AmerGen
An Exelon/British Energy Company

Clinton Power Station
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10 CFR 50.73

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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. 2002-002-00

Enclosed is Licensee Event Report (LER) No. 2002-002-00: Inadequate Preventive Maintenance Program for the Feedwater System Results in Lockup of a Turbine-Driven Reactor Feed Pump and Scram on High Reactor Pressure Vessel Water Level During Extended Power Uprate Testing. This report is being submitted in accordance with the requirements of 10CFR50.73.

Respectfully,

M. J. Pacilio
Vice President
Clinton Power Station

RSF/blf

Enclosures

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.

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TITLE (4)

Inadequate Preventive Maintenance Program for the Feedwater System Results in Lockup of a Turbine-Driven Reactor Feed Pump and Scram on High Reactor Pressure Vessel Water Level During Extended Power Uprate Testing

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
05	13	2002	2002	– 002 –	00	07	11	02	FACILITY NAME: None	DOCKET NUMBER: 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	086

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	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)
T. W. Parrent, Station Engineering	(217) 937-3382

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
X	SJ	TRB	G084	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)**

Extended Power Uprate testing was in progress with the 'A' Turbine-Driven Reactor Feed Pump (TDRFP) in manual on its flow controller and the 'B' TDRFP in automatic on the Startup Level Controller (SLC). When a planned reactor pressure vessel (RPV) water level increase from 32 to 38 inches was performed for startup testing using the SLC, water level increased but the Feedwater Level Control System (FLCS) did not adjust to stabilize water level. The operator took manual control of the SLC and lowered the demand signal to the 'B' TDRFP, but level continued to rise and an automatic reactor scram occurred on high RPV water level before insertion of a manual scram. The 'B' TDRFP did not respond to the lowered demand signal due to a lock-up of the TDRFP from mechanical binding. The cause of this event is an inadequate preventive maintenance program for the Feedwater System that did not identify the TDRFP limit switch guide as a critical component and establish appropriate preventive maintenance to prevent its failure. Corrective action includes developing a preventive maintenance task to inspect, clean and lubricate the limit switch guide mechanism periodically; replacing the 'B' TDRFP horizontal linkage rod, and reviewing the preventive maintenance program for the Feedwater System to assure critical components have adequate preventive maintenance tasks assigned.

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

PLANT CONDITIONS PRIOR TO THE EVENT:

Unit: 1; Event Date: 05/13/02; Event Time: 0016 Central Daylight Time
MODE: 1 (POWER OPERATION); Reactor Power: 086 percent

DESCRIPTION OF EVENT

On May 13, 2002, Extended Power Uprate (EPU) testing was in progress to verify that the Feedwater Level Control System (FLCS) [JB] would respond properly to incremental changes in feedwater flow and reactor pressure vessel (RPV) water level. The testing was being performed in accordance with test procedure CPS 2811.00, "EPU Feedwater Level Control Regulation Test". The procedure included testing each Turbine [TRB]-Driven Reactor Feed Pump (TDRFP) [P] in single element control and in 3-element control. The testing included inserting step level changes in reactor feedwater level of plus or minus 1, 2, 3 and 6 inches into the control system at each power level from 83 to 95 percent in 3 percent power level increments. Key FLCS parameters were to be monitored during the step changes to verify adequate system response. In support of testing the step level changes in single element control, the 'A' TDRFP was being operated in manual on its flow controller and the 'B' TDRFP was being operated in automatic on the Startup Level Controller. The power level was set at 86 percent. The step level change of plus or minus 3 inches had just been completed successfully, with the FLCS adjusting to stabilize RPV water level.

The step to raise the level setpoint up to the plus or minus 6 inches test level, from 32 inches narrow range to 38 inches narrow range, was performed using the setpoint thumbwheel on the Startup Level Controller. In response to the 6-inch step level change, RPV water level increased as expected, but the FLCS did not adjust to stabilize RPV water level as it had in the previous tests. The Reactor Operator took manual control of the Startup Level Controller and lowered the demand signal to the 'B' TDRFP, but RPV water level continued to trend up. When RPV water level reached 48 inches narrow range indication at about 0013 hours, the Reactor Operator placed the reactor mode switch [HS] into the shutdown position and inserted a reactor scram signal. A later review of the alarm printer [PRNT] identified that an automatic reactor scram occurred on high RPV water Level 8 (52 inches narrow range indication) just moments before the Reactor Operator inserted the manual scram. RPV water level reached as high as 90 inches upset range indication. The Main Turbine and both TDRFPs tripped off due to the Level 8 RPV water level trip.

Following the scram, at 0017 hours, RPV water level dropped to the low RPV water Level 3 trip setpoint (8.9 inches narrow range indication) as expected. In response to the reactor scram and the lowering RPV water level, operators entered off-normal procedure 4100.01, "Reactor Scram", and Emergency Operating Procedure (EOP) -1, "RPV Control". The low Level 3 RPV water level trip caused primary Containment isolation valves [ISV] in groups 2 (Residual Heat Removal (RHR)[BO]), 3 (RHR), and 20 (miscellaneous systems) to receive signals to shut. These valves were shut prior to the event in accordance with the normal plant lineup.

