NRC Form 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 94/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE(3)
PILGRIM NUCLEAR POWER STATION	05000-293	1 of 11

TITLE (4)
Recirculation System Loop 'B' Pump Trip due to Surveillance Procedure Inadequacy

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
11	05	97	97	016	00	12	04	97	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)							
POWER LEVEL (10)	100		20.2201 (b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
			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)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Douglas W. Ellis - Principal Regulatory Affairs Engineer	508-830-8160

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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 November 5, 1997, at 1810 hours, an unplanned closing of the recirculation system loop `B' pump discharge valve occurred during the performance of a surveillance of the emergency diesel generator (EDG) `B' initiation circuitry. The closing of the valve resulted in the automatic trip of the recirculation system loop `B' motor-generator (MG) set and pump, consequent decrease in reactor core flow and reactor power, and reactor operation in the caution zone of the reactor core power - flow map. Licensed operator response included the insertion of control rods to achieve reactor operation at less than the 100 percent load line. The root cause of the event was an inadequacy in the test procedure being performed in that the procedure did not include guidance regarding how long the logic reset switch was to be depressed for a reset of the logic circuitry. The EDG `B' surveillance procedure, being performed at the time of the event, will be revised to include the status of additional relays prior to resetting the circuitry. All logic system functional test procedures will be reviewed for similar improvement.

The event occurred during power operation while at 100 percent reactor power with the reactor mode selector switch in the RUN position. The reactor vessel pressure was approximately 1035 psig with the reactor vessel water at the saturation temperature for that pressure. The event posed no threat to public health and safety.

BACKGROUND

The core standby cooling subsystems consist of the high pressure coolant injection system, automatic depressurization system, residual heat removal (RHR) system low pressure coolant injection (LPCI) mode, and the core spray system. Low pressure core cooling is provided by the RHR/LPCI mode and core spray system with the emergency diesel generators (EDGs) providing emergency ac power in the event of a design basis accident (LOCA with loss of off-site power).

Technical Specification 3/4.2.B pertains to the instrumentation that initiates or controls the core and containment cooling systems. Table 3.2.B specifies the minimum number of operable instrument channels that initiate the EDGs and includes low reactor water level (two channels), high drywell pressure (two channels), startup transformer loss of voltage (two channels), and startup transformer degraded voltage (two channels). For the EDGs, Table 4.2.B requires logic system functional testing once per operating cycle. The testing is performed using procedures 8.M.2-2.10.8.1, "Diesel Generator `A' Initiation by RHR Logic," and 8.M.2-2.10.8.2, "Diesel Generator `B' Initiation by RHR Logic." The tests are scheduled and tracked by the master surveillance tracking program.

During the performance of procedure 8.M.2-2.10.8.2, EDG `B' receives separate initiating signals from the RHR channel `B' relays 10A-K10B (high drywell pressure signal) and 10A-K92B (reactor vessel low water level). The signal is initiated by the manual actuation of the respective relay. Procedure 8.M.2-2.10.8.1 is similar to procedure 8.M.2-2.10.8.2 except that EDG `A' receives initiating signals from the RHR channel `A' relays 10A-K10A and 10A-K92A.

The core spray channel `A' control switches and relays that are part of the initiating circuitry to EDG `A', the circuitry of load shedding channels `A' and `B', and the automatic depressurization system are not affected by procedure 8.M.2-2.10.8.2.

On November 5, 1997, at about 1635 hours, multiple 24 hour LCOs (A97-358, A97-359, and A97-360) were entered for the planned performance of procedure 8.M.2-2.10.8.2. The LCOs were entered because of Technical Specifications 3.5.A that includes the RHR/LPCI system, 3.5.B that pertains to containment cooling that includes RHR/LPCI pumps, and 3.5.F that pertains to low pressure core cooling (includes RHR/LPCI) and the EDGs.

