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

October 12, 2001
ENGC Ltr. 2.01.094

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) 2001-006-00, "Automatic Scram During Surveillance Test and Subsequent Reactor Water Level Anomalies," is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

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

Sincerely,

Mike Bellamy

Mike Bellamy

DWE/
Enclosure: LER 2001-006-00

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. Douglas Starkey, Project Manager
Office of Nuclear Reactor Regulation
Mail Stop: 0-8B-1
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

INPO Records

IE22

NRC Form 366 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 16

TITLE (4)
Automatic Scram During Surveillance Test and Subsequent Reactor Water Level Anomalies

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
08	13	01	2001	006	00	10	12	01	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 49

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)	x	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 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
x	EA	XFMR	R439	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 August 13, 2001, during a transient that began at 100 percent power, an automatic scram occurred at 49 percent reactor power. The scram included automatic insertion of the control rods. About 40 minutes after the scram, safety-related reactor water level instrumentation began to trend anomalously from actual water level and caused level instrumentation to be inoperable for approximately 36 minutes.

The root cause of the scram was concurrent trip of RPS channels "A" and "B". Channel "A" tripped due to a loss of Bus "A" power resulting from a surveillance procedure error. Channel "B" tripped due to high neutron flux resulting from a trip of both recirculation MG sets/pumps. The root cause of the anomalous water level indication was unique hydraulic conditions that allowed water to drain from reference legs "A" and "B".

Corrective action taken included the replacement of a fuse in the recirculation MG set field circuit, and isolating the reference leg backfill system. Corrective action planned includes revising the surveillance procedure, and evaluating options to preclude reference leg water draining. The events posed no threat to public health and safety.

BACKGROUND

On August 13, 2001, at 1108 hours, Emergency Diesel Generator (EDG) "A" was removed from service for a planned surveillance activity. When EDG "A" was removed from service all of the offsite and onsite power sources were available, and all of the 345 kV switchyard breakers were closed. The 4.16 kV auxiliary power distribution system (APDS), including nonsafety-related 4.16 kV Buses (A1, A2, A3, A4) and safety-related 4.16 kV Buses A5 and A6 and related AC electrical system and loads, were powered from the Main Generator via the Main Transformer and Unit Auxiliary Transformer. The control switches for a fast transfer of the 4.16 kV buses were in the ON position. A simplified one-line drawing of the offsite power sources, EDGs, and emergency service portion of the APDS and related electrical system is provided at the end of this report.

The EDG "A" maintenance activity consisted of a surveillance of EDG "A" to Bus A5 relays. The surveillance was being performed in accordance with Attachment 8A of PNPS Procedure 3.M.3-1 (Rev. 51), "A5/A6 Buses 4kV Protective Relay Calibration/Functional Test and Annunciator Verification." During the surveillance, 4.16 kV breaker 152-509 (EDG "A" supply breaker to Bus A5) is put in the TEST position and closed. This means that breaker 152-509 (A509) was not connected to Bus A5.

At 1400 hours, the reactor protection system (RPS) motor-generator (MG) set "A" was removed from service due to vibration. To remove the MG set from service, the source of power for RPS Bus "A"/Channel "A" was transferred from Bus A3 to safety-related 480/120-volt standby RPS transformer X20. This configuration meant that RPS Bus "A"/Channel "A" was powered from Bus A5 via 480-volt Bus B1, swing Bus B6, Bus B10, transformer X20, and in-series electrical protection assemblies EPA-5 and EPA-6. These assemblies (EPA-5 and EPA-6) provide a voltage protection function to electrical components that include the RPS Bus "A"/Channel "A" relays (or RPS Bus "A"/Channel "B" relays) when these components are powered from the standby RPS transformer X20. Nonsafety-related RPS Bus "B"/Channel "B" was powered from Bus A4 via Bus B4, Bus B22, RPS MG set "B", and EPA-3 and EPA-4.

The status of other systems and plant conditions were as follows.

- The reactor vessel pressure was approximately 1032 psig with the reactor water at the saturation temperature for that pressure. The reactor water level was approximately +29 inches.

- The nonsafety-related Condensate System, Feedwater System, and Recirculation System were operating with all pumps in service. The Recirculation System loops "A" and "B" pump controls were in the manual control mode.

- The safety-related Core Standby Cooling Systems - High Pressure Coolant Injection, Automatic Depressurization, Core Spray, and Residual Heat Removal - were operable and in standby service.

- The safety-related Reactor Core Isolation Cooling System was operable and in standby service.

- The safety-related Salt Service Water (SSW) System was operable with train "A" pumps and train "B" pumps in operation. The other SSW pumps were in standby service.

