Entergy

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Michael A. Balduzzi
Site Vice President

July 6, 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 2004-003-00

Letter Number: 2.04.053

Dear Sir:

The enclosed Licensee Event Report (LER) 2004-003-00, "Target Rock Relief Valves' Test Pressures Exceeding Technical Specification Tolerance Limit," is submitted in accordance with 10 CFR 50.73

This letter contains no commitments.

Please do not hesitate to contact Bryan Ford, (508) 830-8403, if there are any questions regarding this report.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi

DWE/dm

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INPO Records

IE22

1. NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104
LICENSEE EVENT REPORT (LER)	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 Exceeding 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
05	06	2004	2004	003	00			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 May 6, 2004, Pilgrim Station was notified of test results for the pilot valves of two main steam relief valves that had exceeded the Technical Specification limit of 1115 +/- 11 psig during testing at the Wyle Laboratories test facility. Certified replacement relief valve pilot assemblies were installed in the plant at the time of the notification.

The root cause evaluation identified the most probable cause of initial high as-found popping pressures exceeding the tolerance limit was corrosion bonding of the pilot valve disc and seat. The corrosion bonding most likely developed while the pilot valves were in service. The root cause evaluation identified poor fitting insulation as a contributing cause.

Corrective action taken and planned includes replacing the pilot valves with certification tested pilot valves, reworking the insulation, overhauling the pilot valves, and purchasing new insulation.

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 (EIIS Code RV). These valves are installed in the main steam system (EIIS Code SB) 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 (i.e. pilot valve) 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 relief valves are designed to automatically actuate as a result of a depressurization permissive signal and can be manually actuated from the control room for the depressurization function.

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 setpressure 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 tolerance of +/-3% in Section XI of the ASME Boiler and Pressure Vessel Code.

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 revealed the primary cause of upward setpoint drift in the two-stage relief valves was corrosion bonding of the pilot valve disk to the valve seat. Three different design modifications were found to reduce or counteract the corrosion bonding: 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. Pilgrim Station implemented changes to install the Stellite 21 pilot valve disc design in the mid-1980 timeframe.

NRC review of 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." In the review, 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.

The main steam relief valves were manufactured by the Target Rock Corporation, model 7567F.

In the January 2004 timeframe, the discharge pipe temperature of relief valves RV-203-3A (pilot serial number 1054) and RV-203-3D (pilot serial number 1049) increased, indicating pilot valve leakage.

Pilgrim Station initiated and completed a planned shutdown on March 22, 2004. The purpose of the shutdown was to replace the pilot valves of RV-203-3A and -3D and other planned maintenance. The pilot valves were replaced with certified pilot valves, and Pilgrim Station returned to commercial operation on March 25, 2004. The pilot valves (serial numbers 1049 and 1054) were subsequently sent to the Wyle Laboratories test facility for testing, refurbishment, and certification. The testing was within the allowable one-year ASME Code limit for performing pressure actuation testing.

For pilot valve 1049, a pre-lift leakage test was performed at 1010 +/- 10 psig and the valve exhibited no leakage. During the subsequent pressure testing, the as-found popping pressure of the valve was 1198 psig (initial popping pressure), 1117 psig, 1115 psig, and 1114 psig. Following the as-found pressure testing, a solenoid test was performed to verify the valve could have been actuated using the solenoid, and the test demonstrated the solenoid was able to actuate the valve. After the solenoid test, a post-lift leakage check was performed at 1030 +/- 10 psig and the valve exhibited no leakage.

For pilot valve 1054, a pre-lift leakage test was performed at 1010 +/- 10 psig and the valve exhibited leakage. During subsequent pressure testing, the as-found popping pressure of the valve was 1144 psig (initial popping pressure), 1135 psig, 1124 psig, and 1124 psig. Following the as-found pressure testing, a solenoid test was performed to verify the valve could have been actuated using the solenoid, and the test demonstrated the solenoid was able to actuate the valve. After the solenoid test, a post-lift leakage check was performed at 1030 +/- 10 psig and 600 psig, and the valve exhibited leakage.

EVENT DESCRIPTION

On May 6, 2004, Pilgrim Station was notified of test results for the pilot valves of two main steam relief valves that had exceeded the Technical Specification limit of 1115 +/- 11 psig. The as-found popping pressures were 1198 psig (serial number 1049) and 1144 psig (serial number 1054). Certified replacement relief valve pilot valves were installed in the plant at the time of the notification.

The condition was identified while operating at 100 percent reactor power with the reactor mode selector switch in the RUN position. The reactor vessel pressure was about 1030 psig with the reactor water temperature at the saturation temperature for that pressure.

CAUSE

The root cause evaluation identified that the most probable cause of initial high as-found popping pressures exceeding the TS tolerance limit was corrosion bonding of the pilot valve disc and seat. The corrosion bonding most likely developed while the pilot valves were in service. The pilot valves (serial numbers 1049 and 1054) were installed in RV-203-3A and RV-203-3D during the 2003 refueling outage. Review of the as-found pressure testing results provides evidence of corrosion bonding based on the fact that the initial popping pressure is relatively high because the steam force exerted on the pilot valve disc must overcome the bond between the pilot valve disc and seat. As previously described, pilot valve 1049 experienced the highest initial popping pressure, which was greater than the ASME test criteria (nameplate setpressure of 1115 +/- 3% psig). Pilot valve 1054 met the ASME test criteria (i.e. nameplate setpressure of 1115 +/- 3% psig).

