

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION

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J. DOERING, JR.
PLANT MANAGER
LIMERICK GENERATING STATION

May 10, 1991

Docket No. 50-353
License No. NPF-85

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

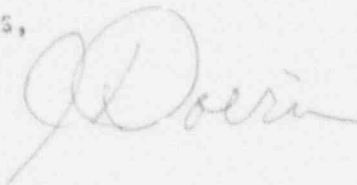
SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 2

This LER reports a condition of the refueling platform monorail mounted auxiliary hoist where operation of the uptravel mechanical stop permitted the handling of a control rod outside the limits of Technical Specifications due to procedure deficiencies.

Reference:	Docket No. 50-353
Report Number:	2-91-007
Revision Number:	00
Event Date:	April 10, 1991
Report Date:	May 10, 1991
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B).

Very truly yours,



JLP:cah

cc: T. T. Martin, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Limerick Generating Station, Unit 2										DOCKET NUMBER (2) 0 5 0 0 0 3 3 3 1 OF 0 5					PAGE (3) 1 OF 0 5									
TITLE (4) Handling of a control rod prohibited by Technical Specifications Surveillance Requirements 4.9.6.3b due to procedure deficiencies.																								
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	4	1	0	9	1	0	0	7	0	0	0	5	1	0	9	1	0	5	0	0	0	0	0	0
OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5 (Check one or more of the following) (11)																						
5		20.402(b)				20.405(c)				50.73(a)(2)(i)(x)				73.71(b)										
POWER LEVEL (10)		0 0 0				20.405(a)(1)(i)				50.73(a)(2)(iv)				73.71(c)										
		20.405(a)(1)(ii)				50.73(a)(2)(v)				50.73(a)(2)(vii)(A)				OTHER (Specify in Abstract below and in Text, NRC Form 368A)										
		20.405(a)(1)(iii)				50.73(a)(2)(vi)				50.73(a)(2)(viii)(B)														
		20.405(a)(1)(iv)				50.73(a)(2)(iii)				50.73(a)(2)(ix)														
		20.405(a)(1)(v)				50.73(a)(2)(ii)				50.73(a)(2)(x)														
LICENSEE CONTACT FOR THIS LER (12)																								
NAME G. J. Madsen, Regulatory Engineer, Limerick Generating Station										TELEPHONE NUMBER														
										AREA CODE 2 1 5 3 2 7 1 - 1 2 0 0														
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)																								
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC					
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 April 10, 1991, a Fuel Handling Supervisor observed the uptravel mechanical stop on the refueling platform monorail mounted auxiliary hoist to be mispositioned outside the limits of Surveillance Test procedure ST-6-107-633-2, "Core Alteration Testing." During the previous shift the uptravel mechanical stop was moved 24 inches to permit movement of a control rod blade from the spent fuel pool to the reactor. On April 12, 1991, during investigation into the event, it was determined that the top of a control rod blade was 1.5 inches less than the required 6 feet 6 inches below the normal fuel storage pool water level; a condition prohibited by Technical Specifications (TS) Surveillance Requirement (SR) 4.9.6.3b. TS SR 4.9.6.3b was amended on August 16, 1990. Prior to August 16, 1990, the interpretation of TS SR 4.9.6.3b did not limit control rod blade movement. The cause of this event was an inadequate maintenance procedure M-041-061 and an inadequate procedure ST-6-107-633-2. A contributing cause to the event was inadequate training of the Fuel Handling Supervisors on invessel maintenance activities following the change to TS SR 4.9.6.3b. The maintenance procedure has been changed to delete the provision to reposition the mechanical stop. Procedure ST-6-107-633-2 has also been revised. Additionally, the appropriate requalification training programs will be revised, the TS amendment control procedure will be revised and previously issued TS amendments will be reviewed to determine if a similar problem occurred.