- The safety-related Reactor Building Closed Cooling Water (RBCCW) System was operable with loop "A" and "B" pumps in service. The other RBCCW pumps were in standby service.

- The nonsafety-related Turbine Building Closed Cooling Water (TBCCW) System loop "A" pump was in service and the loop "B" pump was in standby service.

At 1753 hours, a utility licensed operator at the EDG "A" control panel placed the EDG "A" test control switch in the NORMAL position in accordance with PNPS Procedure 3.M.3-1 (Rev.. 51) Attachment 8A step [6]. When the test control switch was placed in the NORMAL position, the EDG "A" operational mode automatically changed from DROOP to ISOCHRONOUS mode and caused the logic circuitry to sense that both Bus A5 supply breakers 152-505 (A505) and 152-509 (A509) were closed at the same time even though breaker 152-509 was not actually connected to Bus A5 (breaker 152-509 in the TEST position and closed in accordance with PNPS Procedure 3.M.3-1). Consequently, breaker 152-505 opened automatically as designed and caused Bus A5 to de-energize. The loss of power to Bus A5 resulted in the de-energizing of the related AC electrical system and components powered from Bus A5 including breakers that power the Control Rod Drive (CRD) train "A" pump, Residual Heat Removal (RHR) System train "A" pumps, Core Spray System train "A" pump, and Bus B1. Bus B1 electrical loads include safety-related 480-volt Buses B6, B15, B17 and respective loads, and nonsafety-related Bus B29 and TBCCW loop "A" pump.

- The loss of power on Bus B1 was sensed by the loss of voltage control circuitry of safety-related swing Bus B6, and resulted in automatic transfer of the source of power for Bus B6 from Bus B1 (no longer powered by Bus A5) to Bus B2 (powered by Bus A6). The transfer of Bus B6 resulted in a designed voltage interruption for approximately 2 - 3 seconds on Bus B6 and related electrical loads that include safety-related Buses B10 and B20. Bus B10 electrical loads include the standby RPS transformer X20 (powering RPS Bus "A"/Channel "A" relays via electrical protection assemblies EPA-5 and EPA-6), and safety-related 120-volt instrument Panel Y1.

- The loss of power on Bus B1 also resulted in the loss of power for safety-related 480-volt Buses B15, B17, and nonsafety-related 480-volt Bus B29 and TBCCW loop "A" pump (in operation).

- The loss of power on Bus B15 resulted in the loss of power to AC electrical components that include the RBCCW loop "A" pumps, SSW train "A" pumps, and Standby Gas Treatment System train "A" fan and heaters. The source of power to SSW swing pump "C" remained energized (via Bus B6).

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

- The loss of power on Bus B17 resulted in the de-energizing of electrical loads that include 480-volt Core Spray train "A" suction and injection valves, Standby Liquid Control System train "A" injection pump, Control Room High Efficiency Air Filtration System train "A" supply fan and heaters, RHR train "A" pumps' area coolers, RHR train "A" Suppression Pool (LPCI mode) suction valves, RHR train "A" Suppression Pool Cooling mode block valve, RHR train "A" Containment Spray injection valves, 480/120 "A" voltage regulating transformer X55, and 120-volt "A" safeguards Panels Y3/Y31.

- Safeguards Panels Y3 and Y31 electrical loads de-energized, including normally energized AC relays that are part of the logic circuitry of the Primary Containment Isolation Control System (PCIS) Channel "A" and Reactor Building Isolation control System (RBIS) Channel "A".

- The design of Panel Y1 features an automatic transfer from Bus B10 (powered from Bus B6) to Bus B15 (that was de-energized). Bus B10 is the normal source of power to Panel Y1. Panel Y1 did not automatically transfer from Bus B10 to Bus B15 because Bus B15 was de-energized (loss of power from Bus A5). Panel Y1 and related electrical loads re-energized when Bus B10 re-energized after the transfer of Bus B6 from Bus B1 to Bus B2, approximately 2 to 3 seconds after Bus A5 de-energized.

Meanwhile, Recirculation System MG sets/pumps "A" and "B" tripped almost simultaneously while at 100 percent reactor power.

EVENT DESCRIPTION

On August 13, 2001, at 1753 hours, an automatic RPS scram signal and scram occurred while at approximately 49 percent reactor power. The complete insertion of control rods was subsequently verified.

The following responses and conditions occurred:

- Automatic transfer of the source of power for Buses A1, A2, A3, A4, and Bus A6 from the Unit Auxiliary Transformer to the Startup Transformer. Bus A5 and related AC electrical system including Buses B1, B15, B17, and B29 remained de-energized. Bus B6 remained energized from Bus B2.

