Entergy

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Michael A. Balduzzi
Site Vice President

March 11, 2004

U.S. Nuclear Regulatory Commission
Attention: 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 2004-001-00

Letter Number: 2.04.020

Dear Sir:

The enclosed Licensee Event Report (LER) 2004-001-00, "Target Rock Relief Valves' Test Pressures Exceed Technical Specification Tolerance Limit," is submitted in accordance with 10 CFR 50.73

This letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

Sincerely,

Eric Olson for M. Balduzzi

Michael A. Balduzzi

FXM/dm

cc: Mr. Hubert J. Miller
Regional Administrator, Region 1
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Senior NRC Resident Inspector

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INPO Records

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1. NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104
LICENSEE EVENT REPORT (LER) (See reverse for number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If 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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TITLE (4)
Target Rock Relief Valves' Test Pressures Exceed Technical Specification Tolerance Limit

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000
01	16	04	2004	001	00	03	11	2004	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)

	20.2201(b)		22.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	22.2202(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(3)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(3)(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)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Bryan Ford – Licensing Manager	(508) 830-8403

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SB	RV	T020	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 16, 2004, Wyle Laboratories notified Pilgrim Station of test results indicating that three Target Rock relief valve pilot assemblies exceeded the Technical Specification (TS) tolerance limit of 1115 psig ± 11 psi (± 1%) during routine testing at the test facility.

The cause for the as-found relief pressures exceeding the TS tolerance limit is believed to be setpoint drift resulting from corrosion bonding of the pilot valve assembly disk and seat. The corrosion bonding most likely occurred while the pilot valve assemblies were in storage awaiting testing. These three pilot valve assemblies were removed during the May 2003 refueling outage, stored at Pilgrim Station for approximately seven months, then shipped to Wyle for refurbishment and ASME Code Testing. Certified replacement relief valve pilot assemblies were installed in the plant at the time the test was performed.

The condition posed no threat to public health and safety.

BACKGROUND

The Pilgrim Station Pressure Relief System (PRS) is designed to prevent over-pressurization of the ASME Boiler and Pressure Vessel Code qualified nuclear steam supply system. The PRS consists of two safety valves and four two-stage relief valves. These valves are installed in the main steam system piping upstream of the main steam isolation valves and are located within the Drywell. The safety valves are self-actuating, provide over-pressure protection, and discharge directly into the Drywell atmosphere when actuated. The relief valves augment the safety valves and are sized to prevent unnecessary actuation of the safety valves. The relief valves are self-actuating and discharge into the suppression pool through discharge piping connected to the valves. Each two-stage relief valve consists of a pilot assembly and main stage. The pilot assembly provides the pressure sensing function and the main stage provides the pressure relieving function.

The relief valves are also part of the Automatic Depressurization System (ADS). As part of the ADS, the valves are designed to automatically actuate as a result of a depressurization permissive signal, and can be manually actuated from the control room for depressurization.

Technical Specification (TS) 3.6.D.1 specifies that the nominal setpoint of the relief valves shall be selected between 1095 and 1115 psig and that all relief valves shall be set at this nominal set point ± 11 psi. The valves' nameplate setpoint is 1115 psig. Therefore, based on the tolerance limit of 11 psi (±1%), a maximum pressure of 1126 psig and a minimum pressure of 1104 psig are allowed. The established TS limit is stricter than the standard allowable relief valve setpoint drift range given in ASME Boiler and Pressure Vessel Code, Section XI, which is ± 3%.

Since the early 1980s, increased initial lift pressure (or upward setpoint drift) has been an industry concern applicable to the two–stage relief valves found in BWRs. Industry investigation of relief valve reliability problems revealed the primary cause of upward setpoint drift in the two-stage relief valves was corrosion bonding of the pilot valve disk to its seat. Three different design modifications were found to reduce or counteract the corrosion bounding: 1) installation of ion beam implanted platinum pilot valve disks, 2) the installation of stellite 21 pilot valve disks, and 3) the installation of additional pressure actuation switches. PNPS implemented changes to install the stellite 21 pilot valve disk design in the mid 1980 time period.

