Entergy

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

Michael A. Balduzzi
Site Vice President

November 24, 2003

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 2003-006-00

Letter Number: 2.03.136

Dear Sir:

The enclosed Licensee Event Report (LER) 2003-006-00, "Reactor Coolant Pressure Boundary Leakage due to Reactor Vessel Nozzle Weld Crack Propagation," is submitted in accordance with 10 CFR 50.73

This letter contains no commitments.

Please feel free to contact me if there are any questions regarding this subject.

Sincerely,

FOR M.A. BALDUZZI 11/24/03

Michael A. Balduzzi

DWE/dm

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

Senior NRC Resident Inspector

Mr. Travis Tate
Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
1 White Flint North Mail Stop: 0-8B-1
11555 Rockville Pike
Rockville, MD 20555-001

INPO Records

IE22

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)
Reactor Coolant Pressure Boundary Leakage due to Reactor Vessel Nozzle Weld Crack Propagation

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
10	01	2003	2003	006	00	11	24	2003	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	0

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)	X	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)		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
B	AA	NZL	C719	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 October 1, 2003 at about 0430 hours, a reactor coolant pressure boundary leak from a reactor vessel nozzle-to-cap weld was discovered during a planned visual inspection of the Drywell. The inspection was a planned activity to identify the source(s) of Drywell leakage.

The leak was caused by an incipient crack or crevice condition remaining in the weld after repair welding performed as part of the nozzle-to-cap fabrication welding in 1977. Subsequent crack propagation continued through-wall by an interdendritic stress corrosion cracking mechanism due to high residual weld stresses in the Inconel weld metal.

Corrective action taken included a nozzle weld repair consisting of a full structural weld overlay.

The event posed no threat to public health and safety.

BACKGROUND

The reactor vessel (RPV) is located within the Drywell (primary containment) and consists of a hemispherical upper head section, cylindrical mid-section, and hemispherical lower head section. The RPV is equipped with appurtenances including nozzles that connect the RPV to process fluid piping and instrumentation and control devices.

The leakage of water and/or steam within the Drywell is detected and monitored by the Drywell sumps and atmosphere sampling systems. There are two Drywell sumps, the equipment sump and floor sump. The equipment sump receives drainage from identified sources and has a cover and raised curb to prevent floor drainage from intermixing with the equipment drainage. The floor sump receives drainage from unidentified sources. The sumps are routinely pumped in accordance with procedure. Leakage is monitored and calculated, and the leakage rate from each sump and the combined leakage rate for both sumps are reviewed to detect increases. Identified or unidentified leakage in excess of normal background amounts is potentially indicative of a reactor coolant leak. In addition to the Drywell sumps, the Drywell atmosphere sampling system provides for monitoring airborne radioactivity levels within the Drywell and consists of two permanently installed panels. Each panel is capable of monitoring the Drywell atmosphere for particulate, halogen, and gaseous activity.

Technical Specification 3.6.C governs coolant leakage anytime irradiated fuel is in the RPV and coolant temperature is greater than 212 degrees Fahrenheit. Technical Specification 3.6.C.1.a specifies that operational reactor coolant leakage be limited to no pressure boundary leakage, </= 5 gpm unidentified leakage, </= 25 gpm total leakage when averaged over any 24 hour period, and </= 2 gpm increase in unidentified leakage within any 24 hour period when in the RUN mode. Technical Specification 3.6.C.1.d specifies that when any pressure boundary leakage is detected, be in at least HOT SHUTDOWN within the next 12 hours and be in COLD SHUTDOWN within the next 24 hours.

Pilgrim Station re-started from the 2003 refueling outage in May 2003. The Drywell leakage rates observed in the June – August 2003 timeframe were relatively steady, at about 2.4 gpm (total leakage rate). During the month of September 2003, the total leakage rate gradually increased from about 2.5 gpm to about 3.5 gpm (September 28, 2003). An increase was noted in the Drywell unidentified leak rate, from about 1.45 gpm on September 25, 2003 to about 1.71 gpm on September 26, 2003 (and occasionally thereafter). During these and subsequent periods of operation, the unidentified leakage rate, the total leakage rate, and any increase in the unidentified leakage rate remained within the limits specified in Technical Specification 3.6.C.1. Because of ALARA considerations and requirements governing primary containment, a visual inspection within the Drywell is not normally possible during power operation. A visual inspection of the Drywell, with the RPV pressurized, was a planned activity that was part of a planned shutdown scheduled to begin on September 29, 2003.