- Automatic trip of the Turbine-Generator and opening of the 345 kV circuit breakers that connect the Main Transformer to the switchyard.

- Safeguards "B" Panels Y4 and Y41 de-energized but did not automatically re-energize as designed because of the failure of a sub-component in the voltage regulating transformer (x56) that powers the panels . Panels Y4 and Y41 power electrical components that include normally energized relays that are part of PCIS Channel "B" and RBIS Channel "B". The de-energizing of Panel Y4 in conjunction with the previous de-energizing of Panel Y3 resulted in the de-energizing of normally energized AC relays in PCIS and RBIS Channels "A" and "B".

Meanwhile, the trip of the recirculation MG sets/pumps and scram resulted in a decrease in the void fraction in reactor coolant and consequent reactor water level decrease to approximately +4 inches. The decrease in water level to less than +12 inches resulted in an additional automatic RPS scram signal and automatic actuation of PCIS and RBIS as designed.

The additional RPS scram signal resulted in a control room alarm but no control rod movement because the control rods were already inserted.

The PCIS actuation resulted in the following designed responses:

- Automatic closing of the Primary Containment System (PCS) Group 2 isolation valves that were open.
- The PCS Group 3 isolation valves remained closed.
- Automatic closing of the PCS Group 6/Reactor Water Cleanup (RWCU) System isolation valves.

The other PCIS isolation valves (Groups 1, 4, 5, and 7) including the main steam isolation valves, remained open in accordance with design.

The RBIS actuation resulted in the following designed responses:

- Automatic start of Standby Gas Treatment System (SGTS) train "B". SGTS train "A" did not start because Bus B15 remained de-energized.
- Automatic closing of the Reactor Building ventilation system supply and exhaust dampers.

Utility licensed operator response was orderly and included the following. The process began for verifying the full insertion of the control rods. Initially, the reactor control panel did not indicate the position of two rods. The call-rod function of the plant computer indicated the position for two control rods were not fully inserted but almost immediately afterward the computer indicated that all control rods were fully inserted.

At 1754 hours, EOP-01,"RPV Control," was entered because reactor water level was less than +12 inches.

The TBCCW loop "B" pump was started at 1756 hours. This action restored TBCCW cooling water flow to nonsafety-related components.

At 1756 hours, operators began actions to cross-tie RBCCW loop "A" with RBCCW loop "B". This action was taken to restore cooling flow to components cooled by RBCCW loop "A", and was accomplished by 1805 hours.

The control rod drive (CRD) charging water header block valve HO-301-25 was closed at approximately 1756 hours. This action was taken to allow all control rods to settle and confirm a fully inserted position indication ("00") for all control rods on the reactor control panel.

At 1803 hours, the RBIS was manually actuated. This action was taken in accordance with PNPS Procedure 5.3.18, "Loss of 120V AC Safeguard Buses Y3 and Y31," and PNPS Procedure 5.3.19, "Loss of 120V AC Safeguard Buses Y4 and Y41." The actuation resulted in continued operation of SGTS train "B". SGTS train "A" did not start because Bus A5 and related AC electrical system that powers the SGTS train "A" fan and heaters remained de-energized.

Meanwhile, operators noted that the "B" safeguards Panels Y4 and Y41 were de-energized. The panels are powered from Bus A6 via Buses B2, B18, and voltage regulating transformer X56. The operating mode of transformer X56 was changed from the REGULATE mode to the BYPASS mode. This manual action was taken in accordance with PNPS Procedure 5.3.19 and resulted in the energizing of Panels Y4 and Y41 at 1804 hours.

At 1822 hours, EPA-5 and EPA-6 were reset. This manual action re-energized RPS Bus "A" and Channel "A" relays.

Breaker 152-509 was manually opened as part of actions to re-energize Bus A5 at 1825 hours. The opening of the breaker was made with the Bus A6 fast transfer switch in the OFF position. Opening breaker 152-509 in this configuration resulted in the automatic closing of breaker 152-501 as designed. Breaker 152-501 (A501) is the Bus A5 supply breaker from the Shutdown Transformer via Bus A8. The closing of breaker 152-501 re-energized Bus A5 and related AC electrical system that power 4.16 kV components (train "A" pumps) and Bus B1 that re-energized Buses B15, B17, and B29. The re-energizing of Bus B15 re-energized the AC electrical system that power loads including 480-volt train "A" components (including RBCCW and SSW pumps, SGTS fan and heaters). The re-energizing of Bus B17 re-energized the related AC electrical system that power components including train "A" pumps (including standby liquid control pump), train "A" motor operated valves (RHR suction and injection, Core Spray suction and injection), "A" voltage regulating transformer X55 and "A" safeguards Panels Y3 and Y31.

