Entergy

Entergy Nuclear Generation Co.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

Mike Bellamy
Site Vice President

10 CFR 50.73

June 20, 2001
ENGC Ltr. 2.01.067

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

Docket No. 50-293
License No. DPR-35

Dear Sir:

The enclosed Licensee Event Report (LER) 2001-003-00, *"ESF Actuations due to Invalid Water Level Indications,"* is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments. Corrective actions associated with the root cause analysis will be implemented consistent with the Pilgrim Station Corrective Action Program.

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

Sincerely,

Mike Bellamy

JRH/
Enclosure: LER 2001-003-00

cc: Mr. Hubert J. Miller
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Sr. NRC Resident Inspector
Pilgrim Nuclear Power Station

INPO Records
700 Galleria Parkway
Atlanta, GA 30339-5957

IE22

NRC Form 366 (6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

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 5

TITLE (4)

ESF Actuations due to Invalid Water Level Indications

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
04	21	2001	2001	003	00	06	20	2001	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)							
			20.2201 (b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
POWER LEVEL (10)	000		22.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)(B)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)(D)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Bryan S. 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

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 April 21, 2001, Pilgrim Station was cooling down to begin refueling outage 13. At approximately 8:52 A.M. an invalid Group 1 isolation signal closed the main steam isolation valves. The reactor vessel pressure had decreased to approximately 100 psig when vessel level indication notching caused the isolation and subsequent Group 2 and Group 6 isolations. The Reactor Core Isolation Cooling and High Pressure Coolant Injection systems were used in the pressure control mode to reduce reactor pressure.

This condition occurred after a reactor scram that was part of a controlled shutdown to begin refueling outage 13. The reactor mode selector switch was in the SHUTDOWN position. The reactor vessel pressure varied from about 100 psig to about 230 psig with the water temperature at the saturation temperature for those pressures.

The cause of the notching is believed to be inadequate venting of the pump casing and associated piping during recent control rod drive pump maintenance. Corrective actions completed before reactor restart and procedure clarifications should preclude recurrence.

This condition posed no threat to public health and safety.

BACKGROUND

Noncondensible gas may be introduced into the reactor water level instrumentation either when shutdown or during operation. When shutting down from normal power operations, the reactor is depressurized and noncondensible gas moves as bubbles through the reference leg causing false level indication. This phenomena is called notching and has occurred during several reactor shutdowns and depressurization activities at Pilgrim Station from 1991 to 1993. NRC Information Notice No. 92-54, NRC Generic Letter No. 92-04 and NRC Bulletin 93-03 addressed the resolution of the inaccurate water level indication concern.

Extensive investigation by the Boiling Water Reactor (BWR) Owners Group in 1992 and 1993 determined that noncondensible gases in the reference leg of the level transmitters coming out of solution while the reactor is being depressurized causes reactor water level spiking. The presence of the noncondensible gas bubbles in a level transmitter's reference leg displaces water in the reference leg. Therefore, "indicated" level is temporarily greater than actual level. When the gas bubble reaches a horizontal run of piping, it does not measurably affect the weight of the water in the reference leg and the indicated level returns to normal. The piping configuration of the transmitter's reference leg and the amount of gas present determine the characteristic shape of the indicated level, sometimes referred to as "notching" due to its square wave shape.

A backfill system design was developed based on the industry effort and installed by Pilgrim Station to resolve the noncondensible gas buildup in the "cold" reference legs of the vessel level indication transmitters. The reference leg backfill system is designed to prevent gas-saturated water from migrating from the condensing chambers down the reference leg to the level transmitters by providing a continuous low flowrate backfill supplied by Control Rod Drive (CRD) charging water. Small leaks in the reference leg at couplings or valve packing allow the gas-saturated water in the condensing chamber to migrate down the transmitter's reference leg. The backfill system maintains a flowrate greater than the estimated system leakage rate preventing the downward flow of gas-saturated water to the reference legs associated with the transmitters. The system was designed and installed to comply with NRC Bulletin 93-03. For the system to function properly, the reference leg must be purged of gas-saturated water prior to pressurization.

Since the implementation of the hardware and procedure changes associated with the backfill system for the water level indication transmitters'reference legs, there had been no water level indication notching during reactor vessel depressurization prior to this event.

EVENT DESCRIPTION

On April 21, 2001, Pilgrim Station was cooling down to begin refueling outage (RFO) 13. At approximately 5:34 A.M. a manual scram was initiated to complete the shutdown. At 8:52A.M. when the vessel pressure had decreased to approximately 100 psig, Pilgrim Station experienced an invalid high water level Group 1 isolation signal due to water level indication notching. The main steam isolation valves (MSIVs) closed isolating the reactor vessel from the main condenser heat sink. As

the cooldown process continued, additional spikes occurred in vessel level indication. At approximately 9:41 A.M. notching caused an invalid low water level indication producing a scram signal and a Group 2 and 6 and reactor building isolation system (RBIS) isolation signals. Both series of invalid signals were caused by notching in the reactor water level indication reference leg condensing chambers. By 10:25 A.M. all level instruments were behaving normally.