On September 29, 2003, at about 0500 hours, the planned shutdown was initiated for replacement of the unit auxiliary transformer and planned maintenance including the identification and correction of unidentified leak sources within the Drywell. Routine and required surveillance tests and a thermal backwash of the main condenser were performed as part of the shutdown activities. The main turbine-generator was manually tripped as planned at 2300 hours on September 29, 2003.

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An initial Drywell entry for the identification of leak sources began at about 0200 hours on September 30, 2003. Inspection beneath the reactor vessel identified two streams of water from a source(s) above the inspection site. Inspections continued for the location of the source(s) of the water. Later on September 30, 2003, at about 1500 hours, a sample of water was obtained from the Drywell floor sump (unidentified leakage) and by about 1600 hours the sample results indicated reactor water leakage. Inspection activities continued in order to determine the source of the leakage.

The reactor was shut down with all control rods in the fully inserted position by 0418 hours on September 30, 2003. The recirculation system loop 'A' motor-generator set/pump was stopped at 1757 hours as part of preparations for the shutdown cooling mode of operation. The residual heat removal (RHR) system train 'A' was put into service in the shutdown cooling mode at 1821 hours. The reactor vessel water temperature was less than 212 degrees Fahrenheit by 1840 hours on September 30, 2003.

Plant conditions existing just prior to the event on October 1, 2003 included the following. The reactor was shut down with all control rods in the fully inserted position. The reactor vessel was at atmospheric pressure. The reactor vessel water temperature was less than 212 degrees Fahrenheit, with the temperature being maintained in a range of 80 - 120 degrees Fahrenheit. The reactor water level was being maintained in a range of 80 - 90 inches (shutdown range).

EVENT DESCRIPTION

On October 1, 2003 at about 0430 hours, a reactor coolant pressure boundary leak from the RPV N-10 nozzle-to-cap weld was discovered during a planned visual inspection of the Drywell. The leak was located when the insulation installed on the nozzle was removed during the inspection activities to determine the source of reactor coolant leakage that was indicated as a result of sample analysis at about 1600 hours on September 30, 2003. The inspection of the Drywell for the source(s) of Drywell leakage was a planned activity.

The NRC Operations Center was notified of the discovery in accordance with 10 CFR 50.72 at 1112 hours on October 1, 2003.

CAUSE

A historical background for nozzle N-10 is provided as follows. The nozzle was modified in 1977 to prevent cracking due to cyclic thermal stress resulting from the return of relatively colder water to the reactor vessel (nozzle N-10) from the control rod drive system (General Electric Company Service Information Letter 200). The modification consisted of cutting and isolating the (then) existing control rod drive (CRD) system hydraulic return line to nozzle N-10 and rerouting the CRD return line to the CRD cooling water header. The modification included the removal of the safe end and thermal sleeve from nozzle N-10 and installation of an Inconel cap to nozzle N-10. The installation weld was prepared with an Inconel-to-carbon steel butter weld and an Inconel-to-Inconel weld that attached the cap to nozzle N-10. The cap base material was comprised of Inconel 600 (ASME SB-166, UNS number N06600). The cap was butt welded to the nozzle using Inconel 82 (ASME SFA-5.14, UNS number N06082, AWS classification ERNiCR-3) and Inconel 182 (ASME SFA-5.11, UNS number W86182, AWS classification ENiCrFe-3) weld materials.

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Post work radiography identified defects in the weld and the weld was repaired. Final post work testing of the modification was via nondestructive examination and hydrostatic testing in 1977. Crouse Nuclear Energy Services welding procedure specification (WPS) CG-56 and a Pilgrim Station temporary procedure were used to accomplish the nozzle-to-cap butt weld. The weld was most recently examined in the 1999 refueling outage with no crack-like indications noted. The examination was accomplished using the best available manual examination methods (manual ultrasonic techniques with qualified inspectors). Because the weld was a Category D and DM (dissimilar metals) weld and most recently inspected in the 1999 refueling outage, the next examination of the weld was scheduled for the 2005 refueling outage under the Boiling Water Reactor Vessel Internals Project (BWRVIP-75) guidelines.