A pre-evolution briefing was held in accordance with procedure 1.3.34, "Conduct of Operations," and the performance of procedure 8.M.2-2.10.8.2 (Attachment 1) began. After EDG `B' was satisfactorily started at steps [9](p)(2) and (11) and shut down at steps [9](p)(4) and (13), the RHR/LPCI circuitry and affected components were being returned to normal at step [10]. After the control circuit of the EDG `B' output breaker 152-609 was re-powered at Step [10](e), the RHR logic circuitry was being reset at step [10](f).

Conditions existing just prior to the event were as follows:

- the reactor was operating at 100 percent with the reactor mode selector switch in the RUN position. The reactor vessel pressure was approximately 1035 psig with the reactor water at the saturation temperature for that pressure.

- the recirculation motor-generator sets/pumps `A' and `B' were operating at approximately 81 percent speed, and reactor core flow was approximately 56 E+06 pounds per hour.

- the configuration of the RHR loops `A' and `B' included the following. The LPCI loops `A'/'B' injection valves MO-1001-28A/B were open with the in-series valve MO-1001-29A/B closed. The RHR/LPCI loops `A'/'B' suction valves MO-1001-7A/B/C/D were open. The RHR shut down cooling suction valves MO-1001-43A/B/C/D were closed. The RHR loops `A'/'B' heat exchangers' bypass valve MO-1001-16A/B were open. The RHR loops `A'/'B' pumps were not in service. The loops `A'/'B' pumps' minimum flow valves MO-1001-18A/B were open.

- with a noted exception, the reactor building closed cooling water system and salt service water (SSW) system pumps were operating or in standby service. The SSW pump `A' was tagged out of service for maintenance.

- EDG `A' and EDG `A' load shed circuitry were in standby service.

- the automatic depressurization system channel/train `A', the high pressure coolant injection system, and reactor core isolation cooling system were in standby service.

- the core spray system channels/trains `A' and `B' were in standby service.

- the off-site 345 Kv transmission system lines 342 and 355 were energized, the 345 Kv switchyard air type circuit breakers were closed, and the startup transformer was in standby service.

- the 4.16 Kv auxiliary power system was energized from the main generator via the main transformer and unit auxiliary transformer. The offsite 23 Kv distribution system was energized with the 4.16 Kv bus A8, the shutdown transformer, and the station blackout diesel generator in standby service.

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

EVENT DESCRIPTION

On November 5, 1997, at approximately 1810 hours, an unplanned closing of the recirculation system loop `B' pump discharge valve MO-202-5B occurred during the performance of a logic system functional test procedure, 8.M.2-2.10.8.2 (rev. 15). The event occurred when the RHR/LPCI circuitry was being reset at step [10](f) of Attachment 1 of the procedure.

The closing of the valve (i.e., the actuation of the closing coil of the valve's operator) resulted in the automatic trip of the recirculation system loop `B' motor-generator (MG) set and pump, consequent decrease in reactor core flow and reactor power, from 100 percent to approximately 70 percent, and reactor operation in the caution zone of the reactor core power - flow map. A 24 hour LCO (A97-361) was entered in accordance with the facility operating license condition 3.E that pertains to single recirculation system loop operation.

Utility licensed operator response included the entry into procedure 2.4.17 (rev. 20), "Recirculation Pump(s) Trip," and the insertion of control rods to achieve reactor operation at less than the 100 percent load line.

Meanwhile, actions were initiated to determine the cause of the closing of valve MO-202-5B.

A recirculation flow converter failure alarm occurred at about the same time that the recirculation MG set/pump `B' trip occurred. The alarm was the result of a slight mismatch between the `A' & `B' flow converters and the reset setpoint of the `B' flow comparator. The I&C department was notified of the problem with the `B' flow comparator, and PR 97.9674 was written to document the problem with the `B' flow comparator. LCOs were entered in accordance with Technical Specifications 3.1.1 (reactor protection system) and 3.2.C (control rod block system). The LCOs (A97-362, A97-363, A97-364, A97-365, A97-366) were entered as a result of the problem with the `B' flow comparator, for the replacement of the flow comparator, and adjustment of the neutron monitoring system APRMs gain adjustment factor.

The `A' APRM channels (`A', `C', and `E') were momentarily but not significantly affected by the `B' flow comparator trip.

