Boston Edison

Pilgrim Nuclear Power Station
Rocky Hill Road
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10 CFR 50.73

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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 95-008-00, "Primary Containment System Isolation Valve Unable to Close Fully on Automatic Signal due to Wiring Discrepancy", is submitted in accordance with 10 CFR Part 50.73.

In this letter, the following commitments are made:

- Training for maintenance and engineering personnel
- Engineering review of design and testing of similar valve configurations
- Engineering review for common causes of previous wiring discrepancies

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

E.T. Boulette

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JPC/laa/9500800

cc: Mr. Thomas T. Martin
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Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution

NRC Form 366 (5-92)

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 8

TITLE (4)
Primary Containment System Isolation Valve Unable to Close Fully on Automatic Signal due to Wiring Discrepancy

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	14	95	95	008	00	09	13	95	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		20.402(b)		20.45(c)		50.73(a)(2)(iv)		73.71(b)
			20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v))		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)(A)		(specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeffrey P. Calfa - Senior Compliance Engineer	508-830-8108

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
A	BD	20	L200	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 14, 1995, at approximately 1300 hours engineering personnel discovered a discrepancy between the field wiring and the drawings for Primary Containment System/Reactor Water Cleanup System isolation valve MO-1201-80. The 'as found' condition would not allow MO-1201-80 to automatically travel to the fully closed position in the event of an isolation signal. The discrepancy did not affect the full closing of the valve via its manual control switch. The discrepancy was discovered while investigating a Quality Assurance finding related to wiring discrepancies.

The root cause was contractor electrician personnel error during modification installation work performed in 1993. The isolation valve was immediately closed via its manual control switch and deactivated. The wiring discrepancy was corrected and the valve was satisfactorily tested on August 15, 1995, at 0309 hours. Corrective action planned includes training for maintenance personnel on the specific event, an engineering review of past wiring discrepancies to determine potential common causes, and field verifications of potentially impacted components identified as a result of the review.

The discovery occurred with the plant operating at 100 percent power with the reactor mode selector switch in the RUN position. The Reactor Vessel pressure was 1030 psig with the Reactor Vessel water at saturation temperature for the reactor pressure. This event posed no threat to public health and safety.

BACKGROUND

The objectives of the Reactor Water Cleanup (RWCU) System are to maintain high reactor water purity, remove corrosion products from the Reactor Vessel water, and provide a method for decreasing Reactor Vessel water inventory during plant heat-up. The Primary Containment System (PCS) Group 6/RWCU isolation valves are: MO-1201-2, the RWCU pump suction inboard isolation valve; MO-1201-5, the RWCU pump suction outboard isolation valve; and MO-1201-80, the RWCU System Return Isolation Valve. The PCS Group 6/RWCU isolation valves automatically close from any one of the following accident mitigating trip signals: low Reactor Vessel water level, cleanup area high temperature, and cleanup inlet high flow. RWCU System return flow to the Reactor Vessel passes through MO-1201-80, check valve 1201-81, and then through inboard feedwater check valve 6-CK-58A. Valve MO-1201-80 and check valve 6-CK-58A are the PCS isolation valves for the RWCU return flow path through PCS penetration X-9A. Valve MO-1201-80 is an Anchor Darling four inch globe valve with a Limitorque SMB-0 actuator. The MO-1201-80 actuator is AC powered with its power supplied from 480 volt Bus B20, breaker B2056. Breaker B2056 is a 480 volt circuit breaker manufactured by Westinghouse Corporation.

EVENT DESCRIPTION

On August 14, 1995, at approximately 1300 hours engineering personnel discovered a discrepancy between the field wiring and the drawings for MO-1201-80. The 'as found' condition would not allow MO-1201-80 to automatically travel to the fully closed position in the event of an isolation signal. The internal wires within breaker cubical B2056 were found attached to terminal points 17 and 18 instead of terminal points 16 and 17, respectively. As a result, during an automatic closure signal the valve would have driven closed and stopped upon the opening of Limit Switch LS-8. The design for the valve is for it to continue to close until the valve operator torque switch is opened when the valve fully closed. MO-1201-80 would have automatically closed to approximately the 96% to 98% closed position when the limit switch would have opened and stopped the valve travel. Technical Specification 3.7.A.2.a. states that all automatic primary containment isolation valves must be operable when Primary Containment Integrity is required except as provided in Technical Specification 3.7.A.2.b. Technical Specification. 3.7.A.2.b states that in the event any automatic Primary Containment Isolation valve becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. The line containing MO-1201-80 also has primary containment check valve 6-CK-58A (inboard feedwater check valve). As discussed below, MO-1201-80 could not have closed fully on an isolation signal between May of 1993 and August 1995, when the condition was discovered. As such MO-1201-80 should have been closed and deactivated in the isolated condition during this period of time as required by Technical Specification 3.7.A.2.b.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	95	008	00	3 of 8

