Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
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September 10, 2007

Kevin H. Bronson
Site Vice President

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 2007-005-00

LETTER NUMBER: 2.07.074

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2007-005-00, "Reactor Scram resulting from Low Vacuum Turbine Trip," is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please contact Bryan Ford, (508) 830-8403, if there are questions regarding this submittal.

Sincerely,

Kevin H. Bronson

FXM/dl
Enclosure

cc: Mr. James Kim, Project Manager
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IE22

NRR

1. NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION	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.

LICENSEE EVENT REPORT (LER)

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TITLE (4)
Reactor Scram resulting from Low Vacuum Turbine Trip

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
07	10	2007	2007	005	00	09	10	2007	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	050

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)	X	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

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 July 10, 2007 at 1954 hours, an automatic scram occurred with the plant operating at approximately 48 percent reactor power. The automatic scram initiated from a valid Reactor Protection System (RPS) signal resulting from fast closure of the turbine steam control valves. The turbine steam control valves closed in response to an unexpected turbine trip on low vacuum. The low vacuum signal resulted from a calibration error on the low vacuum turbine trip mechanism (VTS-1). The turbine trip mechanism was calibrated and set incorrectly during the April/May 2007 Refueling Outage.

Corrective action taken included recalibrating the vacuum pressure trip mechanism. Additional corrective actions planned include training improvements relative to calibrating the turbine vacuum pressure alarm and trip setpoints and procedure enhancements.

The automatic scram occurred while the reactor mode selector switch was in the “RUN” position. Reactor vessel pressure was approximately 957 psig. The event occurred during a scheduled thermal backwash of the main condensers.

The event posed no threat to public health and safety.

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

BACKGROUND

Pilgrim Station periodically performs thermal backwashes of the sea water inlet sub-systems which feed the main condensers. A thermal backwash is performed for preventative maintenance purposes to remove debris and to control biofouling. The thermal backwash is accomplished by securing one of the two sea water pumps and aligning the sea water sub-systems for forward flow through one side of the condenser tube bundles and reverse flow through other side of condenser tube bundles. The sea water flowing in the reverse direction is heated above normal sea water temperature and as a result main condenser vacuum will decrease when a thermal backwash is performed.

The main condensers have Low Condenser Vacuum alarms (LVA-1 and LVA-2) set at 25.5" Hg. An automatic turbine trip on low vacuum (VTS-1) is normally set at 22.2" Hg. The turbine trip will result in closure of the turbine control and stop valves. Limit switches on the turbine stop valve and pressure switches on the turbine control valves supply input to the Reactor Protection System to scram the reactor when the turbine first stage pressure is above 108 psig. (~25% power).

The setpoint for the low vacuum turbine trip #1 (VTS-1) is verified, and calibrated, every two years during refueling outages (RFO). Prior to the turbine trip on July 10, 2007, the most recent surveillance on VTS-1 was performed during RFO-16 in April / May of 2007. The "as-found" trip setpoint was documented as 24.35" Hg. which was outside of the "No Adjust Limits" (21.95" to 22.45") and therefore required calibration. After the RFO calibration, the "as-left" setpoint was documented as 22.40" Hg. However, due to a calibration error the actual setting was 24.4" Hg.

On July 10, 2007 reactor power had been reduced to approximately 50% RTP to perform a thermal backwash and associated maintenance activities.

EVENT DESCRIPTION

On July 10, 2007, at 1954 hours, an unplanned automatic reactor protection system scram signal and scram occurred while operating at approximately 48% power. The event occurred during the performance of a main condenser thermal backwash evolution.

The scram signal was initiated by fast closure of the turbine steam control valves. The closing of the turbine steam stop valves also resulted in an automatic scram signal. The scram signal resulted in insertion of the control rods that were in a withdrawn position at the time of the event.

The event was initiated by a trip of the main turbine due to low condenser vacuum. Low vacuum actuated the improperly set low vacuum turbine trip assembly resulting in automatic closure of the turbine steam control valves and stop valves. The reactor scram occurred due to closure of the turbine control valves and turbine stop valves. The reactor scram was an expected reactor protection system design response to fast closure of the turbine control valves. Closure of the turbine control valves (and turbine stop valves) was an expected response resulting from actuation of the main condenser low vacuum turbine trip assembly. Main turbine trip on low vacuum during the thermal backwash was not expected. Post scram review of the "as-found" low vacuum turbine trip (VTS-1) calibration revealed that the trip set point was incorrectly set to actuate at 24.4" Hg.