The poor fitting insulation was identified as a contributing cause. The poor fitting insulation was noted during the initial inspection of the Drywell, after the planned shutdown on March 22, 2004. G.E. Service Information Letter 196 (supplement 16) describes the importance of proper fitting insulation on Target Rock relief valve operation, reducing the propensity of pilot valve leakage, and eliminating the conditions conducive to corrosion bonding. With the exception of the new NUKON blanket insulation installed on RV-203-3B during the 2003 refueling outage, the NUKON blanket insulation was installed on RV-203-3A/C/D beginning in November 1993. The NUKON insulation was installed because the previous mirror style insulation had degraded due to improper handling and personnel traffic in the Drywell. A number of straps, which affix the insulation to the valve body, were found broken or detached on RV-203-3A and RV-203-3D during the March 2004 outage. NUKON tends to sag as it ages, causing the insulation to move away from the valve body.

EXTENT OF PROBLEM

The corrosion bonding of the pilot valve disc and seat is limited to the 2-stage model 7567F Target Rock relief valves. Pilgrim Station has eight of these pilot valves, four installed in the relief valves and four spares available for replacement of an installed pilot valve(s) and/or refurbishment. Corrosion bonding has been discovered in these valves in the past, most recently reported in LER 2004-001-00 and addressed in the related root cause and corrective actions. There are no other valves of this design installed in any other Pilgrim Station systems. The insulation on all four installed relief valves was inspected while Pilgrim Station was shut down in March 2004. The insulation on the pilot valves installed in the RV-203-3B and RV-203-3C were noted to be in relatively good condition. The insulation on RV-203-3B was new, installed during the 2003 refueling outage. The insulation on RV-203-3C was found intact, with no large gaps or missing pieces.

CORRECTIVE ACTION

The following corrective actions were taken. The pilot valves were removed from service and replaced with certification tested pilot valves. The insulation on RV-203-3A and RV-303-3D was reworked. The insulation on RV-203-3B and RV-203-3C was inspected and precautionary insulation enhancements were made even thought the insulation was in relatively good condition.

The following corrective actions are planned:

- Inspecting the seats of the pilot valves (1049, 1054) and comparing inspection results with previous results to determine additional actions to preclude recurrence.
- Overhauling and certification testing of the pilot valves.
- Purchasing new insulation for the relief valves for better insulating properties.
- The Drywell closeout inspection procedure will be revised to include steps and guidance for inspections of the relief valves' insulation during closeout inspection.

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 15 is a Feedwater Controller Failure. The Operating Limit MCPR (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 valves do not influence the analysis results with respect to the OLMCPR or MCPR safety limit.

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 15 analysis for overpressure protection predicts a peak vessel pressure of 1305 psig. Given the as-found popping pressures of the pilot assemblies, the peak vessel pressure would increase but would 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 pilot valves (serial numbers 1049 and 1054) was less than 10%.

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. 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 would be controlled at the lowest setpoint of the relief valves. Subsequent sequential test runs on the two pilot valves indicate a decreasing popping pressure, e.g. as-found test pressures of 1198 psig (initial), 1117 psig, 1115 psig, and 1114 psig for pilot valve 1049. 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 the capability to maintain reactor level was unaffected by the as-found popping pressures of the pilot valves (serial numbers 1049 and 1054).

Loss of Coolant Accident (LOCA) – Fuel Peak Clad Temperature. Following a small break LOCA and vessel isolation, reactor pressure would remain high and controlled by cycling (opening and closing) the relief valves. The Pilgrim small break analysis assumes both HPCI and RCIC systems are unavailable. Reactor core cooling is provided by the automatic depressurization system (ADS) in combination with low pressure Core Standby Cooling Systems. Until ADS initiation, the loss of vessel inventory is a function of break area and reactor pressure as controlled by cycling relief valves. After the initial discharge of stored energy from the vessel to the suppression pool by multiple relief valves, a single relief valve is capable of removing decay heat. If the popping pressure of the controlling relief valve is greater

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

than the upper analytical limit of 1126 psig, the vessel inventory loss through the open relief valve and pipe break will be greater than the analysis. A greater depletion of inventory before initiation of ADS will result in a longer period of core uncovery and a higher fuel peak clad temperature. Because the popping pressure of the pilot valve serial number 1049 (and 1054) was less than 1126 psig after the initial popping pressure, the analysis of record is bounding with respect to the reactor pressure and inventory loss from the vessel before depressurization by ADS. Therefore, the existing analysis provides a bounding prediction of core uncovery time, fuel clad heatup and peak clad temperatures.

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. Beginning with fuel cycle 15, the Pilgrim Station licensed thermal power increased from 1998 MW to 2028 MW. The throat diameter was increased on each of the four main steam relief valves, which provides an increase of about 7.5% in steam flow capacity per valve. During the licensing phase for the thermal power increase, a revised ATWS analysis was performed. The analysis resulted in a peak vessel pressure of 1495 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 a faulted condition (i.e., condition associated with extremely low probability postulated events which may impair reactor coolant system (RCS) integrity to the point where public safety might be impacted). 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 integrity prior to initial plant startup. Therefore, given an ATWS event considering the as-found popping pressures of the pilot valves, the estimated peak vessel pressure would not increase significantly and based on engineering judgment, system integrity would not have been impaired.

REPORTABILITY

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because it was conservatively assumed the as-found popping pressures could have been the pressures at which RV-203-3A/D would have actuated for the pressure relief function if a high reactor pressure condition had occurred while the pilot valves 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 2001. The review identified LER 2001-004-00 and LER 2004-001-00.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES. The EIIS codes for this report are as noted in the BACKGROUND portion of this report.