Entergy

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

November 4, 2016

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

SUBJECT: Licensee Event Report 2016-007-00, Manual Reactor Scram Due To Feedwater Regulating Valve Malfunction

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

LETTER NUMBER: 2.16.067

Dear Sir or Madam:

The enclosed Licensee Event Report 2016-007-00, Manual Reactor Scram Due To Feedwater Regulating Valve Malfunction, is submitted in accordance with 10 CFR 50.73.

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,

Everett P. Perkins, Jr.

Everett P. Perkins, Jr.
Manager, Regulatory Assurance

EPP/sc

Attachment: Licensee Event Report 2016-007-00, Manual Reactor Scram Due To Feedwater Regulating Valve Malfunction (4 Pages)

IE22
NRR

cc: Mr. Daniel H. Dorman
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NRC Senior Resident Inspector
Pilgrim Nuclear Power Station

Attachment

Letter Number 2.16.067

Licensee Event Report 2016-007-00

Manual Reactor Scram Due To Feedwater Regulating Valve Malfunction

(4 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)	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 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.

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4. TITLE Manual Reactor Scram Due To Feedwater Regulating Valve Malfunction

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
09	06	2016	2016	- 007	- 00	11	04	2016	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OEPRATING 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)(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)
10. POWER LEVEL	☐ 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)
91	☐ 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 NUMER *(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	SJ	CON	T351	No					

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 September 6, 2016 at 0827 EDT, with the reactor at approximately 91 percent core thermal power, operators manually scrammed the reactor when the benchmark for a reactor water level of +42 inches increasing was reached. Following the scram, all rods fully inserted and the Average Power Range Monitors were downscale, indicating the reactor was shut down. The Main Steam Isolation Valves closed on a Primary Containment Isolation Signal Group 1 isolation and were subsequently reopened to maintain reactor pressure.

The reason for the increasing reactor water level was the malfunction of Feedwater Regulating Valve 'A' (FRV 'A'). The reactor operator at the control panel placed FRV 'A' in remote manual in an attempt to stabilize the feedwater flow oscillations. This had no effect on the performance of FRV 'A'. The operators experienced feedwater flow oscillations from FRV 'A' that resulted in reactor water level high and low level alarms on the control panel. Shortly after this, the benchmark reactor water level of +42 inches increasing was reached and operators manually scrammed the reactor.

There was no impact to public health and safety.

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BACKGROUND

The purpose of the feedwater level control system (FWLC) is to automatically control feedwater flow to the reactor vessel, maintaining vessel water level within a small range during all operating modes. During power generation, the FWLC system regulates feedwater flow maintaining proper water level in the reactor vessel according to the requirements for steam separator carryover and carry under through the entire reactor operating range. The FWLC system ensures sufficient sub-cooled water to the reactor vessel for jet pump and recirculation pump net positive suction head (NPSH) and to maintain normal operating temperatures during power operation.

The primary FWLC system purpose is to maintain reactor vessel water level within a small range during all operational modes. FWLC instrumentation measures reactor vessel water level, feedwater flowrate and exit steam flowrate. During automatic operation these three measurements are used to control feedwater flow.

The ability to maintain vessel level within a small range during load changes is accomplished by the three-element control signal. The total steam flow signal and the total feedwater flow signal are fed into a proportional amplifier. The output from this amplifier is the mismatch between the input signals (steam flow-feedwater flow error signal). If steam flow is greater than feedwater flow, the amplifier output is increased from its normal value, causing the system to increase feed flow to balance with steam flow. This amplifier output is fed to a second proportional amplifier that also receives a reactor vessel water level signal. Adding the reactor vessel water level signal to the steam flow-feedwater flow error signal results in a three-element control signal, which is fed to the level controller.

In May of 2015, during the refueling outage, new connectors on the encoder were installed. The new connectors were crimped onto the existing wires and inserted into the connection on the encoder. The valve was successfully post-work tested and returned to service. During the refueling outage, valve testing on the FRV 'A' was completed, including a pressure drop test on the actuator. However, due to a poorly written test procedure, anomalies were not properly identified or addressed as described below.

During a reactor downpower on October 20 and 21, 2015, a packing leak was identified on FRV 'A'. This resulted in exposing the valve manual locking device to unfavorable conditions which eventually led to degradation of the valve locking mechanism couplings and bearings. The valve packing was adjusted to reduce the packing leak.

