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

December 20, 2017

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

SUBJECT: Licensee Event Report 2017-007-01, Supplement to Potential Inoperability of Safety Relief Valve 3A

Pilgrim Nuclear Power Station
Docket No. 50-293
Renewed License No. DPR-35

LETTER NUMBER: 2.17.071

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2017-007-01, Supplement to Potential Inoperability of Safety Relief Valve 3A, is submitted in accordance with Title 10 Code of Federal Regulations 50.73. Revisions to the previously submitted LER are annotated by a vertical bar to the right of the wording changes.

If you have any questions or require additional information, please contact me at (508) 830-8323.

There are no regulatory commitments contained in this letter.

Sincerely,

for

Everett P. Perkins, Jr.
Manager, Regulatory Assurance

EPP/sc

Attachment: Licensee Event Report 2017-007-01, Supplement to Potential Inoperability of Safety Relief Valve 3A (4 Pages)

IE22
NRR

cc: Mr. David C. Lew
Acting Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
2100 Renaissance Blvd., Suite 100
King of Prussia, PA 19406-2713

Mr. John Lamb, Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8C2A
Washington, DC 20555

USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

Attachment

Letter Number 2.17.071

Licensee Event Report 2017-007-01

Supplement to Potential Inoperability of Safety Relief Valve 3A

(4 Pages)

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/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Pilgrim Nuclear Power Station	05000293	1 OF 4

4. TITLE Supplement to Potential Inoperability of Safety Relief Valve 3A

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	05000 N/A
04	24	2017	2017	- 007	01	12	20	2017	FACILITY NAME	DOCKET NUMBER
									N/A	05000 N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
N	☐ 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)
0	☐ 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. 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				

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

On April 24, 2017, during Refueling Outage 21 while performing testing on the Pilgrim Nuclear Power Station (PNPS) Safety/Relief Valves, a high resistance was measured across the solenoid pilot valve coil of SV203-3A. This solenoid pilot valve was replaced during Refueling Outage 21. After the solenoid pilot valve was removed it was transported to an offsite vendor for additional testing.

This LER supplement is being submitted to provide the NRC with additional information. PNPS continued to follow the testing performed by our offsite vendor and now has additional information to provide the NRC.

PNPS stated at the time that this event was reportable under 10 CFR 50.73(a)(2)(i)(B), as a condition prohibited by Technical Specifications and also, potentially reportable under 50.73(a)(2)(v)(B) and 50.73(a)(2)(v)(D), a condition that could have prevented fulfillment of a safety function needed to remove residual heat and mitigate the consequences of an accident. However, additional information provided by our offsite vendor and an engineering evaluation, support the conclusion that there was never a loss of safety function regarding SV203-3A. Therefore, this event was not reportable under 10 CFR 50.73(a)(2)(i)(B) nor under 10 CFR 50.73 (a)(2)(v)(B) or (D).

This event posed no threat to public health and safety.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Pilgrim Nuclear Power Station	05000-293	2017	- 007	- 01

NARRATIVE

BACKGROUND

The 2-stage pilot operated safety relief valve consists of two principle assemblies: a pilot valve section (top works) and the main valve section. The pilot valve section (first stage) is the pressure sensing and control element and the main valve (second stage) provides the pressure relief function. The first stage consists of a pilot-stabilizer disc assembly. The pilot is the pressure sensing member to which the stabilizer disc movement is coupled. Though not mechanically connected, a light spring keeps the stabilizer in contact with the pilot. A pilot preload spring permits set point adjustment of the valve and provides pilot seating force. The solenoid-operated pilot valve controls the pneumatic pressure applied to a diaphragm actuator which controls the relief valve directly. An accumulator is included with the control equipment for each relief valve to store pneumatic energy for relief valve operation. The second or main stage consists essentially of a large piston which includes the main valve disc, the main valve chamber, and a preload spring.

