AmerGen

An Exelon/British Energy Company

Clinton Power Station
R.R. 3 Box 228
Clinton, IL 61727-9351
Phone: 217 935-8881

U-603524
2C.220
September 21, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Clinton Power Station
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Licensee Event Report No. 2001-003-00

Enclosed is Licensee Event Report (LER) No. 2001-003-00: Failure to Follow Procedure Due to Human Error Results in Incorrectly Lifting Leads, Causing Loss of Feedwater Level Signal, and Reactor Scram on High Vessel Water Level. This report is being submitted in accordance with the requirements of 10CFR50.73.

Respectfully,

J. M. Heffley
Vice President
Clinton Power Station

RSF/blf

Enclosure

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

IE22

NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Clinton Power Station	05000461	1 OF 5

TITLE (4)
Failure to Follow Procedure Due to Human Error Results in Incorrectly Lifting Leads, Causing Loss of Feedwater Level Signal, and Reactor Scram on High Vessel Water Level

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME None	DOCKET NUMBER 05000
07	24	2001	2001 –	003 –	00	09	21	2001	FACILITY NAME None	DOCKET NUMBER 05000

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
OPERATING MODE (9)	1		20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
POWER LEVEL (10)	100		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
P. J. O'Reilly, Outage Scheduler	(217) 937-3864

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

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)

Instrument Maintenance (IM) technicians were performing a quarterly channel functional test on the B Reactor Feedwater level channel with the A reactor feedwater level channel in service. During the surveillance test, the technicians incorrectly performed a step in the procedure and lifted the Channel B-Device GT leads. Per procedure, the step for lifting the leads is only performed if a high reactor water level condition exists, and no such condition existed. Lifting the leads caused a loss of the reactor vessel level input signal for feedwater control, resulting in an immediate demand by the Master Level Controller. A feedwater transient began that resulted in reactor water level rising above the High Level 8 trip initiating an automatic reactor scram. The cause of this event was human error. The technicians involved did not follow the procedure and performed a step not applicable to current plant conditions. Corrective actions for this event include personnel actions for the technicians involved in the event, revising the surveillance procedure and reviewing other IM procedures, issuing a risk review and screening procedure, training on various topics, and increasing management oversight on backshifts.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Clinton Power Station	05000461	2001	-- 003 --	00	2 OF 5

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

DESCRIPTION OF EVENT

On July 24, 2001, the plant was in Mode 1 (POWER OPERATION) with the reactor [RCT] at 100 percent power. At about 0234 hours, Instrument Maintenance (IM) technicians commenced a quarterly channel functional test on the B Reactor Feedwater [SJ] level channel in accordance with surveillance procedure CPS 9538.03, Feedwater Reactor Vessel Water Level C34-N004A (B, C) Channel Functional. The A reactor feedwater level channel was in service.

At about 0251 hours, during the surveillance test, the IM technicians incorrectly performed a step in surveillance procedure CPS 9538.03 and lifted the Channel B-Device GT leads. The procedure directs that the step is only performed if a high reactor water level condition exists, and no such condition existed at the time of the surveillance test. Lifting the leads caused the REACTOR HIGH WATER LEVEL TRIP alarm [ALM] to actuate. Since the A channel was selected as the Feedwater Level Control System [JB] mode, incorrectly lifting the leads caused an interruption of the reactor vessel level input signal to the feedwater control circuit, resulting in an immediate demand for feedwater flow by the Master Level Controller.

A feedwater level transient began as Indicated Narrow Range reactor level (the sum of the A, B, and C feedwater level channel inputs) started to lower, and the feedwater level control system responded by demanding higher feed flow.

The RPV WATER LEVEL HIGH OR LOW alarm that alerts operators of a high water level condition did not actuate due to indicated level lowering. When reactor water level reached the High Level 8 trip setpoint, an automatic reactor scram initiated.

In response to the scram, operators entered off-normal procedures CPS 4100.01, Reactor Scram, and CPS 4002.01, Abnormal RPV Level/Loss of Feedwater at Power, placed the reactor mode switch [HS] in the shutdown position, and verified all control rods were fully inserted. Immediate actions were taken to establish a reactor pressure band of 800 to 1065 pounds per square inch gage (psig) using Turbine Bypass Valves [V], and to maintain reactor water level in a band of Level 3 to Level 8 with the plant in Mode 3 (HOT SHUTDOWN).

At the time of the scram, reactor water level dropped to the Low Level 3 trip setpoint with feedwater still feeding the vessel at a high rate. At this point, operators entered Emergency Operating Procedure (EOP) 1, Reactor Pressure Vessel Control. Additionally, at the Low Level 3 trip, primary containment isolation valves [ISV] in Groups 2 (Residual Heat Removal [BO]), 3 (BO), and 20 (miscellaneous systems) received signals to shut. Operators manually tripped the A Turbine Driven Reactor Feed Pump [P] (TDRFP) per the reactor scram off-normal procedure. Reactor water level again reached the High Level trip setpoint and automatically tripped the Main Turbine [TA] and the B TDRFP. Recorded reactor water level reached the Main Steam [SB] Lines, however, there was no control room indication that moisture entered the steam lines.

The reactor scram logic was reset at about 0324 hours. By about 0453 hours, the plant was in a stable condition, and operators exited from EOP-1.

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

The plant responded as expected to the lifting of the leads. The surveillance procedure was suspended and the lifted leads were re-landed to permit restoration of the feedwater level control system. There was no damage to equipment or personnel injuries associated with this event.

In response to the automatic containment isolation signal, operators completed the Automatic Isolation Checklist and verified that primary containment isolation valves responded as expected.