The initial Group 1 signal closed the MSIVs and they remained closed while the cause for the Group 1 isolation signal was investigated. The reactor pressure began to slowly rise and when the pressure had reached approximately 230 psig at approximately 10:25 A.M., the reactor core isolation cooling (RCIC) system was manually initiated in the pressure control mode. At approximately 11:38 A.M. RCIC was temporarily shutdown and at approximately 11:43 A.M. the high-pressure coolant injection (HPCI) system was placed in pressure control mode to reduce reactor pressure and to continue cooldown with the main steam isolation valves closed.

The NRC was notified at approximately 10:19 A.M. on April 21, 2001, in accordance with 10 CFR 50.72(b)(3)(iv). A problem report was written to document this event and to initiate a root cause analysis.

This condition posed no threat to public health and safety.

CAUSE

The cause for use of RCIC and HPCI and the isolation signals were notching induced invalid water level indications. The cause of the notching is believed to be introduction of air into the reference legs due to maintenance activities on the "B" control rod drive pump between January and March of 2001. Inadequate venting of the pump casing and associated piping are believed to have permitted the introduction of air which caused the notching.

CORRECTIVE ACTION

Corrective actions taken include the following:

- Performed high-pressure backfill of the reference legs associated with condensing chambers 12A and 12B prior to restart.
- Replaced both flow elements for the continuous backflow system.
- Inspected and replaced the reference leg backfill inlet filters.
- Replaced metering valve for "A" reference leg.
- Verified leaktightness of valves and fittings associated with the reference legs prior to restart.
- Verified the adequacy of the flowrate for the continuous backfill system.
- Revised PNPS Procedure 2.2.87 providing additional guidance for venting and draining the CRD pumps.

Corrective action to be taken:

- Review PNPS Procedure 2.1.1 to determine whether additional clarifications are required to improve backfilling of the condensing chambers.
- Revise PNPS Procedures 3.M.2-12.3 and 3.M.2-12.4 to improve backfilling of condensing chambers.

ACTION TO PREVENT RECURRENCE

Procedure revisions and training will clarify how the reactor vessel water level reference legs are to be isolated before maintenance and how they are to be pressure-filled following maintenance to preclude air entrainment. Additionally, although not thought to be the cause of the event, improvements have been made in the backfill system operating procedures and component maintenance. Continued use of the current backfill system with the enhancements should preclude recurrence.

SAFETY CONSEQUENCES

The condition posed no threat to public health and safety.

A water level indicating notching event occurred while depressurizing the reactor vessel to begin RFO 13. The reactor had been scrammed for more than three hours and the pressure had decreased to approximately 100 psig when false high water level indication initiated a Group 1 isolation signal. The main steam isolation valves closed and the reactor pressure slowly increased to 230 psig. RCIC and HPCI were manually initiated to reduce reactor pressure.

Industry analyses have concluded that automatic safety system actuations will occur at pressures well above pressures where water level notching has been observed. Even though it is believed that air was introduced into the reference legs during operation, safety system functions were not impacted and therefore operable.

Operations personnel have been trained on water level indication, notching, and the backfill modification installed to prevent the migration of air in the water level indication reference leg. The operators were aware that the variations in level indication were invalid and responded appropriately to the isolation signals received. The actual water level was normal at the time and alternate level indication was available to assess reactor conditions. Both HPCI and RCIC, functioned as designed and when the pressure decreased they were removed from service. The MSIVs were reopened to complete the cooldown.

Both RCIC and HPCI were available during the transition from normal reactor pressure to approximately 100 psig, while low-pressure coolant injection (LPCI) and core spray (CS) were also available to mitigate any accident consequences. Operators are trained and procedures address the identification and response to water level notching. A significance determination assessment screened the notching incident as low safety significance since the event occurred after a reactor scram, while at low pressure and while several core standby cooling

systems were available. The significance determination (NRC green indicator) was based on the reasons discussed above, the multiple systems available, and the narrow period of time when notching could impact level indication.

REPORTABILITY

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A).

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) for water level notching. LER 91-008-00 and LER 92-004-00 were written to describe false water level indication initiating Group I isolations during reactor depressurization, prior to hardware modifications.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Water level indicating control	LIC

SYSTEMS	CODES
High Pressure Coolant Injection	BG
Reactor Core Isolation Cooling	BN