Entergy

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Pilgrim Nuclear Power Station
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Kevin H. Bronson
Site Vice President

August 13, 2007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

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

Licensee Event Report 2007-004-00

LETTER NUMBER: 2.07.057

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2007-004-00, "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 contact Stephen Bethay at (508) 830-7800, if there are any questions regarding this subject.

Sincerely,

Steve Bethay for KHB

Kevin H. Bronson

MJG/dl
Enclosure

cc: Mr. Samuel J. Collins
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NRR

LICENSEE EVENT REPORT (LER)

(See reverse for number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE(3)
PILGRIM NUCLEAR POWER STATION	05000-293	1 of 6

TITLE (4)
Target Rock Relief Valves' Test Pressure Exceeded Limit Due to Corrosion Bonding and Setpoint Variance

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	DOCKET NUMBER
									N/A	05000
06	13	2007	2007	004	00	08	13	07	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 Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(vi)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Stephen J. Bethay – Director, Nuclear Assessment	(508) 830-7800

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	Target Rock	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 June 13 2007, Pilgrim Station was notified that three of the four Target Rock relief valve pilot assemblies exceeded the Technical Specification (TS) tolerance limit of 1115 ±11 psi (± 1%) during routine testing at the Wyle Laboratories test facility. Certified replacement relief valve pilot assemblies were previously installed in the plant.

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

Corrective actions taken include replacing the pilot valves with certified tested replacements.

The conditions posed no threat to public health and safety because an evaluation of the effect of the as-found set pressures concluded that no design or licensing basis limits would have been exceeded had the SRVs been required to operate.

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

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 to 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 a 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 relief valves are designed to automatically actuate as a result of a depressurization permissive signal, and can also 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 setpoint ± 11 psi. The valves' nameplate setpoints are 1115 psig each. 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 that the standard allowable relief valve setpoint drift range of ± 3% given in Section XI of the ASME Boiler and Pressure Vessel Code.

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

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 investigations of relief valve reliability problems revealed that one of the primary causes 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 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. PNPS implemented changes to install the Stellite-21 pilot valve disk design in the mid 1980 timeframe. Another cause of upward setpoint drift was a phenomenon know as "setpoint variance" identified in an EPRI document published in 1996, Technical Report TR-105872, "Safety and Relief Valve Testing and Maintenance Guide." The information contained within the report indicated that most main steam safety valve high lift failures were outside of the ± 1% allowable tolerance but within the ± 3% vendor design tolerance. The report went on to state that these failures were principally driven by the close tolerance between Technical Specification requirements and the actual ability of the valve to perform within the required pressure band. The report identified this phenomenon as "setpoint variance" which occurs when the first lift is between 1% and 3% and the following lifts are within 1%. This is a well recognized occurrence that the National Board of Boiler and Pressure Vessel Inspectors has observed for many years. The cause of this phenomenon has not been identified.

Additionally, 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." 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. 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 Refueling Outage 16.

EVENT DESCRIPTION

On June 13, 2007 Pilgrim Station was notified that three of the four pilot valve assemblies previously installed had as-found popping pressures that exceeded the maximum TS tolerance limit of 1126 psig. The as-found popping pressures were 1131 psig (serial number 1207), 1137 psig (serial number 1025), and 1126.7 psig (serial number 1049). The as-found popping pressure of serial number 1054 was 1123 psig.

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 "set point variance" as the most probable cause of initial high as-found popping pressures exceeding the TS tolerance limit for two of the pilots (1207 and 1025) because these two pilots had a step change in set pressures between the first and subsequent lifts during both as-left and as-found testing. Minor corrosion bonding is the cause of the initial high as-found popping pressure for the third pilot (1049).

CORRECTIVE ACTION

All four pilots were removed and replaced with spare pilots. The spare pilot valve assemblies that were installed were tested and certified to be within 1% of the normal set point required by Technical Specifications.

An electric lift system which will provide a non-safety backup to the safety-related mechanical lift is scheduled to be installed during the next refueling outage.

Any planned corrective actions will be implemented consistent with the Pilgrim Station corrective action program.