During the reactor scram, the Division 2 Nuclear Systems Protection System [JG] inverter [INVT] transferred from its normal source to its alternate source. The inverter is not considered operable unless it is being supplied by its normal source. The inverter was restored to the normal source per system procedure CPS 3509.01, Instrument Power System, at 0840 hours and declared operable. The transfer to the alternate source was evaluated by engineering and determined to be a spurious operation of the static switch. The engineering evaluation concluded that certain Division 2 instrumentation would not be available during a Station Blackout (SBO); however, all monitored parameters have Divisions 1 and 2 instruments. Therefore, operators would have the capability to monitor essential plant parameters using Division 1 instrumentation.

No other automatic or manually initiated safety system responses were necessary to place the plant in a safe and stable condition. Other inoperable equipment or components did not directly affect this event.

Condition Report (CR) 64602 was initiated to investigate the cause and identify corrective actions for this event.

Investigation of this event identified that IM personnel were scheduled to perform several activities during their shift period (2300 to 0700 hours). These activities included installation of a resistance temperature detector to diesel generator heat exchangers, surveillance testing for Average Power Range Monitor Gain and Flow Gain adjustments, and the feedwater channel functional test. The IM Group Leader held a pre-job brief for the feedwater channel functional test; however, the briefing was not completed in accordance with the procedure requirements. Additionally, the briefing focused on two items, ensuring the correct channel was tested (3 channels, A, B, and C were to be tested) and coordinating the switching of channels with Operations. Only one of the two technicians performing the surveillance was present at this briefing. Prior to beginning the work, the technicians were diverted to a higher priority task.

Following the completion of the higher priority task, the IM technicians informed their Group Leader and returned to the MCR to perform the feedwater channel functional test on the B and C channels. The Work Control Supervisor (WCS) performed a partial review of the test procedure and released the test for performance. The technicians signed on to the test with the Control Room Supervisor. A risk review was not performed, and performance of the surveillance test was not identified as a high level of awareness activity (HLA). No critical points were identified in the procedure and during reviews no additional focus was placed on the performance of the procedure. Reviews for risk, HLAs, and identification of critical points were not expectations at the time of this event. The focus remained on the activity to swap feedwater channels after the performance of testing on channels B and C.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Clinton Power Station	05000461	2001	-- 003 --	00	4 OF 5

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

CAUSE OF EVENT

The cause of this event is personnel performance (human error). During the performance of surveillance CPS 9538.03, the test performer is procedurally directed to perform step 8.1.1.1 only if a high reactor water level condition existed, otherwise the direction is to proceed to step 8.1.2. During this event, a high reactor water level condition did not exist, yet the IM technician inappropriately performed step 8.1.1.1 instead of proceeding to step 8.1.2. Surveillance procedure CPS 9538.03 had been performed over 90 times successfully with various levels of preparation for the task.

Two contributing factors were identified during the cause investigation for this event. First, adequate work direction was not provided for overall preparation, execution and oversight of the task. If this task had been adequately prepared and managed, this event would have been prevented. Second, the surveillance procedure contained steps that created an inappropriate risk. The procedure contained actions that would possibly be performed once every refueling cycle.

CORRECTIVE ACTIONS

Personnel actions have been taken regarding the performance issues of the Group Leader and technicians involved in this event.

An IM supervisor (management) is now on each of the backshifts (they had been hired and were being qualified when this event occurred). For the remainder of this year, a senior management level representative is observing plant activities during off-dayshift hours to assist in coaching and mentoring personnel.

Procedure HU-AA-101, Human Performance Tools and Verification Practices, has been implemented to provide workers with several human performance tools to promote safe, error-free operation.

Training & Reference Material AD-AA-1211, Pre-Job, Heightened Level of Awareness, Infrequent Plant Activity and Post-Job Briefings, has been implemented. This document provides direction for performing pre-job briefs for normal activities, HLA briefings for more significant activities, infrequent plant activity briefings for HLA activities of particular importance requiring senior line manager involvement, and post-job critiques to capture lessons learned and opportunities for improvement.

Briefings were conducted with Maintenance personnel immediately following the event to reinforce expectations for pre-job briefings and peer checking. Anatomy of event training was given to Maintenance personnel covering procedure adherence, place-keeping, noting procedure steps as not applicable, pre-job briefs, concurrent verification and the use of critical steps.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Clinton Power Station	05000461	2001	-- 003 --	00	5 OF 5

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

A risk review is now required for all activities in accordance with procedure WC-AA-104, Review and Screening for Production Risk. A risk review would have identified surveillance procedure CPS 9583.03 as a high-risk activity and step 8.1 as a critical point, and would have added significant defenses to the performance of the surveillance. In addition the risk review would have involved senior management and Operations in HLA briefings and in various oversight roles.

Surveillance procedure CPS 9538.03, has been revised and other IM procedures have been verified to not have similar multiple mode issues (outage vs. on-line).

Anatomy of event training was given to Maintenance personnel covering the Roles of the First Line Supervisor and Group Leader.

Classroom training will be provided to the Maintenance personnel on procedure use and adherence, human performance tools and verification practices, and reviewing and screening work for high production risk activities and work authorization.

A Dynamic Learning Activity will be provided for IM technicians on the performance of surveillance testing to reinforce procedure expectations.

ANALYSIS OF EVENT

This event is reportable under the provision of 10CFR50.73(a)(2)(iv)(A) due to an automatic actuation of the reactor protection system [JC]. This event was compared to the analyzed transients discussed in Chapter 15 of the Updated Safety Analysis Report (USAR) and was determined to be within the design basis of the plant.

ADDITIONAL INFORMATION

No equipment failed as a result of this event.

A review of Clinton Power Station events for the previous 3 years did not identify any reactor scrams caused by failure to follow surveillance test procedures.

For further information regarding this event, contact P. J. O'Reilly, Outage Scheduler at (217) 937-3864.