Operators reset the scram signal at 0042 hours. By 0142 hours, the plant was in a stable condition and plant parameters were being controlled in accordance with normal station procedures, operators exited EOP-1.

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

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

During this event the Division 2 Nuclear Systems Protection System (NSPS) [JG] inverter [INVT] transferred to its alternate power source. This was a known deficiency, and an operability determination previously performed for the inverter was verified to be valid for this event. The inverter was restored to its normal power source by 0100 hours.

An investigation following the event identified that the 'B' TDRFP did not respond to the lowered demand signal due to a lock-up of the TDRFP from mechanical binding at the limit switch mounting unit, a bronze/brass metal guide that slides on a steel rod. The guide is linked to a hydraulic operating cylinder rod to detect the operating cylinder's position for interlock purposes. Mechanical binding occurred as a result of two contributors: mechanical wear between the guide and the rod, and the extended travel distance of the guide on the rod due to EPU testing that required additional opening of the low-pressure control valve relative to its prior full-power rating requirement. The guide of the limit switch mechanism did not move freely once it reached the new position on the rod, where previous contact and wear had never occurred.

A review of this event identified that the RPV water level at which the operator would insert a reactor scram was established at 48 inches and rising. When RPV water level began to rise more than expected, the Control Room Supervisor (CRS) moved up to the panel and focused on the water level indication. As the 'A' reactor operator was announcing the RPV water level status and his intention to move the reactor mode switch to the shutdown position, the CRS spoke to and distracted the operator causing the operator to hesitate during movement of the reactor mode switch. This hesitation resulted in an unplanned automatic reactor scram instead of a manual reactor scram. Immediate action was taken to hold tailgates with Senior Reactor Operators regarding expectations for initiating a reactor scram when testing limits are reached and for the CRS role in maintaining oversight. In addition, this event was included in the Just-In-Time (JIT) training performed prior to restarting the EPU testing.

Condition Report (CR) 107813 was initiated to track the investigation and resolution of this event.

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.

CAUSE OF EVENT

The cause of this event is attributed to an inadequate preventive maintenance program for the Feedwater System [SJ] that did not identify the TDRFP limit switch guide as a critical component and establish appropriate preventive maintenance to prevent its failure.

Contributing to the cause was personnel involved in the replacement of a failed horizontal linkage rod on the 'A' TDRFP in December 2000 (a part similar to the 'B' TDRFP part) did not have a "questioning attitude".

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

On December 21, 2000 a broken horizontal linkage rod that connects the indicator rod to the cylinder rod was replaced on the 'A' TDRFP. The linkage rod broke at the threaded end that engaged the cylinder rod. At that time, this critical component was considered to have failed at the weakest point where the rod was turned down and threaded. Due to the lack of a "questioning attitude," no Condition Report was initiated, investigation conducted, or extent of condition review performed of the failure. These actions could have identified inadequacies in the preventive maintenance program for the Feedwater System which would have resulted in establishing a task for this critical component to periodically inspect, clean, and/or lubricate the Feedwater level control limit switch mounting unit. These actions would have resulted in inspection of the 'B' TDRFP for a similar issue.

SAFETY ANALYSIS

This event is reportable under the provision of 10CFR50.73(a)(2)(iv)(A) due to the automatic reactor scram on high reactor vessel water level.

The plant response and behavior during this event were compared to the Feedwater Level Controller Failure - Maximum Demand discussed in Chapter 15 of the Updated Safety Analysis Report and the Feedwater Control System Failure - Flow Increase discussed in the General Electric Transient Safety Analysis Report and were determined to be within those analyses. This event posed no challenges to fission product barriers.

No safety system functional failures occurred during this event.

CORRECTIVE ACTION

Immediate corrective action included cleaning and lubricating the limit switch guide mechanism on the 'B' TDRFP to prevent mechanical binding, and inspecting the 'A' TDRFP to determine the need for cleaning and lubricating its limit switch guide mechanism to prevent mechanical binding.

Corrective actions to prevent recurrence of this event include: a preventive maintenance task is being developed to periodically inspect, clean and lubricate the limit switch guide mechanism on the TDRFPs; and the horizontal linkage rod was replaced on the 'B' TDRFP. Additional corrective action includes: a review of the preventive maintenance program for the Feedwater System will be performed to assure critical components have adequate preventive maintenance tasks assigned.

PREVIOUS OCCURRENCES

LER Number   Title

None known

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

COMPONENT FAILURE DATA

Manufacturer	Nomenclature	Model Number	Manufacturer Part Number
General Electric (GE) Steam Turbine Division	Rod & Block for Limit Switch Mounting Unit	N/A	Mechanism shown on GE Drawing 509E186AC, Sheet 1 of 2, items 5 & 6