Also at 1825 hours, the RPS was reset. The RPS reset includes the automatic closing of the scram valve CV-301-126 in each of the 145 hydraulic control units (HCUs). The CRD charging water header block valve HO-301-25 was still closed and the reactor vessel pressure was approximately 725 psig at the time of the reset. All reactor vessel water level indications were tracking consistently at that time.

At approximately 1833 hours, the reactor vessel water level indications of instrumentation connected to the reference legs of Condensing Chambers 12A and 12B began to diverge from the water level indications of instrumentation connected to the other reference legs (Condensing Chambers 11, 13A and 13B). In response to the indications and training, the operators were directed to open valve HO-301-25. Located at the end of this report is a simplified drawing of the reactor vessel, Condensing Chambers 12A/B, reference legs and related backfill system, and reactor water level instrumentation.

The PCIS (Group 2) and RBIS circuitry were reset by 1842 hours.

At 1859 hours, valve HO-301-25 was opened. Within approximately two minutes of opening valve HO-301-25, the reactor water level indications of instrumentation connected to the reference legs of Condensing Chambers 12A and 12B began to converge with the water level indications of the reactor water level instrumentation connected to the reference legs of Condensing Chambers 11, 13A, and 13B. By approximately 1909 hours the reactor water level indications from instrumentation on all reference legs were consistent.

At 1932 hours, EOP-01 was exited.

The NRC Operations Center was notified of the scram in accordance with 10 CFR 50.72 at 2130 hours on August 13, 2001. A follow-up notification regarding the valid actuation of the PCIS and RBIS that occurred at the time of the scram was made to the NRC Operations Center in accordance with 10 CFR 50.72 at 1338 hours on August 14, 2001.

On August 13, 2001 at 2122 hours, 345 kV switchyard breakers 104 and 105 were closed. This action re-established the 345 kV switchyard ring bus.

The source of power to Bus A5 was transferred from the Shutdown Transformer to the Startup Transformer (SUT) and Bus A5 was powered from the SUT by 2233 hours on August 13, 2001.

CAUSE

The direct cause of the RPS scram signal that resulted in the scram was co-incident trip signals on RPS Channels "A" and "B".

- The RPS Channel "A" trip signal was the result of Bus A5 becoming de-energized while the RPS Bus "A"/Channel "A" was powered from Bus A5 via Buses B1, B6, B10, RPS standby transformer X20, EPA-5 and EPA-6. The loss of power resulted in the de-energizing of the coils of RPS Channel "A" relays. The RPS relays are normally energized and are designed to de-energize for the trip function.

- The RPS Channel "B" trip signal occurred due to a trip of the Neutron Monitoring System Channel "B" average power range monitor (APRM) 'D'. Immediately preceding the trip signal, both Recirculation System MG sets/pumps tripped while at 100 percent reactor power. A decrease in reactor core flow and reactor power is the expected consequence of the trip of the Recirculation System MG sets/pumps during reactor power operation. The APRMs responded as expected as reactor power decreased and approached the APRMs' flow biased setpoint. The APRM flow biased trip setpoints incorporate the Enhanced Option 1A stability long term solution. Post-trip review data revealed that APRM 'D' tripped (high neutron flux) and cleared in less than one-half second, approximately eight seconds after the trip of the Recirculation System MG sets/pumps. The trip functioned as intended and precluded reactor operation in the Enhanced Option 1A scram region.

The cause of the trip of the Recirculation System MG sets/pumps "A" and "B" is as follows.

- The MG set/pump "A" trip was caused by the loss of power to Bus A5 and related AC electrical system loads that included the recirculation MG set "A" lube oil pumps. The loss of power to the lube oil pumps resulted in the loss of lube oil pressure and consequent MG set "A" scoop tube lock-up and trip of MG set/pump "A".

- The MG set/pump "B" trip was caused by an MG set lockout that was due to a false MG set "B" low generator field voltage signal. The MG set voltage regulator utilizes 120-volt power from safety-related instrument Panel Y1 for the volts/hertz reference signal. This signal was lost momentarily and reapplied approximately 2 seconds later (Bus B6 transfer from B1 to B2) and caused the voltage regulator to lower generator field voltage. The lowering of the field voltage combined with a high resistance connection between the fuse and fuse holder associated with the relay resulted in a false low generator field voltage to be sensed by the protective relay scheme. The protective relay scheme caused a MG set "B" lockout which tripped the MG set drive motor breaker and MG set field breaker.