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

Engineers were evaluating Problem Report (PR) 95.9427 when the wiring discrepancy was discovered. Quality Assurance engineers wrote PR 95.9427 during an audit of the Core Spray system. PR 95.9427 identified wiring discrepancies in Core Spray pump 'A' and pump 'B' test return valves MO-1400-4A and MO-1400-4B. These valves are designed to automatically close upon automatic system actuation. The wiring configuration for MO-1400-4B was determined to be electrically equivalent to the design drawings. MO-1400-4A was found with a lifted lead that should have been terminated. The lifted lead in the wiring for MO-1400-4A would have resulted in the valve stopping in the close direction on a limit switch instead of its torque switch. MO-1400-4A would have closed to approximately the 96% to 98% closed position. MO-1400-4A and MO-1400-4B are the same design and model and are expected to have similar flow characteristics. Both valves are six inch wedge gate valves with a safety function to cut off test return flow on a system initiation. Engineers reviewed diagnostic flow test results for MO-1400-4B when it was closed to a similar position as MO-1400-4A. The engineers determined that at the 96% to 98% closed position MO-1400-4A would have closed sufficiently to satisfy the design function. This was due to the wedge gate design and flow cut off function in contrast to the MO-1201-80 throttling globe design and containment isolation function. The Core Spray System was operable with the 'as found' condition of valves MO-1400-4A/B.

Corrective action for PR 95.9427 included field walk-downs and review of the other Core Spray design drawings. No similar problems were found within the Core Spray System. Due to the unique design of the Core Spray Valves, MO-1400-4A/B diagnostic valve testing could be successfully performed without detecting the wiring error. The valves would have closed sufficiently such that Core Spray flow requirements were met even with the 'as found' condition. Of all the valves included in the Generic Letter 89-10 motor operated valve program, only three valves were found to have the susceptibility of avoiding discovery of similar discrepancies during diagnostic and logic system functional testing. These valves were MO-1400-4A, MO-1400-4B and MO-1201-80. The field walk down of the wiring for MO-1201-80 identified the discrepancy described above.

Upon discovery of the wiring discrepancy in MO-1201-80, Problem Report (PR) 95.9439 was written to document the problem and MR 19502529 was written to correct the discrepancy.

The discovery occurred while the plant was operating at 100 percent reactor power with the reactor mode selector switch in the RUN position. The Reactor Vessel pressure was approximately 1030 psig with the Reactor Vessel water at saturation temperature for the reactor pressure.

CAUSE

The root cause of the 'as found' condition of MO-1201-80 was personnel error during a modification installation performed in 1993.

In April of 1993, contract electricians replaced the control power transformer (CPT) for breaker B2056 in accordance with Plant Design Change (PDC) 92-39 via Maintenance Request (MR) 19202111. Removal of the breaker's Motor Control Center (MCC) compartment was required during this work. The MCC compartment is electrically connected to external wiring through a terminal strip located within the compartment. The contract electricians removed the MCC compartment and documented the electrical leads that were removed on the termination summary form of Attachment 1 of Procedure 8.Q.3-3, "480 VAC Motor Control Center Maintenance and Testing". For breaker B2056 to operate as designed, there should be one internal wiring connection on each of the terminal points 16 and 17 with no internal connection on point 18; there should be two external wiring connections each on terminal points 17 and 18, with no external connection on point 16. The two external connections on terminal point 17 should be connected to Control Room Panel C942 and to MO-1201-80 limit switch LS-8. The two external connections on terminal point 18 should be connected to Control Room Panel C942 and to MO-1201-80 limit switch LS-4. The completed termination summary of MR 19202111 indicated the external terminations for points 17 and 18 were returned to the pre-modification configuration.