NRC review of upward setpoint drift is documented in NRC Regulatory Issue Summary 2000-12, Resolution of Generic Safety Issue B-55, "Improved Reliability of Target Rock Safety Relief Valves," and Generic Issue 165, "Spring Actuated Safety and Relief Valve Reliability." The NRC staff found that the industry has significantly improved valve performance and is continuing efforts to evaluate and improve performance, as necessary. Therefore, the staff found no new requirements were necessary and that existing quality assurance, maintenance rule, and code testing requirements were adequate to ensure reliable valve performance in the future.

All four pilot assemblies for main steam relief valves RV-203-3A, 3B, 3C and 3D were removed during the May 2003 Refueling Outage (RFO-14). These pilot assemblies were removed after main steam line flood up. None of the pilot valve assemblies had experienced steam leakage problems while installed during the May 2001 to April 2003 operating cycle (Cycle–14). The valves were manufactured by Target Rock Corporation, model 7567F.

To ensure the availability of a spare pilot assembly after RFO-14, one of the assemblies (serial number 1208) was sent to Wyle Labs for testing and refurbishment. This pilot assembly was tested on June 5, 2003 and results revealed slight air leakage in the assembly and an as-found popping pressure of 1092 psig (below the minimum TS tolerance limit of 1105 psig). At the time of the test, four certified pilot assemblies were installed in the plant, and no evidence was available to indicate that a defect existed while the valve was installed. Therefore, the failure to meet the TS 3.6.D.1 tolerance limit was determined to be not reportable.

Approximately seven months later, within the allowable one-year ASME Code limit for performing pressure actuation testing, the remaining three pilot assemblies removed during RFO-14 were sent to Wyle for testing, refurbishment, and certification.

EVENT DESCRIPTION

On January 16, 2004 Pilgrim Station was notified that three main steam relief pilot valve assemblies had as-found popping pressures that exceeded the maximum TS tolerance limit of 1126 psig. The as-found popping pressures were 1138 psig (serial number 1040), 1235 psig (serial number 1207), and 1173 psig (serial number 1025).

CAUSE

The root cause evaluation identified that the cause for the as-found popping pressures exceeding the TS tolerance limit was setpoint drift resulting from corrosion bonding of the pilot valve disk to its seat. The corrosion bonding most likely occurred while the pilots were in storage awaiting testing.

CORRECTIVE ACTION

The following corrective actions were taken:

Two of the four pilot valve assemblies (serial numbers 1208 and 1040) that were removed during RFO-14 were tested, refurbished, and certified to have setpoints within the TS limits.

The following corrective actions are planned:

Test the Target Rock pilot assemblies sooner following removal and sealing the assemblies to prevent oxygen and moisture from entering the pilot seat area while the valve is in storage after removal.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Perform a microscopic evaluation of the pilot valve seat and disc then complete an internal report to document inspection results.

Review latest industry data with regard to Target Rock setpoint drift and document results.

SAFETY CONSEQUENCES

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

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

The limiting pressurization transient for Cycle 14 was the Feedwater Controller Failure. The Operating Limit Minimum Critical Power Ratio (OLMCPR) was established based on the analysis of the event to protect against exceeding the Technical Specification MCPR safety limit of 1.06. This analysis used an assumed relief valve setpoint of 1126 psig. A review of the graphical analysis results provided in the analysis shows both the peak neutron and heat flux precede the opening of the relief valves. Therefore, the higher popping pressures of the relief valves pilot assemblies do not influence the analysis results with respect to the operating or safety limit MCPR.