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Closure of the turbine steam control valves and stop valves resulted in a reactor/steam system pressure increase. The turbine steam bypass valves responded as expected to control the initial pressure spike. The reactor vessel pressure increase and the insertion of the control rods each contributed to a decrease in the reactor water void fraction (shrink). The decrease in the void fraction resulted in a decrease in the reactor water level. The reactor water level decreased to about –10" (narrow range). The decrease in reactor water level, to less than the low water level setting of about +12" (narrow range), resulted in the expected automatic actuation of the Primary Containment Isolation Control System Group 2 (Sampling System) and Group 6 (Reactor Water Cleanup System) and Reactor Building Isolation Control System (RBIS).

The steam bypass valves responded to control reactor/main steam system pressure. The main steam relief valves did not open. Maximum reactor pressure observed during the event did not exceed 975 psig. After the initial scram, reactor pressure oscillations (10 psig swings) were observed with reactor pressure at 922 psig. Operator action was taken to control pressure with the bypass valve opening jack (BVOJ). The pressure oscillations stopped immediately and reactor pressure steadily decreased as expected. Post trip reviews determined that the reactor pressure oscillations were attributed to a degraded pilot valve in the mechanical pressure regulator.

The post trip reviews verified that other plant systems operated as expected in response to the turbine trip and reactor scram event.

The NRC Operations Center was notified of the event in accordance with 10 CFR 50.72 at 2255 hours on July 10, 2007.

CAUSE

The direct cause of the scram was reactor protection system actuation resulting from automatic closing of the turbine control valves. The closing of the turbine control valves (and stop valves) was the result of the turbine trip due to actuation of low vacuum turbine trip #1 (VTS-1).

The direct cause of the event was an improperly calibrated low vacuum turbine trip #1 setting during RFO 16. The root cause analysis identified that technicians failed to properly apply human performance tools and knowledge of this particular instrument when the instrument calibration was performed.

CORRECTIVE ACTION

Corrective action taken included recalibrating the low vacuum turbine trip #1 setting. The low vacuum turbine trip #1 setting was recalibrated and set to 22.5" Hg on July 11, 2007. Subsequently on July 12, 2007, the low vacuum turbine trip #1 setting was recalibrated consistent with the setpoint adjust limits previously established in plant design change PDC 02-047 to 20.0" Hg. An extent of condition review was performed and determined that other station vacuum instruments were properly calibrated.

Additional corrective actions planned include additional training relative to calibrating the turbine vacuum pressure alarm and trip setpoints and procedure enhancements. These actions are being tracked in accordance with the corrective action program.

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SAFETY CONSEQUENCES

The event posed no threat to public health and safety.

The maximum reactor power experienced during the event was below 50%.

Based on post trip reviews, plant safety systems responded as designed to the transient. The preferred and secondary sources of offsite power remained energized. The EDGs and Station Blackout Diesel Generator remained available during the event. The Core Standby Cooling Systems (HPCI System, Automatic Depressurization System, Residual Heat Removal System, Core Spray System) and the RCIC System remained available during the event to provide makeup water or core cooling if necessary.

The maximum reactor pressure that occurred during the event was 975 psig. This is within Technical Specification limits and well below relief valve and safety valve actuation setpoints. The relief valves and safety valves did not actuate during the event.

The minimum reactor water level that occurred was about –10" (narrow range). The level was above the low-low water level (about –46") for automatic actuation of the Core Standby Cooling Systems and automatic actuation of the Group I portion of the PCIS. The level was also above the level (about –127") corresponding to the top of the active fuel zone.

The turbine vacuum trip system is non-safety related. A turbine trip is a plant transient that the plant is designed to experience without safety consequence. A turbine trip at greater than 25% rated power results in a reactor scram.

No fuel, reactor or pressure boundary safety design limits were challenged by this event.

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REPORTABILITY

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

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station LERs since 1974. The review identified previous automatic reactor protection system scrams resulting from turbine trip, but did not identify any previous events that involved reactor protection system scrams that were caused by low vacuum and improper turbine vacuum pressure switch calibration.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Switch, Pressure	PS

SYSTEMS	CODES
Engineered Safety Features Actuation (RPS, PCIS, RBIS)	EA
Turbine Supervisory Control System	JJ