The trip of the recirculation MG set/pump `B' disturbed the equilibrium of the neutron flux in the reactor core, and was sensed by the neutron monitoring system local power range monitor (LPRMs). The LPRMs are spatially located throughout the reactor core and provide signals to the APRMs.

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At 1810 hours, APRM channel `A' operation, described as spiking, above the flow bias rod block and scram setpoints was observed. Subsequent investigation revealed the observation was due to a low reference signal from the APRM `A' flow reference trip card. The low reference signal affected the flow biased scram and rod block setpoints of APRM `A'. APRM `A' was bypassed by moving its mode switch from the OPERATE to the BYPASS position, and an LCO (T97-286) was entered in accordance with Technical Specification 3.1.1/Table 3.1.1. No similar operation (i.e., spiking) was noted from APRMs `B', `C', `D', `E', and `F'. The significance of the noted operation of APRM `A' was that it was the only APRM equipped with a new flow reference trip circuit board. The new circuit board is part of Pilgrim Station actions being taken for recirculation/reactor core thermal-hydraulic stability (BWROG option 1A).

Reactor operation at less than 100 percent load line was achieved by 1843 hours with reactor power at approximately 64.5 percent and core flow at approximately 43.85 E+06 pounds per hour.

By 1845 hours, the initial investigation of the cause of the closing of valve MO-202-5B had been completed and the RHR/LPCI logic circuit was reset. After the reset, valve MO-202-5B was opened in accordance with procedure 2.2.84 (rev. 55), "Reactor Recirculation System." Procedure 2.4.17 was subsequently terminated at 1917 hours.

Reactor operation in the caution zone of the reactor core power - flow map was re-entered, and the reactor control operator was briefed regarding reactor operation in the caution zone. A further reactor core power decrease was initiated by decreasing the recirculation loop `A' MG set/pump speed to 35 percent and was completed by 1930 hours.

At 1940 hours, an LCO was entered in accordance with Technical Specification 3.1.1/Table 3.1.1. The LCO (A97-363) was entered because the decrease in core flow resulting from the decrease in recirculation loop `A' flow affected the APRM gain adjustment factor (AGAF) of APRMs `A', `C', and `E. The AGAF of APRMs is adjusted via procedure 9.1, "APRM Calibration."

At 1953 hours, the reactor feedwater pump `C' was stopped in accordance with procedure 2.2.96 (rev. 47), "Condensate and Feedwater System," and 2.1.14 (rev. 28), "Station Power Changes."

By 2016 hours, reactor core flow at approximately 33.5 E+06 pounds per hour and reactor power at approximately 45 percent, thereby, exiting reactor operation in the caution zone.

The main turbine speed-load changer was decreased to approximately 67 percent in accordance with procedure 2.1.14 at 2020 hours.

At 2030 hours, the LCO (T97-286) entered for APRM `A' was terminated, and the LCO (A97-363) entered for adjusting the gain factor of APRMs `A', `C' and `E' was terminated at 2050 hours after the AGAF adjustments were completed.

At 2107 hours, the recirculation loop `B' MG set/pump lockout, that occurred as a consequent of the closing of valve MO-202-5B, was reset.

At 2200 hours, APRM `A' was bypassed and an LCO (T97-287) was entered. This action was taken because reactor manual control system rod block signals were experienced from APRM `A'.

At 2205 hours, the recirculation loop `A' MG set/pump speed was decreased to less than 35 percent in accordance with procedure 2.2.84 (rev. 55), "Reactor Recirculation System." These actions were taken as part of preparations for re-starting the recirculation loop `B' MG set/pump. The MG set/pump `B' was started at 2225 hours, and the speed was increased to approximately 35 percent, the speed of the MG set/pump `A'. The 24 hour LCO (A97-361) entered for operating with a single recirculation loop was terminated at 2225 hours.

At 2344 hours, the LCOs (A97-358, A97-359, and A97-360) entered for the performance of procedure 8.M.2-2.10.8.2 were terminated after the satisfactory completion of the procedure.

On November 6, 1997, at 0344 hours, the LCO (T97-287) entered for APRM `A' was terminated when it was returned to the OPERATE mode.

