



Nuclear

Clinton Power Station
8401 Power Road
Clinton, IL 61727-9351

10 CFR 50.73
SRRS 5A.108

U-604011

April, 6, 2011

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

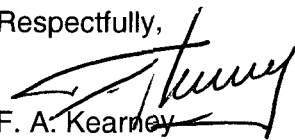
Subject: Licensee Event Report 2011-001-00

Enclosed is Licensee Event Report (LER) No. 2011-001-00: Postulated Spurious High Pressure Core Spray Initiation Result Unanalyzed. This report is being submitted in accordance with the requirements of 10 CFR 50.73.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this report, please contact A. Khanifar, at (217)-937-3800.

Respectfully,


F. A. Kearney
Site Vice President
Clinton Power Station

RSF/blf

Enclosures: Licensee Event Report 2011-001-00

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – IEMA Division of Nuclear Safety



LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@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 an 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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4. TITLE
Postulated Spurious High Pressure Core Spray Initiation Result Unanalyzed

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	08	11	2011	001 - 00		04	06	2011		05000
									FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE 1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: <i>(Check all that apply)</i>									
10. POWER LEVEL 96.9	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)						
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input checked="" type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)						
<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)							
<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER							
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A							

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME A. Khanifar, Site Engineering Director	TELEPHONE NUMBER <i>(Include Area Code)</i> (217) 937-3800
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13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

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

14. SUPPLEMENTAL REPORT EXPECTED <input type="checkbox"/> YES <i>(If yes, complete 15. EXPECTED SUBMISSION DATE)</i> <input checked="" type="checkbox"/> NO	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On 2/8/11, Engineering reviewed the current off-normal procedural methodology for closing the High Pressure Core Spray (HPCS) injection valve in lieu of opening the pump breaker. The review identified the HPCS system could be postulated to spuriously initiate due to concurrent fire induced hot short cable damage to the two automatic initiation logic instrument cables routed in the same raceway. In this event, the reactor pressure vessel (RPV) would fill and flood the Main Steam Lines because shorting would prevent HPCS pump shutdown and shutting the injection valve. In response to the continuing HPCS injection, Main Steam Safety Relief Valves (SRVs) would lift after RPV pressure reaches the SRV setpoint and RPV water would be discharged through the SRVs and downcomers to the suppression pool. This would result in high pressure, high temperature water discharge through SRVs, which would flash to two phase flow. SRVs and associated downcomers have not been analyzed in the current design basis for stresses expected due to two-phase flow at high pressure. The cause of this event is failure to follow a procedure that required engineering review of the revised off-normal procedure. Corrective actions include briefing procedure writers on procedure requirements, reviewing selected procedures, and completing a transient and stress analysis of high pressure two-phase SRV discharge flow.

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor, 3473 Megawatts Thermal Rated Core Power

Energy Industry Identification System (EIS) codes are identified in the text as [XX].

EVENT IDENTIFICATION

Postulated Spurious High Pressure Core Spray Initiation Result Unanalyzed

A. CONDITION PRIOR TO EVENT

Unit: 1 Event Date: 2/8/11 Event Time: 0800 hours CST
 Reactor Mode: 1 Mode Name: Power Operation Power Level: 96.9 percent

B. DESCRIPTION OF EVENT

On 11/18/10, a self assessment performed in preparation for a NRC inspection identified that a manual action in off-normal procedure CPS 4003.01, "Remote Shutdown," was not consistent with the station's 10 CFR 50, Appendix R Safe Shutdown Manual Action Feasibility Study. The calculation for the study states that the High Pressure Core Spray (HPCS) [BG] Pump [P] is tripped in response to a fire in the Main Control Room (fire zone CB-6) [NA], by depressing the internal breaker [BKR] (4 kilovolts) trip plunger at the local panel [PL] to avoid a Reactor Pressure Vessel [RPV] overfill event. This action was implemented in CPS 4003.01 until 1/22/08 at which time the procedure was revised to locally close HPCS injection valve [INV] 1E22-F004 in lieu of tripping the breaker. The procedure revision was initiated due to personnel safety concerns of arc flashing while operating the plunger. However, this change in the manual action was not evaluated and/or reconciled within the Fire Protection Program in accordance with the station's Appendix R Safe Shutdown Manual Action Feasibility Study.

On 2/8/11, at 0800 hours, during Engineering's review of the current methodology to close the HPCS injection valve rather than open the breaker, it was determined that the entire HPCS system could conservatively be postulated to spuriously initiate due to concurrent fire induced hot short cable [CBL] damage to the two automatic initiation logic instrument cables routed in the same raceway in Fire Zone CB-6. In this event, the RPV would fill and flood the Main Steam [SB] Lines (MSLs) because the shorting would prevent HPCS pump shutdown and shutting the injection valve to terminate HPCS flow into the RPV. In response to the continuing HPCS injection, Main Steam Safety Relief Valves (SRVs) [RV] would lift after RPV pressure reaches the SRV setpoint and RPV water would be discharged through the SRVs and downcomers to the suppression pool. This would result in high pressure, high temperature water discharge through SRVs, which would flash to two phase flow. SRVs and their associated downcomers have not been analyzed in the current design basis for the stresses expected due to this two phase flow. This analysis gap has potential impact on the quencher and the pressure suppression function of the pool should the two phase flow loads cause quencher failure. This condition is an analytical nonconformance due to a missing analysis rather than a physically degraded component. This scenario could occur due to fire with the unit being controlled from either the Main Control Room or the Remote Shutdown Panel (RSP).