Engineers investigating the August 1995 'as found' wiring discrepancy in MO-1201-80 found that the April 1993 MCC compartment re-installation resulted in the two pairs of external wires that should have been installed on terminal points 17 and 18 being installed on terminal points 16 and 17, respectively. This resulted in the seal-in circuitry on an automatic closure signal to permit the valve to move in the closed direction only until a limit switch opens versus the design for the valve which is for it to close until the valve operator torque switch is opened. Engineers were able to determine the April 1993 CPT post-modification 'as left' condition when they reviewed the work package for MR 19202808 for work performed in September of 1993. MR 19202808 was written in September 1992 to address Problem Report 92.9176 which identifies a minor wiring discrepancy in the circuitry for MO-1201-80. A September 24, 1992 engineering evaluation concluded the earlier (1992) wiring discrepancy did not impact valve operability but did recommend returning the wiring to the design configuration. MR 19202808 was intended to return MO-1201-80 to its design configuration. The work was performed in September 1993, approximately five months after the CPT replacement work in B2056. The work package for MR 19202808 indicates Boston Edison Company electricians found the two pairs of external wires that should have been installed on terminal points 17 and 18 were installed on terminal points 16 and 17, respectively.

The September 1993 'as found' condition within B2056 was not identified in a Problem Report as required by Procedure (NOP) 92A1, "Problem Report Program". Electrical maintenance personnel contacted an on-site electrical engineer and requested assistance in determining the correct configuration. The maintenance work plan indicates the engineer informed the electricians to terminate the wires in accordance with Drawing E217 sheet 65 (Revision E7). Drawing E217 sheet 65 illustrates only the external wiring for breaker B2056 with the internal wiring being illustrated on Drawing E8-2-7. The maintenance work plan indicates the electricians moved the external and the internal wires from terminal points 16 and 17 to points 17 and 18, respectively, in accordance with drawing E217 sheet 65. There is no mention of the internal wiring diagram E8-2-7. The internal wires should have remained installed on terminal points 16 and 17 in order for MO-1201-80 to close fully automatically. The documented 'as left' configuration of September 1993 was the 'as found' configuration found in August 1995.

A personnel error in April 1993 (RFO-9) resulted in MO-1201-80 being unable to fully close upon an automatic isolation signal. Another personnel error in September 1993, when the earlier condition (April 1993) was not corrected, contributed to the length of time MO-1201-80 was unable to fully close automatically. A contributing cause was that personnel did not document the 'as found' condition on a Problem Report in September 1993. The Problem Report process would have required an evaluation of the condition to assure corrective actions returned the circuitry to the original design configuration.

In May of 1993, plant personnel tested the CPT replacement for MO-1201-80 in accordance with Temporary Procedure (TP) 92-048, "Pre-Op Test of Replaced Control Power Transformers in 480 V MCC (PDC 92-39), Procedure 8.Q.3-3, Procedure 8.Q.3-8, "Limitorque SB/SMB Valve Operator Maintenance", and Procedure 8.6.5.2, "Reactor Water Cleanup Valve Quarterly Operability". These combined tests involved moving the valve using the manual control switch in the Main Control Room and verifying that full valve closure was achieved. This testing was also identified as a contributing cause because the wiring discrepancy was not identified during the post work test.

CORRECTIVE ACTION

Valve MO-1201-80 was closed via its manual control switch and de-energized upon receipt of PR 95.9439 as required by Technical Specification 3.7.A.2.b. Maintenance personnel working under MR 19502529 restored the wiring to the design configuration within B2056 and Operations completed post work testing on August 15, 1995 at 0329 hours.

Training for maintenance and engineering personnel will include a discussion on this specific event including its causes and significance and a review of the requirements for identifying problems as required within NOP92A1.

Engineers determined the uniqueness of the configuration of MO-1400-4A, MO-1400-4B, and MO-1201-80 relative to testing full automatic valve closure. Engineers will review the configuration, current required testing, and make hardware or testing changes of these three uniquely designed valves, as required, to ensure testing will detect full automatic closure.

