures are provided below.

Pilot S/N	SRV Position	As-Found (psig)	Deviation (psi)	Deviation (%)
1207	RV-203-3A	1129	+14	1.3
1046	RV-203-3B	1191	+76	6.8
1049	RV-203-3C	1272	+157	14.1
1208	RV-203-3D	1126	+11	1.0

This event had no affect on plant operation. The plant was operating at 100% reactor power with the mode switch in run when the SRV test results were received.

CAUSE

Previous SRV setpoint test failure events and their associated root cause evaluations applicable to the two-stage SRVs were used to identify the root cause for this event. These root cause evaluations provide a comprehensive review of failure mechanisms applicable to these valves and identified that the SRV setpoint test failures resulted from "setpoint variance" and "corrosion bonding."

CORRECTIVE ACTION

Corrective action was taken to preclude future SRV setpoint pressure test failures during RFO 18. The corrective actions involved removal of all four two-stage Target Rock SRVs (Model# 7567F) and replacement with four new three-stage Target Rock SRVs (Model# 0867F). Based on implementation of the larger capacity three-stage SRV design, a new technical specification for SRV tolerance limits was implemented at the beginning of Cycle 19 (May 2011, see Amendment 235). In addition, this design change addressed revisions to the PNPS safety analyses for pressure transients.

Condition Report CR-PNP-2012-1453 was entered into the Corrective Action Program (CAP) to address this event.

SAFETY CONSEQUENCES

The condition posed no threat to public health and safety. A review of the applicable accident analysis revealed the following:

Minimum Critical Power Ratio (MCPR) Safety Limit – Fuel Clad Protection:

The limiting pressurization transient for Cycle 18 was inadvertent HPCI injection followed by a turbine trip. The Operating Limit Minimum Critical Power Ratio (OLMCPR) was established based on the analysis of this event to protect against exceeding the Technical Specification MCPR safety limit of 1.06. This analysis used an assumed SRV set pressure value of 1126 psig. A review of the graphical analysis results provided in the Supplemental Reload Licensing Report shows both the peak neutron and heat flux precede the opening of the relief valves. Therefore, the higher as-found relief valve set pressures do not influence the analysis results with respect to the MCPR operating or safety limit.

Overpressure Protection for Reactor Coolant Pressure Boundary (RCPB):

The MSIV Closure Flux Scram event used to design/verify adequate overpressure protection to avoid exceeding the ASME Code upset limit of 1375 psig is the MSIV closure event with a neutron flux scram. The valve position anticipatory scram is neglected in this analysis. This analysis used an assumed SRV set pressure of 1126 psig and a SSV set pressure of 1253 psig.

The Cycle 18 analysis for overpressure protection predicted a peak vessel pressure of 1298 psig (EC22573) which results in a margin of 77 psig to the ASME Code upset limit. For the as-found set pressures, the peak vessel pressure would increase but would not have exceed the acceptance limit of 1375 psig. This conclusion is based on the results of a sensitivity analysis documented in NEDE-30476 ("Setpoint Drift Investigation of Target Rock two-Stage Safety / Relief Valves," February, 1984) that estimates a 40 psig increase in the peak vessel pressure for a 10% increase in the set pressure for each of the four SRVs. This translates to a 4 psig increase for each 1% the set pressure increases. Since the average increase of the bank of four SRVs was 4.8%, the increase in peak vessel pressure is less than 20 psig and significant margin remains between the predicted peak vessel pressure and the ASME code limit of 1375 psig. The difference between the average SSV as-found and assumed set pressures is 1 psig and increases the peak pressure by less than 1 psig.

Anticipated Transient Without a Scram (ATWS):

For ATWS analysis, a 1500 psig peak vessel pressure limit is used. The Cycle 18 analysis resulted in essentially no margin to this limit; 1499.8 psig. Assuming that the same sensitivity of peak pressure to set pressure applies to the ATWS analysis as applies to the overpressure analysis, the high SRV and SSV set pressures would have resulted in a peak vessel pressure of 1520 psig. This exceeds the emergency limit used for ATWS analysis but is less than the hydrostatic pressure of 1560 psig and the stress analysis limit of 1875 psig for faulted conditions. Based on the hydrostatic test pressure described in the Updated Final Safety Analysis (UFSAR), system integrity would not have been impaired.

Loss of Feedwater – Core Coverage:

In the event of a loss of all feedwater with reactor vessel isolation, the RCIC system or its backup (HPCI) is required to maintain reactor water level above the top of the active fuel. After the initial discharge of stored energy from the reactor vessel to the suppression pool, a single SRV is capable of removing decay heat. Reactor pressure will be controlled at the lowest set pressure of the four SRVs. Core coverage is ensured as long as RCIC or HPCI can maintain rated flow at that set pressure. Despite the initial high set pressures, each SRV lift after the initial test was at or below 1126 psig. Since the RCIC system is capable of maintaining rated flow of 400 gpm with reactor pressure between the 150 psig and 1126 psig and HPCI is capable of much greater flow rates over the same pressure range, the capability to maintain reactor level is unaffected by the SRVs with initial lift as-found set pressures above 1126 psig.

Loss of Coolant Accident (LOCA) – Peak Clad Temperature:

Following a small break LOCA and vessel isolation, reactor pressure will remain high but controlled by cycling SRVs. The small break analysis for PNPS assumes that both HPCI and RCIC are unavailable. Core cooling is provided by the Alternate Depressurization System (ADS) in combination with low-pressure CSCS. Until ADS initiation, the loss of inventory from the vessel is a function of break area and the reactor pressure is controlled by the SRVs. After the initial discharge of stored energy from the reactor vessel to the suppression pool by multiple SRVs, a single SRV is capable of removing decay heat. If the set pressure of the controlling SRV is higher than the upper analytical limit of 1126 psig, the inventory loss from the vessel through the open SRV and pipe break will be greater than the analysis of record. Greater depletion of vessel inventory prior to ADS initiation will result in a longer period of core uncovery and a higher peak clad temperature. Since the set pressure of the SRVs on subsequent lifts is less than 1126 psig, the analysis of record is bounding with respect to the reactor pressure and inventory loss from the vessel prior to depressurization by ADS. Therefore, the existing LOCA analysis provides a bounding prediction of core uncovery time, clad heatup, and peak clad temperature.

REPORTABILITY

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because it was conservatively assumed that the as-found popping pressures could have been the pressure at which the relief valves would have operated if a high reactor pressure condition had occurred while the pilot assemblies were installed. The condition is assumed to have existed for a period greater than the 24 hours limiting condition of operations specified in Technical Specifications 3.6.D.1 and 2, as applicable to the two-stage SRVs installed during Cycle 18 (May 2009 to May 2011).

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station LERs related to relief valve setpoint test failures. This review identified LER 2001-004-000, LER 2004-001-00, LER 2005-003-00, LER 2007-004-00, and LER 2009-001 were written to address SRV setpoint tolerance test failures.

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Pilgrim Nuclear Power Station	05000293	2012 –	001 –	00	5 OF 5

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Valve, Relief	RV
SYSTEMS	CODES
Main Steam	SB