The cause of the leakage from RPV nozzle N-10 was through-wall leakage from the nozzle-to-cap butt weld. The through-wall leakage was caused by an incipient crack or crevice condition remaining in the weld after repair welding performed as part of the nozzle-to-cap fabrication welding in 1977. Subsequent crack propagation continued through-wall by an interdendritic stress corrosion cracking mechanism due to high residual weld stresses in the Inconel 82/182 weld metal as a result of the repair in 1977. The crack was located entirely within the Inconel weld metal.

The flaw characteristics and location are consistent with crack initiation that occurred entirely within the weld metal due to several potential causes related to the repair of fabrication weld flaws, including weld solidification cracking, incomplete fusion, or other tight crevice conditions.

The following conditions existed where this cracking occurred:

- The materials (base metal and weld metal) were susceptible to various stress corrosion cracking mechanisms.
- The fabrication weld was repaired by excavation that included the removal of a portion of the consumable insert.
- The weld repair in 1977 required inner diameter back-grinding to remove slumped weld metal and/or flaw indications caused by solidification cracking, incomplete fusion, or surface oxidation.
- There is minimal protection from stress corrosion cracking mechanisms via hydrogen water chemistry due to the location of the nozzle cap and stagnant flow conditions.

There were no notable contributing causes.

CORRECTIVE ACTION

Corrective action taken included an automated, full structural weld overlay allowed by an NRC approved relief request. The weld overlay was installed with Inconel 52 weld metal, which is highly resistant to stress corrosion cracking. The weld overlay process also imparts a compressive residual stress due to the welding process, which prevents further crack growth.

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The N-10 nozzle-to-cap weld was scheduled for a routine inservice ultrasonic examination in the 2005 refueling outage under BWRVIP-75 guidelines. This examination is performed to the updated requirements for such examinations as required by ASME Code Section XI Appendix VIII and the Performance Demonstration Initiative (PDI) program that is administered by the Electric Power Research Institute (EPRI). The PDI program is a collaborative effort to develop alternative workable methodologies to implement the ultrasonic examination Supplements to ASME Section XI Appendix VIII.

Ultrasonic examinations performed using the PDI methodologies have improved the capability to detect flaws related to stress corrosion cracking mechanisms and flaws that occur entirely within the weld metal over the methodologies used previously. It was concluded that an examination of this type would have increased the likelihood of detecting this weld cracking if it had not first progressed through-wall and been discovered on October 1, 2003.

A scope expansion evaluation was performed concurrently with the weld overlay repair of the N-10 nozzle so that appropriate actions could be taken as needed. The evaluation assessed the critical attributes related to the nozzle N-10 weld failure and considered the entire reactor primary system weld population to determine welds with similar attributes. The evaluation concluded that no immediate augmented inspection of any additional nozzle-to-safe end or pipe welds was necessary and that the currently planned programmatic inspections of Category D welds in accordance with the established Pilgrim Station BWRVIP-75 Program is sufficient.

SAFETY CONSEQUENCES

This event posed no threat to public health and safety and was not risk significant.

The leakage from the nozzle N-10 weld and other leak sources was less than the technical specifications limits for unidentified leakage and total leakage (combined unidentified and identified).

- If the leakage from the nozzle or other leak sources had increased during power operation such that the unidentified leakage rate increased more than the administrative limit of 1 gpm in a 24 hour period or became greater than the administrative limit of 2.5 gpm, both of which are less than the technical specifications limit(s), a plant shutdown would have been initiated and completed.

- If the leakage from the nozzle and/or other leak sources had increased during power operation such that the unidentified leakage rate increased more than 2 gpm in a 24 hour period or became greater than 5 gpm, a shutdown would have been initiated and completed as required by Technical Specification 3.6.C.