EVENT DESCRIPTION

At 07:55 on September 6, 2016, the control room received an instantaneous core thermal power (CTP) blinking annunciator light. This was the first indication to the control room operators that there was a problem.

At 08:00 operations received an alarm for thermal power due to suspect value (feedwater flow).

At 08:10 operators placed FRV 'A' in remote manual. Three unsuccessful attempts were made to manually control the valve.

At 08:14 operators were dispatched to the Condenser Bay to lockup FRV 'A' in accordance with procedure. The control room operators established scram benchmarks of +15 inches lowering, +42 inches increasing reactor

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water level, or +/- 10 percent power swings.

At 08:19 the control room operators received reactor water high level alarm.

At 08:20 an operator at the local digital control panel identified Error message for Friction Warnings and proceeds to FRV 'A' to lockup the valve.

At 08:21 operators at FRV 'A' cannot operate the mechanical operator due to degraded condition of the actuator.

At 08:25 FRV 'A' goes full open while operators are located at the valve.

At 08:26:12, a "Friction Failure" error message was received at the local digital control panel.

At 08:26:28 operators in the control room received an alarm for reactor water level high.

At 08:27 operators manually scrammed the reactor at a water level of +42 inches increasing and received APRM downscale with all rods in.

CAUSE OF THE EVENT

The direct cause of this event is the neutral common wire lost connection to the encoder stab on the stepper motor resulting in a loss of feedback to the control system for the valve.

The causes of the event are that the work package quality standards for critical maintenance work order package planning were not known or understood by some planning personnel. This resulted in the inadequate installation, during the May 2015 refueling outage, of the neutral common wire that became loose and lost connection between the encoder and the digital controller. In addition, the air operated valve testing procedure for the FRVs did not contain criteria for valve packing friction or stem load evaluation. This resulted in a missed opportunity to identify and correct the valve packing leakage condition and the degraded condition of the mechanical locking device prior to the event.

CORRECTIVE ACTIONS

The following corrective actions were performed:

1) Replaced the FRV 'A' valve stem
2) Replaced the FRV 'A' and FRV 'B' valve packing
3) Refurbished the FRV 'A' manual actuator
4) Reassembled the encoder electrical connector
5) Performed a pull test of each individual wire in the encoder electrical connection satisfactorily before the connector was reinstalled and sealed in place for both FRV 'A' and 'B'.

The following corrective actions are currently planned to be performed:

1) Generate a maintenance planning standards checklist for critical maintenance work package quality and train maintenance planners on use of the new checklist.
2) Train station personnel in problem identification and resolution.
3) Train operating crews and station leadership on the event, causes and corrective actions.

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

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NARRATIVE

SAFETY CONSEQUENCES

The actual consequences were a loss of ability to control FRV 'A' which resulted in increasing reactor water level and an operator-initiated manual scram. There were no other actual consequences to general safety of the public, nuclear safety, industrial safety, and radiological safety for this event.

The potential consequences to general safety of the public, nuclear safety, industrial safety, and radiological safety of this event if the operator-initiated manual scram did not occur were minimal. Had the manual scram not been initiated, in a short time there would have been an automatic turbine trip, feedwater pump trip, and Primary Containment Isolation Signal Group 1 isolation from the high reactor water level condition. These automatic actions would have immediately caused an automatic reactor Scram.

The conditional core damage probability of this event has been estimated to be 3.49E-06. This is the value associated with a Main Steam Isolation Valve (MSIV) isolation initiator. The risk from the actual event was less than this calculated value because the Scram was manually initiated prior to the turbine trip and other automatic actions. The scram was considered uncomplicated as defined in Nuclear Energy Institute 99-02.

Therefore, based on the above 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)(iv)(A). Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section including 50.73(a)(2)(iv)(B)(2), general containment isolation signals affecting containment isolation valves in more than one system or multiple MSIVs.

PREVIOUS EVENTS

A review of Pilgrim Nuclear Power Station Licensee Event Reports for the past five years did not identify any similar occurrences of manual scrams being initiated by feedwater oscillations.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for Components and Systems referenced in this report are as follows:

SYSTEMS	CODES
Feedwater System	SJ

REFERENCES:

CR-PNP-2016-6635