The direct cause of Bus A5 de-energizing was procedure error. The root cause was a utility non-licensed electrical maintenance engineer personnel error that introduced the error when PNPS Procedure 3.M.3-1 was revised. The procedure revision (to Rev.. 50) was made to separate the testing of Technical Specifications related relays and relays that are not related to Technical Specifications. The revision included the addition of two new attachments, 8A and 19A. Attachment 8A (19A) was added to test the EDG "A" breaker 152-509 relays (EDG "B" breaker 152-609 relays). The procedure error was introduced into both attachments and involved the restoration of the circuitry after relay testing. Specifically, the error consisted of not including a step to open breaker 152-509 (152-609) prior to restoring the circuitry to normal. Attachment 8A step [6] is part of the sequence of steps for restoring the circuitry to normal after the EDG "A" to Bus A5 relays are tested. Neither Attachment 8A nor Attachment 19A had been performed until the EDG "A" relay test on August 13, 2001.

The cause of the "A" safeguards panels not re-energizing automatically as designed was the de-energizing of Bus A5 and related AC electrical system that power the panels via voltage regulating transformer X55. The direct cause of the "B" safeguards Panels Y4 and Y41 not re-energizing automatically as designed after the Bus A6 transfer was the failure of a sub-component mounted on a portion of the tap control board in voltage regulating transformer X56. The failure was a new failure for the four voltage regulating transformers installed at Pilgrim Station, safety-related transformers X55, X56, X57, and X58. Transformer X58 experienced the same voltage transient on August 13, 2001, and X58 operated as designed. A review was conducted of previous problems with the Pilgrim Station regulating transformers and a similar search was conducted of the INPO Nuclear Plant Reliability Data System. The review and search concluded this was the first instance of a problem with a portion of the tap control board of a voltage regulating transformer. Transformer X56 was manufactured by Rapid Power Technologies Incorporated (model number PWTAB015120E).

The cause of the anomalous reactor water level indications were hydraulic conditions unique to this event.

Analysis indicates the water level changes sensed by the related reactor vessel water level instrumentation resulted from the draining of water from the reference legs connected to Condenser Chambers 12A/B. The draining of water was the result of reverse flow past the reference legs' backfill system check valves. The draining reduced the water inventory from the reference legs and caused the instrumentation (level transmitters) to sense a higher than actual reactor water level.

The backfill system for the reference legs connected to Condensing Chambers 12A/B was designed to be backfilled with water from the CRD System charging water piping. The backfill flow was interrupted when the CRD charging water block valve (HO-301-25) was closed to allow all CRDs to settle and achieve a fully inserted position indication on the reactor control panel after the scram. Later, the reset of RPS was performed when the reactor vessel pressure was approximately 725 psig. The 725 psig pressure was greater than the CRD System HCUs' accumulator pressure of approximately 570 psig. The reset of the RPS includes the automatic closing of the CRD scram valves which result in the HCUs' accumulators being available to accept water displaced from the reference legs. The resulting hydraulic conditions of reactor pressure greater than CRD charging water header pressure (downstream of valve HO-301-25), no backfill flow from a CRD pump (valve HO-301-25 closed), and reverse flow through the backfill system check valves, allowed the displacement of water from the reference legs to the HCUs' accumulators.

The displacement occurred past the backfill system check valves and was greater than would be normally experienced because the backfill system flow metering valves were at a maximum flow setting. The check valves did not actuate because the differential pressure across the check valves was not sufficient to cause the check valves to close. The check valves are designed to close at a differential pressure of six psi (psid) or less. After the event, calculations were performed and the check valves were tested. The calculations concluded the check valves experienced a differential pressure of approximately one and one-half psid for this event. The results of the as-found check valve testing determined the valves closed between two and six psid. Therefore, the check valves would not have been expected to close under the conditions experienced for this event.

The anomalous indications of the magnitude experienced were unique to this event. The magnitude of the deviation from actual reactor water level was approximately +26 inches on the "A" reference leg instrumentation that occurred at approximately 1837 hours, and approximately +11 inches on the "B" reference leg instrumentation that occurred at approximately 1853 hours.

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

CORRECTIVE ACTION

PNPS Procedure 3.M.3-1 Attachment 8A and Attachment 19A are being revised. The corrective action program (PR 01.9779) will track this action and any additional corrective actions.

The fuse for the Recirculation System MG Set "B" field voltage relay was replaced. The corrective action program will track any additional corrective actions (PR 01.9775).