By 0810 hours, the LCOs (A97-362, A97-365, and A97-366) entered for the `B' flow comparator problem, replacement of the flow comparator, and calibration of the related APRMs (`B', `D', `F') were terminated.

At 0915 hours, the withdrawal of control rods began at approximately 46 percent reactor power.

At 0920 hours, the main turbine bypass valve BPV-1 opened slightly in response to the turbine speed load changer setting (at 67 percent) with the turbine load at 48 percent. The speed load changer setting was increased to 77 percent in accordance with procedure 2.1.14. PR 97.9677 was written to document the problem with the bypass valve.

At 1210 hours, the reactor feedpump `C' was started.

At 1217 hours, APRM `A' operation, described as spiking, at the "high" setpoint was experienced with reactor core flow at approximately 38 E+06 pounds per hour and reactor power at 66 percent. Reactor core flow was increased to approximately 39 E+06 pounds per hours to maneuver away from the APRM `A' high setpoint.

At 1400 hours, APRM `A' was bypassed because its conservative AGAF adjustment and reactor core flow-power conditions, approximately 38.8 E+06 pounds per hour flow and 76.5 percent power, were causing a rod block condition from APRM `A'. An LCO (T97-289) was entered because the APRM was bypassed, and PR 97.9679 was written to document the rod blocks that occurred. By 1414 hours, APRM `A' was returned to operating status with reactor core flow at approximately 39.8 E+06 pounds per hour and reactor power at 78 percent, and the LCO (T97-289) was terminated.

Beginning at 1718 hours, adjustments of the gain factor for APRMs `A', `B', `C', `D', `E', and `F' were made individually. The adjustments were made to each APRM with the applicable APRM bypassed. The adjustments were completed by 1805 hours.

At 1847 hours, an LCO (T97-290) was entered to perform a functional test of APRM `A'. The functional test was performed in accordance with procedure 8.M.1-3 (rev. 33), "APRM Functional." The functional test was completed with satisfactory results by 2046 hours, and the LCO (T97-290) was later terminated at 2210 hours.

At 2155 hours, 100 percent reactor power was achieved.

Problem Report 97.9673 was written to document the unplanned closing of valve MO-202-5B and consequent responses. The NRC Operations Center was notified in accordance with 10 CFR 50.72 at 2105 hours on November 5, 1997.

A critique of the event was held on November 6, 1997. The critique was conducted in accordance with procedure 1.3.63 (rev. 11), "Conduct of Critiques and Investigations." The critique was attended by applicable personnel including I&C technicians personnel who were performing the test and investigated the cause of the event, and licensed operators on-shift at the time of the event.

CAUSE

The root cause of event was an inadequacy in the test procedure, 8.M.2-2.10.8.2. The procedure did not provide guidance to the utility licensed operator, who reset the logic at step [9](p)(14), regarding how long to depress the RHR/LPCI loop `B' reset switch (10A-S1B). The reset switch was not depressed for a sufficient period of time for some of the RHR channel `B' circuitry, that includes some time delay relays, to reset. The reset switch energizes the normally de-energized reset relay 10A-K11B that, when energized for a sufficient period of time, functions to de-energize relays that become energized during the test.

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The reset switch was depressed and relays 10A-K9B and 10A-K10B (RHR channel `B' high drywell and/or low reactor water level relays) were verified de-energized in accordance with procedure step [9](p)(14). The procedure step, however, did not include a verification for the status of other relays including those in the RHR/LPCI pipe break detection and loop selection circuitry.

At step [10](f), the RHR logic test reset switch (10A-S46) was positioned to the RESET position. This action caused the RHR channel `B' logic test relay 10A-K104B to de-energize, and because of the insufficient period of time the reset switch (10A-S1B) was depressed at step [9](p)(14), relay 10A-K42B energized. Valve MO-202-5B is designed to close if relay 10A-K42B energizes. The valve's closing coil actuated to close the valve as designed and resulted in the event.