Engineers are evaluating previous problem reports for wiring discrepancies. This is to identify any commonalties or trends. This may result in further field wiring verifications if other susceptible configurations are identified. A supplemental LER will be written if significant new information is identified.

SAFETY CONSEQUENCES

This condition posed no threat to the public health and safety.

The safety objective of the Primary Containment is to limit the release of fission products in the event of a postulated design basis accident so that off-site doses would not exceed the requirements of 10 CFR Part 100. Engineers completed an engineering assessment on August 17, 1995 on the 'as found' condition of the Primary Containment, including valve MO-1201-80. The evaluation concluded the total containment leakage was less than the Technical Specification limits while the condition with MO-1201-80 existed. Since the Technical Specification limits were not exceeded and the basis of these limits is to prevent off-site doses in excess of 10 CFR 100, the off-site doses would not have exceeded the limits of 10 CFR 100 had an accident occurred while the condition existed. This evaluation considered leakage through check valve 6-CK-58A and did not credit MO-1201-80 and RWCU check valve 1201-81 which is located between MO-1201-80 and 6-CK-58A. The evaluation also included consideration of Primary Containment leak rate testing in RFO-9 (April-May 1993) and RFO-10 (March-June 1995). Based upon the above, the safety objective of the Primary Containment was satisfied during the period the condition existed.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because valve MO-1201-80 was (unknowingly) not capable of automatically closing fully and was not closed and de-energized as required by Technical Specification 3.7.A.2.b.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since 1984. The review focused on LERs involving non-licensed personnel error and the primary containment boundary. The review identified LERs 89-037-01, 90-001-01, and 94-007-00.

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

For LER 89-037-01, on November 30, 1989, one of the Traversing In-Core Probe solenoid operated ball valves that is part of the PCS was discovered to be almost full open when it was thought to be in the closed position. The shear valve, in-series with the ball valve, was operable during the time period the ball valve was open. The failure apparently occurred during manual manipulation of the ball valve on November 15, 1989. Corrective action taken included replacement of the ball valve and actuator and training of Instrument and Control (I&C) Technicians.

For LER 90-001-01 on February 9, 1990, a 24-hour limiting condition for operation was entered because the operability of two (one-inch) reactor coolant system (RCS) instrument line excess flow check valves had been inappropriately verified during a Technical Specification required functional test on November 3, 1989. The other 80 RCS instrument line excess flow check valves were satisfactorily tested. The cause of this event included nomenclature errors in the 1987 excess flow check valve surveillance procedure used as a post modification test, and an inappropriate sign-off of the November 1989 surveillance. Interim compensatory measures taken included increased controls for access and work in the vicinity of the instrument lines, routine visual operation inspections of the instrumentation lines, and issuing a radiation work permit to promptly allow the closing the related manual isolation valves upstream of the check valves if necessary. Corrective actions included the replacement of the two excess flow check valves.

For LER 94-007-00, on December 28, 1994, a plug for the Drywell-to-Torus atmosphere differential pressure transmitter that is part of the Torus atmosphere portion of the PCS pressure boundary was discovered to be not installed. The condition was discovered during calibration of the transmitter while at 100 percent power. Immediate action included the installation of a plug for the transmitter. The cause of was non-licensed I&C Technician and Supervisor error during the previous calibration of the transmitter on November 22, 1994. Corrective action taken included Maintenance Manager direction to maintenance personnel regarding procedure sign-offs and verification, senior management briefings to maintenance personnel regarding the uninstalled plug and management expectations, strengthening the calibration procedure and tagging of instruments that are part of the Primary Containment pressure boundary. Although Primary Containment integrity, as specified by Technical Specification 3.7.A.2.a, was not maintained at all times, evaluation of the potential radiological consequences of the uninstalled plug concluded that 10 CFR 100 and 10 CFR 50 Appendix A exposure guidelines would not have been exceeded if a design basis accident had occurred while the plug was not installed.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	95	008	00	8 of 8

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

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Breaker (B2056)	BKR
Operating Mechanism (Motor Operator)	84
Valve, Electrically Operated (MO-1201-80)	20

SYSTEMS	
Reactor Water Cleanup (RWCU) System	CE
Primary Containment System (PCS)	JM
Containment Isolation Control System (PCIS)	JM
Containment Leakage Control System	BD