The tap control board of voltage regulating transformer X56 was replaced and functionally tested with satisfactory results prior to startup following the event. The tap control board has been sent to a third party supplier for additional testing. The focus of this action is to determine the root cause of the failure. This action and any additional actions will be tracked in the corrective action program (PR 01.9776).

As an immediate action, the backfill system was isolated from both reactor vessel reference legs connected to Condensing Chambers 12A/B. An engineering evaluation provided the basis for operability of the related reactor vessel instrumentation with the backfill system isolated. Longer term corrective action(s) were being considered when this report was prepared. The focus of the corrective action(s) is to preclude the draining of water from the reference legs connected to Condensing Chambers 12A/B. This action(s) will be tracked in the corrective action program (PR 01.9774).

SAFETY CONSEQUENCES

The events posed no threat to public health and safety.

The following is a discussion concerning Bus A5 becoming de-energized.

The de-energizing of Bus A5 de-energized the related AC electrical system; essentially, safety-related train "A" pumps, valves, fans, heaters, and control circuitry. Except for safeguards Panels Y4 and Y41, Bus A6 and the related AC electrical system remained energized and was not affected. EDG "B" was operable, both sources of offsite power were energized and available, and the Station Blackout Diesel Generator was available. All of these sources are designed to power Bus A6 and related AC electrical system loads including safety-related train "B" pumps, valves, fans, and heaters

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

The following is a discussion concerning the RPS actuation that resulted in the scram:

The automatic actuation of the RPS was the result of the de-energizing of normally energized RPS Channel "A" relays and a high neutron flux condition sensed by the Neutron Monitoring System Channel "B" APRM. The RPS relays are designed to de-energize for the scram function. The high neutron flux condition was the result of the trip of the Recirculation System MG sets/pumps "A" and "B". The Neutron Monitoring System APRMs are designed to initiate a trip if a high neutron flux condition occurs. The RPS functioned as designed in response to the concurrent trip of RPS Channels "A" and "B".

The following is a discussion concerning the safeguards "A" Panels Y3 and Y31, and "B" Panels Y4 and Y41 that became de-energized.

Panels Y3 and Y31 were de-energized for 32 minutes (1754 hours to 1826 hours). Panels Y4 and Y41 were de-energized for 10 minutes (1754 hours to 1804 hours). Thus, both "A" and "B" safeguards panels were de-energized for approximately 10 minutes.

The significance of the effects of a simultaneous loss of power to the panels powered by regulating transformers X55 and X56 was assessed previously. The assessment concluded the loss of power to Panels Y3/Y31 and/or Panels Y4/Y41 is detectable, the actions to re-energize the panels are contained in approved procedures, immediate safety functions are not adversely affected, and the panels can be powered in sufficient time to support longer term safety functions.

The following is a discussion concerning the anomalous reactor vessel water level indications. The indications began approximately 40 minutes after the scram and ended approximately 36 minutes after the indications began.

The reactor water level indications from the other instrumentation not connected to Condensing Chambers 12A/B were monitored by the licensed operators and the operators knew the actual reactor vessel water level. The operators knew the actual water level because the instrumentation connected to the reference legs of Condensing Chambers 11, 13A, and 13B were not affected because these reference legs are not equipped with a reference leg backfill system. The backfill system is installed only on the reference legs connected to Condensing Chambers 12A/B and it was the draining of water from these reference legs that caused the anomalous reactor water level indications. The operators have been trained on anomalous water level indications and operations procedures include the identification of and response to anomalous reactor water level indication(s).

The reactor water level during the event was controlled automatically by the normal water level control system that receives reactor water level signals from reactor vessel water level instrumentation connected to the reference legs of Condensing Chamber 13A or 13B.

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

The RPS, ATWS, PCIS, and RBIS water level initiation functions were impacted and inoperable for approximately 36 minutes. The RPS low water level scram function and ATWS low-low water level initiation functions were not required when the indications were experienced because the control rods were already inserted and the mode switch was in the SHUTDOWN position.

The PCIS and RBIS actuation that occurred as a consequence of the time of the scram was initiated by the reactor water level instrumentation in accordance with the design setpoints. The water level anomaly resulted in the RBIS and portions of the PCIS (Groups 1, 2, 3, and 6) being inoperable for the automatic isolation function (water level) for 36 minutes while these system were required to be operable. The automatic high Drywell pressure initiation function for the RBIS and Group 2 and 3 portions of the PCIS was not affected and was operable. Although the other portions of the PCIS (Groups 1, 4, 5, 6, and 7) do not receive a high Drywell pressure isolation signal, all portions of the PCIS and RBIS can be manually initiated by the licensed operators if necessary.

