10 CFR 50.73

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Roy A. Anderson
Senior Vice President — Nuclear

April 24 , 1992
BECo Ltr. 92- 048

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 92-003-00, "Reactor Core Isolation Cooling System Made Inoperable per Technical Specifications due to an Inoperable Primary Containment System Isolation Valve", is submitted in accordance with 10 CFR Part 50.73.

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

R. A. Anderson

JPC/bal

Enclosure: LER 92-003-00

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NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Reactor Core Isolation Cooling System Made Inoperable per Technical Specifications due to an Inoperable Primary Containment System Isolation Valve

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 NAMES	DOCKET NUMBER(S)
0 3	2 5	9 2	9 2	0 0 3	0 0	0 4	2 4	9 2	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9)	N
POWER LEVEL (10)	1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v) D		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	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	
	AREA CODE	
Jeffery P. Calfa - Senior Compliance Engineer	5 0 8	7 4 7 - 8 1 0 8

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
D	B N	I S V	W 1 2 0	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 fifteen single-space typewritten lines) (16)

On March 25, 1992 at 0059 hours, the unsatisfactory position indication for a Primary Containment System (PCS) isolation valve located within the Drywell caused the Reactor Core Isolation Cooling (RCIC) System to be declared inoperable. The reactor was shutdown in order to repair the valve. The repairs were not expected to exceed the seven day Limiting Condition of Operation of Technical Specification 3.5.D.2.

The RCIC System was removed from service on March 23, 1992 for minor maintenance. While preparing to return RCIC to service, the inboard steam isolation valve, MO-1301-16, indicated closed but was actually open. The valve could not be closed. Following shutdown, the MO-1301-16 motor operator was found to be detached from its yoke. The motor operator-to-valve yoke attaching capscrews had been insufficiently torqued during maintenance. A table in the procedure used to torque the capscrews incorrectly referenced mild steel for the capscrew material and the valve drawing incorrectly referenced the capscrew size. The drawing and the procedure have been revised. The MO-1301-16 actuator has been installed using the valve manufacturer's requirements. All other safety-related valve motor operator fasteners have been either torque checked for proper fastener preload or visually inspected to verify the fasteners were in place. This condition occurred during power operation at 100 percent reactor power. The reactor mode selector switch was in the RUN position. Reactor pressure was approximately 1030 psig and reactor water temperature was approximately 550 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(v)(D). This condition posed no threat to the public health and safety.

EVENT DESCRIPTION

On March 25, 1992 at 0059 hours, the unsatisfactory position indication for a Primary Containment System (PCS) isolation valve located within the Drywell caused the Reactor Core Isolation Cooling (RCIC) System to be declared inoperable. The reactor was shut down in order to repair the valve. The repairs were not expected to exceed the RCIC System seven day Limiting Condition of Operation of Technical Specification 3.5.D.2.

The RCIC System was removed from service for planned maintenance on March 23, 1992 at 0500 hours. On March 25, 1992, the RCIC turbine steam supply piping was being lined up for warming in preparation for returning the system to service. At 0059 hours, the outboard steam isolation valve, MO-1301-17, was opened. RCIC turbine steam inlet pressure increased to approximately 1000 psig as measured downstream of MO-1301-17. This was unexpected as valve MO-1301-16 position indicated closed. The MO-1301-17 valve was then closed and RCIC inlet pressure began to decrease. The MO-1301-17 valve was reopened and pressure returned to vessel pressure. At this point MO-1301-16 was stroked open and closed several times with control room light indication changing respectively. MO-1301-17 was then closed and RCIC turbine inlet pressure once again decayed away gradually. Approximately five minutes after the MO-1301-17 valve was closed, light indication for MO-1301-16 changed from closed to open (green to red) with no operator action. MO-1301-17 was closed and de-activated in accordance with Technical Specification 3.7.A.2.b, thereby, making the RCIC System inoperable.

A normal reactor shutdown began at 0009 hours on March 26, 1992 and shutdown was completed at 1438 hours when all control rods were in the fully inserted position.

Failure and Malfunction Report 92-76 was written to document the problem with MO-1301-16. The NRC Operations Center had been notified of RCIC being made inoperable for maintenance on March 23, 1992 at 0515 hours.

The problem with MO-1301-16 occurred during power operation with the reactor mode selector switch in the RUN position. The reactor power level was 100 percent. Reactor pressure was approximately 1030 psig and reactor water temperature was approximately 550 degrees Fahrenheit.

CAUSE

A Failure Analysis Team comprised of Technical Section and Nuclear Engineering Department (NED) personnel was formed to investigate the problem with MO-1301-16. Upon entry to the Drywell, plant personnel discovered the MO-1301-16 motor operator had become disconnected from the valve yoke. The actuator-to-yoke capscrews had loosened and backed out of the sockets. The valve operator had risen up the valve stem.