Procedure 8.M.2-2.10.8.2 has been completed previously with satisfactory results. An LCO was entered on November 14, 1997, at 0940 hours, to troubleshoot the RHR/LPCI reset switch (10A-S1B). The troubleshooting (via MR19702761) was completed with satisfactory results, and the LCO (A97-373) was terminated at 1125 hours. The procedure inadequacy was determined to be the root cause of the event.

CORRECTIVE ACTION

The EDG `A' initiation procedure, 8.M.2-2.10.8.1, was revised (to rev. 15), effective November 13, 1997. The revision was made to incorporate lessons learned as a result the critique stemming from the event on November 5, 1997. The revision included the addition of steps to verify the status of additional relays prior to resetting the logic circuitry at the completion of the test. Procedure 8.M.2-2.10.8.1 (rev. 15) was performed with satisfactory results on November 14, 1997.

The EDG `B' initiation procedure, 8.M.2-2.10.8.2, will be changed similar to the revision made to procedure 8.M.2-2.10.8.1, before it is next performed.

All logic system functional test procedures will be reviewed for similar improvement. This action is being tracked by the integrated action database (PR 97.9673).

The routine licensed operator training program includes events or conditions reported in Pilgrim Station Licensee Event Reports, and this report will be included in the program.

SAFETY SIGNIFICANCE

The event posed no threat to public health and safety.

There were no component or system failures that caused the event.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the closing of valve MO-202-5B and consequent, resultant responses, although in accordance with design for the RHR/LPCI function, was not planned.

SIMILARITY TO PREVIOUS EVENTS

A review for similarity was conducted of Pilgrim Station LERs submitted since January 1984. The review focused on events involving an unplanned actuation of the RHR/LPCI circuitry. The review identified previous events involving the RHR/LPCI circuitry that were reported in LERs 89-012-00, 89-017-00, 89-027-00, and 92-011-00.

For LER 89-012-00, an unplanned actuation of a portion of the RHR/LPCI logic circuitry occurred during cold shutdown conditions on March 9, 1989, at 1220 hours. The event occurred during the performance of a temporary procedure, TP 88-78 (rev. 1), used to test type CR 2820 time delay relays. The event resulted in no injection of water into the reactor vessel. The cause of the event was utility non-licensed I&C technician error in that the technician failed to properly insulate (i.e., boot) a normally closed pair of contacts of a type HFA relay (10A-K37A). Corrective action taken included an I&C workshop discussion on the proper installation of insulating boots on relay contacts.

For LER 89-017-00, unplanned actuations of the RHR/LPCI circuitry occurred during cold shutdown conditions on May 20, 1989, at 2215 hours, and on May 21, 1989, at 2025 hours. The events occurred during the performance of a surveillance procedure, 3.M.3-37, used to functionally test the control circuitry of the recirculation system loops `A' and `B'' motor generator sets' drive motor circuit breakers and lockout relays. Neither event resulted in the injection of water into the reactor vessel. The cause was utility non-licensed electrical maintenance engineer error in that on both occasions, the same engineer ineffectively blocked open a pair of contacts on relay 10A-Y105B. Factors contributing to the error(s) were the type and location of the relay. Corrective action taken included satisfactory troubleshooting of the related circuitry, and review and revision of test procedures that involve the blocking of contacts of type HGA relays installed in safety-related applications.

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For LER 89-027-00, an unplanned actuation of a portion of the RHR/LPCI logic circuitry occurred during cold shutdown conditions on September 5, 1989, at 1805 hours. The actuation occurred during the performance of the temporary procedure, TP 88-78 (rev. 3), used to test type CR2820 time delay relays. The event resulted in no injection of water into the reactor vessel. The cause was utility non-licensed I&C technician error in that the technician jumpered the termination points for contacts of the incorrect relay for the test. Termination points for a pair of contacts of RHR relay 10A-K14A were jumpered instead of the termination points for a pair of contacts of a Core Spray relay (14A-K14A). A contributing factor may have been internal lighting within the panels. Corrective action taken included the revision of the procedure (TP 88-78) to include verification for those procedure steps involving the installation of a jumper or insulating boot.