The water level anomaly resulted in the Core Standby Cooling Systems (CSCS) - HPCI, ADS, RHR, and Core Spray -- being inoperable for the automatic initiation/control function (water level) for approximately 36 minutes while these systems were required to be operable. The automatic high Drywell pressure initiation function for these systems was not affected and was operable. Although not part of the CSCS, the RCIC System functions to provide high pressure core cooling similar to the HPCI System. The RCIC System is not initiated by a high Drywell pressure. The HPCI System is a backup to the RCIC System and the Automatic Depressurization System (ADS) is a backup to the HPCI System. The HPCI System and ADS are automatically initiated by a high Drywell pressure. All of these systems including the RCIC System can be manually initiated by the licensed operators if necessary.

The anomalous water level indications of the magnitude experienced during the event were the result of unique hydraulic conditions and not indicative of an accident response. As discussed previously, the draining of water from the reference legs of Condensing Chambers 12A and 12B was the result of the HCUs' accumulators being available to accept the water displaced from the subject reference legs. The accumulators would only be available to accept water from the reference legs if the following conditions existed: the CRD pump(s) is not in service or the CRD charging water block valve HO-301-25 is closed, an RPS scram has occurred, reactor pressure is greater than the accumulators' gas pressure (approximately 570 psig), and a reset of the RPS is initiated. Without these conditions the draining of water from the reference legs would not occur or, it is believed, the draining would be minor and would not significantly impact the level instrumentation and designed response of the related systems. The necessity of a reset of RPS prevents this condition from significantly impacting the designed response during an accident because the RPS reset is a manual action which would not be performed in the first few minutes of an accident when systems' actuation would be occurring.

REPORTABILITY

This report was submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A) because the RPS scram signal, although a designed response to the trip of RPS Channel "A" due to the de-energizing of Channel "A" relays and trip of RPS Channel "B" due to neutron flux conditions resulting from the trip of the recirculation MG sets/pumps, was not planned.

This report was also submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A) because the PCIS and RBIS actuation, although a designed response to a low reactor vessel water level resulting from the trip of the recirculation MG sets/pumps and scram, was not planned.

This report was also submitted in accordance with 10 CFR 50.73(a)(2)(vii) subparts (B), (C), and (D) because the reactor vessel water level instrumentation connected to the reference legs of Condensing Chambers 12A and 12B were inoperable for the high water level, low water level, and low-low water level initiation/control/trip of the systems that function to remove residual heat, or control the release of radioactive material, or mitigate the consequences of an accident. The reactor water level instrumentation was inoperable for approximately 36 minutes when those systems that receive reactor water level signals were required to be operable.

This report was also submitted in accordance with 10 CFR 50.73(a)(2)(vii)(B) because the de-energizing of safeguards "A" panels and safeguards "B" panels affected the operability of more than one train in more than one system.

SIMILARITY TO PREVIOUS EVENTS

A review for similarity was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since 1997. The review focused on LERs involving PNPS Procedure 3.M.3-1 or similar reactor water level anomaly. The review identified no similar events.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