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Limerick Generating Station, Unit 2	0 5 0 0 0 3 5 3	9 1	— 0 0 7	— 0 0	0	2	OF 0 5

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

Unit Conditions Prior to the Event:

Unit 2 Operational Condition was 5 (Refueling) at 0% Power Level.

There were no structures, systems or components out of service or being tested which contributed to this event.

Description of the Event:

On April 10, 1991, at 1630 hours, a utility employed, licensed Fuel Handling Supervisor observed the uptravel mechanical stop on the refueling platform monorail mounted auxiliary hoist to be mispositioned outside the limits of Surveillance Test (ST) procedure ST-6-107-633-2, "Core Alteration Testing." An investigation was initiated to determine if Technical Specifications (TS) Surveillance Requirement (SR) 4.9.6.3b was violated. TS SR 4.9.6.3b requires demonstrating operation of the uptravel mechanical stop on the monorail hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level. This SR is required to be performed within seven days prior to the use of such equipment.

On April 9, 1991, procedure ST-6-107-633-2 was performed to ensure compliance with TS SR 4.9.6.3b prior to control rod blade (CRB) movement. Procedure ST-6-107-633-2 sets the mechanical stop to ensure the loose end of the hoist cables, with no tools attached, will remain at least 5 feet below the surface of the water. With an invessel tool attached, the distance from the hoist cable loose end to the top of the CRB bail handle provides the other 18 inches necessary to meet the 6 feet 6 inch TS SR. The actual distances provided are 19.875 inches by the CRB grapple, 20.5 inches by the jet pump grapple, and 39.5 inches by the control rod blade latch tool (Figure 1). A CRB can be transferred between the reactor vessel and the spent fuel pool following performance of procedure ST-6-107-633-2 using the CRB grapple or jet pump grapple but not by using the CRB latch tool. While the CRB grapple and jet pump grapple provide 9.125 inches of clearance from the bottom of the CRB to the transfer canal (cattle chute) floor, the CRB latch tool extends the bottom of the CRB 10.25 inches below the transfer canal floor.

On April 10, 1991 at 1240 hours, the mechanical stop was moved 24 inches to transfer a CRB from the spent fuel pool to the reactor vessel using the CRB latch tool. This resulted in the top of the CRB being 6 feet 3.5 inches below the surface of the water in violation of TS SR 4.9.6.3b. At shift turnover on April 10, 1991, at 1630 hours the oncoming, a licensed Fuel Handling Supervisor observed the uptravel mechanical stop to be mispositioned. The mechanical stop was reset and procedure ST-6-107-633-2 was performed to verify compliance with TS SR 4.9.6.3b.

On April 12, 1991, during investigation into this event, the actual measurements of the hoist cable and the tools and the distance that the mechanical stop was moved were obtained and analyzed. Station personnel then determined that the TS

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/85

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TEXT (If more space is required, use additional NRC Form 266A's) (17)

SR 4.9.6.3b was not met and the action of TS 3.9.6, suspension of handling CRB's, was not complied with. This resulted in a condition prohibited by TS. This report is being submitted in accordance with the requirements of 10CFR50.73(a)(2)(1)(B).

Analysis of the Event:

The consequences of this event were minimal. There was no release of radioactive material to the environment as a result of this event. A local radiation survey meter in use did not register any change in area radiation levels during this event. Any increased radiation dose to the operators from a 2.5 inch reduction in water shielding was of a limited duration. The dosimetry of the operators was checked and the radiation exposure was found to be similar to that collected on previous shifts. The possible consequences from this event was bounded by the fuel handling accident discussed in section 15.7.4 of the Updated Final Safety Analysis Report (UFSAR). Fuel handling personnel reset the mechanical stop and performed procedure ST-6-107-633-2 to verify compliance with TS SR 4.9.6.3b and limit the event duration.