For LER 92-011-00, an unplanned actuation of a portion of the RHR/LPCI logic circuitry occurred during operation while at 100 percent reactor power. The event occurred during the performance of a surveillance procedure, 8.M.2-2.10.4-2, used to test the high pressure coolant injection (HPCI) system high water level circuitry. The event resulted in the automatic closing of the recirculation loop `A' pump discharge valve MO-202-5A, trip of recirculation loop `A' MG set/pump `A', automatic start of the RHR loop `B' pumps `B' and `D', and automatic start of EDG `B'. The event resulted in no injection of water into the reactor vessel. The cause was utility non-licensed I&C technician error in that the technician mistakenly actuated an RHR channel `B' high drywell pressure relay (10A-K10B) instead of the RHR channel `B' low reactor water level relay 10A-K8B, both type HFA relays. A factor contributing to the event was that the procedure, although including double verification for steps including the lifting/relanding of electrical jumpers, did not include a double verification for the relay to be actuated.

Corrective action taken included technician counseling. Preventive action taken included discussion of the event at an I&C workshop meeting, revision of the procedure that included the addition of a double verification to the procedure steps that require the manual actuation of a relay, and an assessment of other surveillance procedures for improvement.

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ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS Codes for this report are as follows:

COMPONENTS	CODES
Control rod	ROD
Generator, diesel (EDG)	DG
Generator set, motor (MG set)	MG
Relay, time-delay starting or closing	2
Valve, motor operated (MO-202-5B)	MO

SYSTEMS	
Control rod drive system	AA
Emergency on-site power supply system (EDG)	EK
Reactor recirculation system	AD
Residual heat removal system/LPCI	BO

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For LER 89-027-00, an unplanned actuation of a portion of the RHR/LPCI logic circuitry occurred during cold shutdown conditions on September 5, 1989, at 1805 hours. The actuation occurred during the performance of the temporary procedure, TP 88-78 (rev. 3), used to test type CR2820 time delay relays. The event resulted in no injection of water into the reactor vessel. The cause was utility non-licensed I&C technician error in that the technician jumpered the termination points for contacts of the incorrect relay for the test. Termination points for a pair of contacts of RHR relay 10A-K14A were jumpered instead of the termination points for a pair of contacts of a Core Spray relay (14A-K14A). A contributing factor may have been internal lighting within the panels. Corrective action taken included the revision of the procedure (TP 88-78) to include verification for those procedure steps involving the installation of a jumper or insulating boot.

For LER 92-011-00, an unplanned actuation of a portion of the RHR/LPCI logic circuitry occurred during operation while at 100 percent reactor power. The event occurred during the performance of a surveillance procedure, 8.M.2-2.10.4-2, used to test the high pressure coolant injection (HPCI) system high water level circuitry. The event resulted in the automatic closing of the recirculation loop `A' pump discharge valve MO-202-5A, trip of recirculation loop `A' MG set/pump `A', automatic start of the RHR loop `B' pumps `B' and `D', and automatic start of EDG `B'. The event resulted in no injection of water into the reactor vessel. The cause was utility non-licensed I&C technician error in that the technician mistakenly actuated an RHR channel `B' high drywell pressure relay (10A-K10B) instead of the RHR channel `B' low reactor water level relay 10A-K8B, both type HFA relays. A factor contributing to the event was that the procedure, although including double verification for steps including the lifting/relanding of electrical jumpers, did not include a double verification for the relay to be actuated.

Corrective action taken included technician counseling. Preventive action taken included discussion of the event at an I&C workshop meeting, revision of the procedure that included the addition of a double verification to the procedure steps that require the manual actuation of a relay, and an assessment of other surveillance procedures for improvement.

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ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS Codes for this report are as follows:

COMPONENTS	CODES
Control rod	ROD
Generator, diesel (EDG)	DG
Generator set, motor (MG set)	MG
Relay, time-delay starting or closing	2
Valve, motor operated (MO-202-5B)	MO

SYSTEMS	
Control rod drive system	AA
Emergency on-site power supply system (EDG)	EK
Reactor recirculation system	AD
Residual heat removal system/LPCI	BO