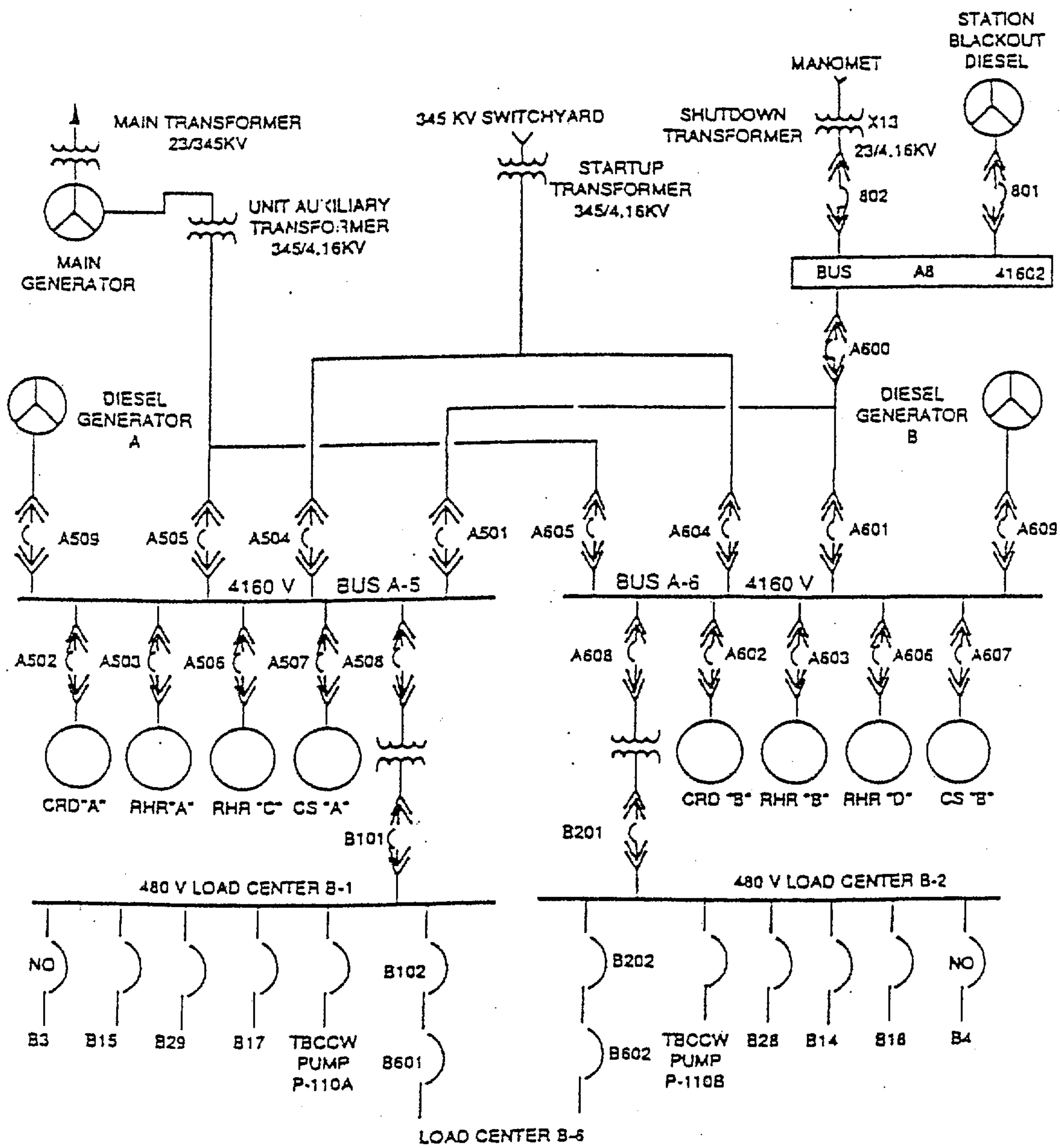
COMPONENTS	CODES
Bus (A5/A6, B1/B2, B6, B10/B20)	BU
Circuit breaker, AC	52
Condensing unit (Chamber 12A/B)	CDU
Fuse	FU
Generator (EDG)	DG
Hydraulic control unit	HCU
Indicator, level	LI
Panel (Y3/Y31, Y4/Y41, C-905, Y1)	PL
Piping specialties (reference leg)	PSX
Pump	P
Relay	RLY
Relay, tripping	94
Switchgear (152-509, 152-505)	SWGR
Transformer (X55, X56)	XMFR
Transmitter, level	LT
Valve (HO-301-25)	V
Valve, electrically operated (CV-301-126)	92
Valve, injection	INV
Valve, isolation	ISV
Vessel, reactor	RPV

SYSTEMS	
Component/closed cooling water system (RBCCW, TBCCW)	CC
Containment isolation control system (PCIS/RBIS)	JM
Control rod drive system	AA
Emergency on-site power supply system (EDG)	EK
Engineered safety features actuation system (RPS/PCIS/RBIS)	JE
Essential service water system (SSW)	BI
High pressure coolant injection system	BJ
Incore monitoring system (neutron monitoring system)	IG
Low pressure core spray system	BM
Low-voltage power system (480-volt) - Class 1E	ED
Medium-voltage power system (4.16 Kv) - Class 1E	EB
Plant protection system (RPS)	JC
Reactor building	NG
Reactor recirculation system	AD
Reactor core isolation cooling system	BN
Residual heat removal system	BO
Standby liquid control system	BR
Standby gas treatment system (SGTS)	BH

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	15 of 16
PILGRIM NUCLEAR POWER STATION	05000-293	2001	006	00	

SIMPLIFIED DRAWING of
PREFERRED and SECONDARY OFFSITE POWER SOURCES
and
EMERGENCY SERVICE PORTION of the
AUXILARY POWER DISTRIBUTION SYSTEM



EMERGENCY AC DISTRIBUTION

Simplified Drawing of Reference Leg Backfill System
(Condensing Chambers 12A & 12B)