Cause of the Event:

The cause of the event was an inadequate maintenance procedure M-041-061, "Unlatching and Shuffling of Control Rod Blades," and an inadequate ST procedure ST-6-107-633-2. Procedure M-041-061 allowed adjustment of the mechanical stop to raise a CRB to clear the transfer canal floor without identifying the maximum amount of adjustment permitted. Procedure ST-6-107-633-2 positioned the mechanical stop at the same height irrespective of the tool to be used to move a CRB even though the height position was unsuitable for the tool normally used to move blades between the spent fuel pool and the reactor vessel (i.e. the CRB latch tool). A contributing cause to the M-041-061 procedure deficiency was a failure of the Licensing group to notify maintenance that TS SR 4.9.6.3b was amended on August 16, 1990, which changed the limit for handling CRB's.

A contributing cause to the event was inadequate training of the licensed Fuel Handling Supervisors on in-vessel maintenance activities (those maintenance activities carried out inside the reactor vessel that are not core alterations) following the August 16, 1990 change to ST SR 4.9.6.3b. Prior to August 16, 1990, the interpretation of the operability requirement for the mechanical stop did not limit CRB movement. TS SR 4.9.6.3b originally limited the uptravel of the top of active fuel to eight feet six inches below the fuel pool water level while UFSAR section 9.1.4.3 analyzes CRB movement with seven feet of water shielding. After August 16, 1990, the training provided to the licensed Fuel Handling Supervisors did not adequately explain this change and the impact in CRB movement.

Additionally, the adequacy of compliance with TS SR 4.9.6.3b after August 16, 1990, was not determined by either the plant staff self assessment or by Nuclear Quality Assurance (NQA) Department.

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

Insufficient evidence exists to determine if TS SR 4.9.6.3b was violated between August 16, 1990, and April 9, 1990.

Corrective Actions:

The mechanical stop was reset and procedure ST-6-107-633-2 was performed to verify compliance with TS SR 4.9.6.3b. The Plant Manager issued a letter on April 11, 1991, forbidding further adjustment of the mechanical stop until authorized by the Plant Manager or his designated alternate. Procedure M-041-061 was revised on April 19, 1991, to delete the provision that permitted repositioning of the mechanical stop. CRB movement from the spent fuel pool to the reactor vessel was completed using the CRB grapple which allows CRB passage through the transfer canal without mechanical stop adjustment. Procedure ST-6-107-633-2 has been revised to set the mechanical stop for the CRB latch tool at an appropriate height that is different from the height necessary for the CRB and jet pump grapple.

The TS Amendment issued on August 16, 1990, was included in Licensed Operator required reading in October, 1990, but no training was provided to Fuel Handling Supervisors on TS SR 4.9.6.3b. In vessel maintenance activities will be included in initial and requalification training of licensed fuel handling supervisors.

In order to ensure that there were no similar problems occurred related to inadequate communication and review of TS amendments involving refueling and in-vessel maintenance activities, all TS amendments issued from August, 1990, to the present were reviewed. No additional TS amendment was found to affect any refueling or in-vessel maintenance activity. Administrative procedure A-29, "Control of Revisions Due to License Document Revisions," will be revised by July 31, 1991, so that all affected organizations will be notified of future amendments so that all effected procedures are revised. Additionally all TS amendments implemented since issuance of the Unit 2 license will be reviewed to identify any deficiencies in implementation. Administrative procedure AG-82, "Self Assessment," will be revised by July 31, 1991 to require that procedure and program changes to be the subject of self-assessment within the same self assessment cycle. Procedure A-29 will also be reviewed to provide copies of all future TS amendments to the NQA Department. The NQA Department will review the change and perform an appropriate review of the implementation of the TS amendment.

Previous Similar Occurrences:

None

Tracking Codes: D2 - Inadequate procedure did not cover situation
A99 - Other personnel errors

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Limerick Generating Station, Unit 2	05000353	91	007	00	05	OF	05

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

FIGURE 1

AXIAL MEASUREMENTS FOR VARIOUS SFP TOOLS