Overpressure Protection for the Reactor Coolant Pressure Boundary:

The event used to verify adequate overpressure protection to avoid exceeding the ASME Code limit of 1375 psig for upset conditions is the Main Steam Isolation Valve (MSIV) closure event with a neutron flux scram. The MSIV position anticipatory scram is neglected in the analysis. The analysis used an assumed relief valve setpoint of 1126 psig. The Cycle 14 analysis for overpressure protection predicted a peak vessel pressure of 1301 psig. Given the as-found popping pressures of the pilot assemblies, the peak vessel pressure will increase but will not exceed 1375 psig. This conclusion is evident given the results of a Pilgrim Station relief valve setpoint sensitivity analysis that estimates a 40 psig increase in the peak vessel pressure, assuming a 10% increase in the setpoint for each of the four relief valves. The average increase in the as-found popping pressures of the four relief valve pilot assemblies was less than 3%.

Loss of Feedwater - Core Coverage:

In the event of a loss of all feedwater and reactor vessel isolation, the Reactor Core Isolation Cooling (RCIC) system or its backup, the High Pressure Coolant Injection (HPCI) System is required to maintain reactor water level above the Top of Active Fuel (TAF). After the initial discharge of stored energy from the reactor vessel to the suppression pool, a single relief valve is capable of removing decay heat. Reactor pressure will be controlled at the lowest as-found setpoint of the four relief valves (i.e. 1092 psig, pilot assembly serial number 1208). The RCIC system is capable of maintaining rated flow of 400 gpm with reactor pressure between the 150 psig and 1126 psig while the HPCI System is capable of much greater flow rates over the same pressure range. Therefore, the analysis that evaluated capability to maintain reactor level was unaffected by the pilot assemblies with as-found popping pressures exceeding TS tolerance limits.

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

Following a small break LOCA and vessel isolation, reactor pressure will remain high and is controlled by cycling (opening and closing) relief valves. The small break analysis for Pilgrim Station assumes both HPCI and RCIC are unavailable. Core cooling is provided by the Automatic Depressurization System (ADS) in combination with low-pressure Core Stand-by Cooling Systems (CSCS). Until ADS initiation, the loss of inventory from the vessel is a function of break area and reactor pressure as controlled by cycling relief valves. After the initial discharge of stored energy from the reactor vessel to the suppression pool by multiple relief valves, a single relief valve is capable of removing decay heat. Since the lowest as-found popping pressure of the four relief valve pilot assemblies was 1092 psig (serial number 1208), the analysis of record is bounding with respect to 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.

Anticipated Transient Without Scram (ATWS):

The ATWS rule requirements are prescriptive in nature and do not require plant specific analysis of each abnormal operational transient accompanied by an ATWS. ATWS rule compliance is mechanistic and does not require creation or maintenance of event specific ATWS analysis.

During the mid to late 1980 timeframe, General Electric (nuclear steam system supplier) studied ATWS scenarios at Pilgrim Station to develop recommendations to enhance plant and operator response to ATWS events. The GE ATWS analyses predicted that peak vessel pressure would be 1497 psig. An evaluation of the as-found condition indicates that in the worst case, the peak vessel pressure resulting from the as-found popping pressures is estimated to be slightly greater than 1500 psig for approximately 30 seconds, well below the stress analysis limit of 1875 psig for faulted conditions (i.e., conditions associated with extremely low probability postulated events which may impair the integrity and operability of the nuclear system to the point where public safety is involved). It also should be noted that the estimated peak vessel pressure does not exceed the hydrostatic test pressure (125% of design pressure or 1560 psig) required by the ASME code to verify RCS System integrity prior to initial plant startup. Therefore, given an ATWS event considering the as-found relief valve popping pressures, the estimated peak vessel pressure did not increase significantly and based on engineering judgment, system integrity would not have been impaired.

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REPORTABILITY

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because it is conservatively assumed that the as-found popping pressures could have been the pressures that 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 hour limiting condition for operation specified in Technical Specification 3.6.D.2 for the relief valves.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station LERs. The review focused on relief valve test related LERs that were submitted since 1984. The review identified LER 91-014-01, LER 93-011-00, LER 99-004-00 and LER 2001-004-00.

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