The maximum Drywell unidentified leakage rate experienced during power operation before the discovery of the nozzle N-10 weld leak was about 1.7 gpm, the majority of which was found to be from a flange and vent valves located on the reactor vessel head vent piping, and some of which was from minor valve packing leaks. The leakage rate was within the makeup capacity of the feedwater and/or control rod drive systems, and was within the makeup capacity of the core standby cooling systems (CSCS). The CSCS consist of the high pressure coolant injection (HPCI), automatic depressurization, residual heat removal (RHR), and core spray systems. Although not part of the CSCS, the reactor core isolation cooling (RCIC) system is capable of providing high-pressure core cooling, similar to the HPCI system.

Nozzle N-10 is a 4-inch nominal internal diameter penetration located about 440 inches above the reactor vessel zero reference. This penetration is about 42.5 inches below the CSCS reference instrument zero (i.e. –42.5 inches), which places the nozzle centerline about 83.7 inches above the level corresponding to the top of the active fuel zone.

- If the nozzle N-10 weld, in which the through-wall leak occurred, had experienced a complete circumferential failure, an unisolable 4-inch diameter pipe break would have occurred. During power operation when the reactor vessel is pressurized, the break would have resulted in an increase in the Drywell atmosphere pressure. The increase in Drywell pressure would have been detected by the Drywell pressure instrumentation that function to monitor the Drywell pressure and automatically initiate a shutdown of the reactor, close applicable primary containment isolation valves, and initiate the HPCI system and low-pressure CSCS systems (RHR and Core Spray). The 4-inch diameter of the nozzle is equal to an opening area of 0.087 square feet. For a pressurized fluid leak, the design makeup capability of the HPCI system has an upper bound of 0.10 square feet. Therefore, the leakage from a complete circumferential failure of the weld during power operation was within the makeup capability of the HPCI system. The operation of the HPCI system would function to depressurize the reactor vessel well below the pressure at which the low-pressure CSCS systems (RHR and/or Core Spray) would function for core cooling and reactor vessel water inventory control. Therefore, the fuel would have remained covered by water and no fuel damage would have occurred if a complete circumferential failure of the nozzle N-10 weld had occurred during power operation.

- Additional protection for immediate and long-term core cooling in the event of a complete circumferential failure of the nozzle N-10 weld would be provided by the automatic depressurization system (ADS) in combination with the low-pressure CSCS systems (RHR and/or Core Spray). The automatic (or manual) actuation of the ADS could result in a brief period when the fuel would not be completely covered with water and consequently, result in the heatup of the fuel cladding before fuel submergence and core cooling is restored by the low-pressure CSCS (RHR and Core Spray). During this period the fuel clad temperatures are analyzed to remain well below the limits established in 10 CFR 50.46.

The nozzle N-10 weld leak was discovered after the nozzle insulation was removed during the inspection to identify the reactor coolant leak source. Although the insulation was found discolored and with some damage due to impingement, the damage was not significant. Based on the as-found condition of the insulation, the unidentified leakage trend before the discovery of the N-10 nozzle weld leak, and the other sources of Drywell leakage that were found and corrected while shut down, it is not possible to determine precisely when the leak from the nozzle-to-cap weld progressed completely through the weld.

A risk assessment was conducted of the nozzle N-10 weld leak. The Pilgrim Station probabilistic risk assessment (PRA) was used as part of the assessment. While the leak was small and presented no challenges to the plant, the potential for a medium LOCA may have been increased. Several conservative sensitivity cases were conducted involving the medium LOCA initiator frequency. The maximum credible delta core damage probability was about 5E-07.

REPORTABILITY

This report was submitted in accordance with 10 CFR 50.73(a)(2)(ii)(A) because the nozzle N-10 weld leak, a defect, was not acceptable under ASME Section XI.

SIMILARITY TO PREVIOUS EVENTS

A review for similarity was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs that involved leakage from the reactor coolant pressure boundary (RCPB).

The review identified LER 85-004-00 that reported RCPB leakage from a socket weld (12-BC-14) located on the reactor vessel drain line, LER 86-006-00 that reported RCPB leakage from a weld crack located at a reducing coupling connecting a reactor water level instrumentation line to the RPV nozzle N-16A, and LER 93-018-00 that reported RCPB leakage from a coupling weld (12-BC-15) located on the reactor vessel drain line.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Nozzle	NZL
Vessel, reactor	RPV
SYSTEMS	
Control rod drive system (CRD)	AA