This scenario is very similar to the event postulated in NRC Unresolved Item (URI) 2005006-01 which discussed fire induced electrical faults in the control cables and control logic of the HPCS pump and injection valve from a fire in the Division 3 switchgear [SWGR] room (Fire Zone CB-5a) which could result in spurious

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actuation of the HPCS pump and reactor core injection. In this scenario, the faults would also impair the capability to shut off the HPCS pump and stop it from injecting water into the core.

Engineering completed an evaluation to address this event and determine additional actions to be taken. The evaluation reviewed the adequacy of the proposed alternative compensatory measures, as compared to the existing required alternate compensatory measures developed for Multiple Spurious Operations (MSO) (required shiftly Operator rounds, the shiftly transient combustible surveillance, Fire Marshal tours, administrative controls on combustible material and Operations fire brigade training). The evaluation demonstrated that the alternative compensatory measures would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire and that compliance with 10 CFR 50, Appendix A, General Design Criterion III, and 10 CFR 50.48(a), "Fire Protection," is met.

The station implemented an alternative compensatory action by adding Fire Zone CB-5a as a line item in eSOMS for Operator rounds.

No inoperable equipment or components directly affected this event.

Issue Report 1172335 was initiated to investigate this event and initiate corrective actions.

The NRC Operations Center was notified about this condition via Event Notification number 46603 on 2/8/11 at 1406 hours EST.

C. CAUSE OF EVENT

The cause of this event is attributed to a failure to follow procedure AD-CL-101-1004, "CPS SPECIFIC 'xxxx.xx' NUMBERED PROCEDURES WRITERS GUIDE." The writer's guide establishes performance of disciplinary reviews for specifically coded procedures to ensure the procedures are technically and functionally accurate for all functional areas. Procedure CPS 4003.01 is a procedure that specifically requires an engineering review of changes for impacts. Operations procedure writers did not ensure an engineering review of changes to procedure CPS 4003.01.

D. SAFETY CONSEQUENCES

This event is reportable under the provisions of 10 CFR 50.73(a)(2)(ii)(B) as a condition that resulted in the nuclear power plant being in an unanalyzed condition that degraded plant safety.

The risk significance of an overfill event is low. An overfill event represents a deviation from the intended shutdown strategy and reactor response, but does not result in any immediate core cooling challenges. The steam lines downstream of the SRV discharge flange are not part of the reactor coolant pressure boundary. Damage to this piping inside the drywell would pressurize the drywell with flashing steam. This steam pressure would act through the drywell Loss of Coolant Accident (LOCA) vents rather than the SRV quenchers and remain bounded by LOCA drywell/containment performance. It would not impact safe shutdown per 10 CFR 50, Appendix R. The continued operation of HPCS will ensure adequate RPV water inventory to maintain adequate fuel temperatures to protect the fuel cladding, and plant capability to operate multiple SRVs will control pressure and dedicated safe shutdown equipment can be controlled from either the Control Room or RSP for a fire in any plant area. Review of calculations related to the quenchers identified that sufficient margin is available to provide assurance that the quenchers will not fail even with LOCA and earthquake loads being considered.

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E. CORRECTIVE ACTIONS

Operations procedure writers and supervisor have been briefed on the requirements of procedure AD-CL-101-1004, and the need to formally request review by other departments, as required.

Operations will validate that procedures related to arc flash concerns have had the cross-disciplinary reviews performed for procedure technical and functional accuracy.

Operations will review a sample of procedures with certain Class Codes related to Engineering, Environmental and In-Service Inspection procedure revisions for the past five years to ensure the cross-disciplinary reviews were performed to ensure the procedure was technically and functionally accurate, if necessary.

A transient and stress analysis of high pressure, high temperature, two phase SRV discharge flow will be completed.

F. PREVIOUS OCCURRENCES

The condition discussed in this report is similar to NRC URI 2005006-01 identified 6/30/2005. This document discusses postulated fire-induced electrical faults in the control cables and control logic of the HPCS pump and discharge valve from a fire in the Division 3 switchgear room which could result in spurious actuation of the HPCS pump and core injection. These faults could impair the capability to shut off the pump and stop it from continually injecting into the core.

In 2005, the ability to trip the HPCS breaker was still an action in procedure CPS 4003.01, and the ability to shutoff the HPCS pump before it could lead to an overflow event that would lead to SRVs opening as designed to relieve pressure, with water exiting a fully flooded reactor pressure vessel through open SRVs, through the downcomers to the suppression pool was considered achievable by Engineering and Operations.

In the July, 2008 CPS NRC Triennial Fire Protection Inspection (2008006), the inspectors followed-up on the URI, but again concluded that no action was required on the part of the licensee, and that the issue required resolution by NRC Office of Nuclear Reactor Regulation.

G. COMPONENT FAILURE DATA

None