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

The root cause was insufficient torquing of the actuator-to-yoke fasteners. This was caused by errors within the installation procedure and the valve drawing. MO-1301-16 is a three inch motor operated gate valve manufactured by Westinghouse. The motor operator is a Limitorque Model SMB-000-10. Maintenance was last performed on the valve in June of 1991. Additionally, the attaching bolts had been torqued to 8 foot pounds as required for 5/16 inch fasteners by revision 7 of Procedure 3.M.3-24.1, "Limitorque Valve Operator Removal and Reinstallation".

The procedure contained a table which provided the required torque for fasteners of various sizes and materials. A note on the table stated Allen Head Capscrews should be torqued using the lesser mild steel requirements. The mild steel torque requirements were used on the MO-1301-16 in accordance with this procedure. However, the cap screws were not mild steel and required a higher torque. During the root cause investigation, Limitorque Corporation was contacted for guidance on the proper torque to be used for the operator-to-yoke capscrews. Limitorque advised they had addressed this issue generically and the valve manufacturer should provide the torque to be used, as the valve manufacturer provided the original fasteners as well as the valve/actuator seismic analysis. While valve drawing M137A-1 indicates the capscrews are 5/16 inch diameter, the actual diameter is 3/8 inch. Typically, SMB-000 actuators are attached with 5/16" inch capscrews, but the MO-1301-16 Westinghouse valve was unique in using 3/8 inch capscrews. The valve manual recommends a torque of 30 foot pounds plus or minus 3 foot pounds for 3/8 inch cap screws. The MO-1301-16 Allen Head Capscrews and the operator threaded sockets were inspected and no damage was observed.

Subsequent to overhauling the motor operator in July of 1991, the MO-1301-16 torque switch setting was increased from an "as-found" maximum thrust of 6583 lbs. to an "as left" maximum thrust of 7412 lbs. This was within existing design margins for the valve. This was done to address NRC Generic Letter 89-10. This would explain why the operator detached from the yoke at this time and not in the past.

During the maintenance overhaul of MO-1301-16 following the operator detachment, the capscrews were found to have only 1/8 inch of thread engagement into the operator due to their length and the thickness of the flange. As a design enhancement the original one inch long 3/8 inch diameter capscrews were replaced with 1 1/2 inch long 3/8 inch diameter capscrews.

CORRECTIVE ACTION

NED personnel provided one-time torque checking requirements to check for adequate pre-load for the actuator attachment fasteners for safety-related motor operated valves. These requirements took into account the operator thrust settings. These torque checking requirements were incorporated by reference into revision 9 of Procedure 3.M.3-24.1 on March 31, 1992. Procedure 3.M.3-24.1 will be further revised to remove the one-time torque checking and only refer technicians to valve manufacturer manuals or drawings or Procedure 3.M.4-92, "Bolting and Torquing Guidelines", as appropriate.

Maintenance personnel either checked the torque or performed a visual inspection of all safety-related motor operated valve actuator attaching fasteners prior to plant startup. All valves in the Drywell safety or non-safety related were torque checked. The valve fasteners that were torqued were chosen based on their location in the Drywell or their safety significance. Minor discrepancies were found in the visual inspections and problem reports have been initiated. The valve fasteners that were visually inspected will be torque checked after startup.

The motor operator for MO-1301-16 was overhauled and re-installed using new 3/8 inch capscrews with the torque requirements provided by the valve manufacturer and verified by Engineering. The valve was diagnostically tested with satisfactory results.

Valve drawing M137A-1 has been revised to show the actuator-to-yoke attaching bolts as 3/8 inch capscrews.

SAFETY CONSEQUENCES

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

The RCIC System was operable prior to being taken out of service for maintenance on March 23, 1992. The inability to close MO-1301-16 would not have prevented RCIC from providing makeup water to the reactor vessel following a reactor vessel isolation.

The High Pressure Coolant Injection System was operable during the period RCIC was inoperable.

In the event a Primary Containment System (PCS) isolation signal had occurred after the MO-1301-16 actuator detached from its yoke, the in-series outboard RCIC Turbine Steam Supply Isolation Valve MO-1301-17 would have provided the containment isolation function. After the shutdown, MO-1301-17 had a local leak rate test. The "as found" leakage rate was essentially zero leakage.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(v)(D) because RCIC was made inoperable.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs). The review focused on LERs involving loose or detached motor operator-to-yoke fasteners. The review identified two LERs in which a safety related motor operator-to-yoke fasteners became detached from the valve.

For both LERs 83-010/03L-0 and 83-035/03L-0, the attaching capscrews became detached from 'A' Core Spray Test Valve, MO-1400-4A, on February 22, 1983 and June 10, 1983, respectively. The valve yokes and fasteners for both MO-1400-4A and MO-1400-4B were replaced with new yokes and fasteners.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Valve, Isolation (MO-1301-16)	ISV
SYSTEMS	
Primary Containment System	PCS
Reactor Core Isolation Cooling (RCIC) System	BN