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 16 was the 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):

MSIV Closure Flux Scram is the event used to design/verify adequate overpressure protection to avoid exceeding the ASME Code upset limit of 1375 psig. The valve position anticipatory scram is neglected in this analysis. This analysis used an assumed SRV set pressure value of 1126 psig. The Cycle 16 analysis for overpressure protection predicted a peak vessel pressure of 1298 psig which results in a margin of 77 psig to the ASME code upset limit. Given the as-found set pressures, the peak vessel pressure will increase slightly (less than 4 psig) but peak vessel pressure will not exceed the acceptance limit of 1375 psig. This conclusion is based on a sensitivity analysis documented in NEDE-30476 "Setpoint Drift Investigation of Target Rock 2-Stage Safety/Relief Valves", dated February 1986. The sensitivity analysis estimated that a 10% increase in set pressure for each of the 4 SRVs would produce a 40 psig increase in peak vessel pressure. The 40 psig peak vessel pressure change translates to a 4 psig increase for each 1% of set pressure increase. The average increase in as-found set pressure of the bank of 4 SRVs was 0.3 % (less than 1%). Therefore, the peak vessel pressure increase would be less than 4 psig and significant margin remains between the predicted peak vessel pressure and the ASME code limit of 1374 psig.

Anticipated Transient Without a Scram (ATWS)

Beginning in Cycle 15, the licensed thermal power level for PNPS was increased from 1998 MWth to 2028 MWth. During the licensing phase of the power uprate, a revised ATWS analysis was performed that evaluated the MSIV closure and pressure regulator failed open (PRFO) events. The PRFO ATWS event resulted in a higher peak vessel pressure (1495 psig) than the MSIV closure ATWS pressure (1464 psig). This analysis used the General Electric computer code ODYN. The analytical set pressures used were 1126 for three SRVs and 1136 for one SRV. The average set pressure for the bank of four SRV was 1128.5 psig. Given the 0.9 psig difference between the average as-found set pressure of 1129.4 and the ATWS analysis value of 1128.5 psig, the peak vessel pressure increase would be small.

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

Anticipated Transient Without a Scram (ATWS) con't

Using the same method as used for the over pressure protection evaluation, the increase would be less than 4 psig and the calculated peak vessel pressure was expected to be below the emergency limit of 1500 psig. Therefore given an ATWS event considering the as-found relief valve opening pressures, the estimated peak vessel pressure did not increase significantly based on engineering judgment, and 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 Reactor Core Isolation Cooling (RCIC) system or its backup, the High Pressure Coolant Injection (HPIC) 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 SRV is capable of removing decay heat. Reactor pressure will be controlled at the lowest set pressure of the four SRVs (i.e. 1123 psig for pilot assembly serial number 1054). A pilot assembly typically exhibits a higher setpoint during the initial opening and a slightly lower setpoint during subsequent lifts. 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 pilot assemblies with as-found popping pressures exceeding the TS limits.

Loss of Coolant Accident (LOCA) – Peak Clad Temperature

Following a small break LOCA and vessel isolation, reactor pressure will remain high and is controlled by cycling the SRVs. The small break analysis for Pilgrim Station assumed that both HPCI and RCIC were unavailable. Core cooling is provided by the Alternate Depressurization System (ADS) in combination with low-pressure Core Standby Cooling Systems (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. Since the lowest as-found popping pressure of the four relief valve pilot assemblies was 1123 psig (serial number 1054), the analysis on record is bounding with respect to reactor pressure and inventory loss from the vessel prior to depressurizations by ADS. Therefore the existing LOCA analysis provides a bounding prediction of core uncovery time, fuel clad heat-up and peak clad temperatures.

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 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 LERs related to relief valve tests submitted since 2001. The review identified LER 2001-004-000, LER 2004-001-00, and LER 2005-003-00.

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

Licensing Correspondence Control Sheet

Letter Number: 2.07.057

Licensing Engineer (LE): MJ Gatslick

Letter Number Cross-References: N/A

Verify and Distribute Letter to the organization as needed: MJG
LE / Initials

Update Log as needed: MJG
LE / Initials

Commitment Review per EN-LI-110 completed: MJG
LE / Initials

Results: No new commitments were identified.

Posting Required? (10 CFR 19.11(a)(4): Y / $\boxed{N}$

If Yes, Posting completed by ______________ on ______________
LE / Initials Date

Action Required?: Y / $\boxed{N}$

Action Items:

Tracking Number	Action