PNPS has four safety relief valves. Each of the four relief valves is equipped with an accumulator and check valve arrangement. These accumulators are provided to assure that the valves can be held open following failure of the nitrogen supply to the accumulators, and are sized to contain sufficient nitrogen for a minimum of 20 valve operations for each safety relief valve. Bottled gas can be used to manually recharge the accumulators associated with two safety relief valves. This capability was installed to address a potential loss of normal nitrogen supply to the accumulators which was identified during seismic reviews.

EVENT DESCRIPTION

On April 24, 2017, while performing testing on the Pilgrim Nuclear Power Station (PNPS) safety relief valves a high resistance was measured across the solenoid valve coil circuit of SV203-3A.

CAUSE OF THE EVENT

The degradation mechanism has been determined to be the solenoid pilot valve coil with high electrical resistance. Per input from our offsite vendor, corrosion of the SV203-3A crimp connections inside the coil created the high resistance indicated by a 9 VDC multimeter.

CORRECTIVE ACTIONS

Removed and replaced solenoid pilot valve assembly for SV203-3A.

ADDITIONAL INFORMATION FROM THE OFFSITE VENDOR

After performing multiple tests including destructive examinations of the SV203-3A solenoid valve coil the offsite vendor provided the following conclusions: Degradation of SV203-3A was limited to corrosion at the copper crimp connectors used to attach coil lead wires to the coil winding wire; the construction of the crimp connectors in SV203-3A included an insulating PVC sleeve material; although installation of the coil winding wires in the SV203-3A crimp connectors are inconsistent with industry practices, the coil winding wires were securely joined to the stranded lead wires at the stranded lead wire end of the crimp; the PVC sleeve material on the SV203-3A crimp connectors likely released chlorine, resulting in corrosion of the crimp connectors and wires; corrosion of the SV203-3A crimp connections created the high resistance indicated by a 9 VDC multimeter prior to application of higher voltage; application of voltage as low as 30 VDC was sufficient to overcome the corrosion product layer allowing the SV203-3A valve to actuate with no nitrogen pressure applied to the inlet port; application of higher voltages up to and including 125 VDC during electrical testing disturbed the corrosion layer sufficiently to allow a 9 VDC powered multimeter to measure SV203-3A coil resistance values representative of actual service conditions with respect to voltage. The valve operated as intended throughout the electrical testing despite the corrosion product accumulation on the crimp connectors; and subsequent to electrical testing, coil resistance values remained within acceptable limits.

SAFETY CONSEQUENCES

There are no consequences to the general safety of the public, nuclear safety, industrial safety and radiological safety from this event. The original concern was that there was a potential inoperability of the Automatic Depressurization System (ADS) which 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. An Engineering evaluation determined that the safety relief valve was fully operable at all times and remained available and capable of performing its intended safety function.

The engineering evaluation that was performed concluded that this event did not constitute a Safety System Functional Failure. (Reference NEI 99-02, Revision 7, Regulatory Assessment Performance Indicator Guideline, Section 2.2, Mitigating Systems Cornerstone, Safety System Functional Failures, Clarifying Notes, Engineering Analyses.) As such, this event will not be reported in the NRC Performance Indicator for Safety System Functional Failures since an engineering evaluation was performed which determined that the system was capable of performing its safety function.

No actions to reduce the frequency or consequence are necessary.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Pilgrim Nuclear Power Station	05000-293	2017	- 007	- 01

REPORTABILITY

PNPS believed at the time of the event that it was reportable under 10 CFR 50.73(a)(2)(i)(B), as a condition prohibited by Technical Specifications and also, potentially reportable under 50.73(a)(2)(v)(B) and 50.73(a)(2)(v)(D), a condition that could have prevented fulfillment of a safety function needed to remove residual heat and mitigate the consequences of an accident. However, additional information provided by our offsite vendor and an engineering evaluation, support the conclusion that there was never a loss of safety function regarding SV203-3A. Therefore, this event was not reportable under 10 50.73(a)(2)(i)(B) nor under 10 CFR 50.73 (a)(2)(v)(B) or (D).

PREVIOUS EVENTS

LER 2015-002-00, Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation

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

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

REFERENCES

CR-PNP-2017-5067

CR-PNP-2017-5386

CR-PNP-2017-6183