



An Illinois Company

Illinois Power Company
P.O. Box 678
Clinton, IL 61727
Tel 217 935-8881 x3900
Fax 217 935-4632

John P. McElwain
Chief Nuclear Officer

U-603169

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Docket No. 50-461

10CFR50.73

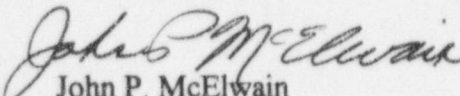
Document Control Desk
Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 1998-034-01

Dear Madam or Sir:

Enclosed is Licensee Event Report (LER) No. 1998-034-01: Inappropriate Assumption During Performance of Design Change Causes One Train of Suppression Pool Cooling Mode of Residual Heat Removal System to be Outside of Its Design Basis. This report is being submitted in accordance with the requirements of 10CFR50.73. This revision updates the cause and corrective actions taken for this event.

Sincerely yours,


John P. McElwain
Chief Nuclear Officer

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Enclosure

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-690
Acting Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety
INPO Records Center 70070

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LICENSEE EVENT REPORT (LER)

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digits/characters for each block)

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DOCKET NUMBER (2)

05000461

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TITLE (4)

Inappropriate Assumption During Performance of Design Change Causes One Train of Suppression Pool Mode of Residual Heat Removal System to be Outside of Its Design Basis

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 NAME	DOCKET NUMBER
10	02	1998	1998	- 034	- 01	03	10	1999	None	05000
									FACILITY NAME	DOCKET NUMBER
									None	05000
OPERATING MODE (9)		4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)							
			20.2201(b)			20.2203(a)(2)(v)			50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)		000	20.2203(a)(1)			20.2203(a)(3)(i)		X	50.73(a)(2)(ii)	50.73(a)(2)(x)
			20.2203(a)(2)(i)			20.2203(a)(3)(iii)			50.73(a)(2)(iii)	73.71
			20.2203(a)(2)(ii)			20.2203(a)(4)			50.73(a)(2)(iv)	OTHER
			20.2203(a)(2)(iii)			50.36(c)(1)			50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)			50.36(c)(2)			50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME

M. J. Walther, Engineer, Nuclear Station Engineering Department

TELEPHONE NUMBER (Include Area Code)

(217) 935-8881, Extension 4024

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE).		X	NO	EXPECTED	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 22, 1997, while operating the Residual Heat Removal system, Operations personnel discovered that motor operated valve 1E12-F024B would not operate when the control room handswitch was manipulated to open the valve. Review of the electrical drawings for the system revealed that valve 1E12-F024B will not open when the circuit breaker for valve 1E12-F006B is open. The circuit breaker for valve 1E12-F006B was open as part of a design change installed in July 1997 to address a 10CFR50 Appendix R concern. Because valve 1E12-F024B could not be operated remotely from the control room as required, the plant was in a condition outside of its design basis. The cause of this event was determined to be human error. The engineers involved with the design change made an inappropriate assumption based on previous experience of the interlock configuration. Corrective actions for this event include: installing a design change to allow the circuit breaker for valve 1E12-F006B to be open without affecting the operation of valve 1E12-FC24B and presenting a seminar to engineering personnel on this event.

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

DESCRIPTION OF EVENT

On December 22, 1997, Operations personnel were operating the Residual Heat Removal (RHR) [BO] system "B" to lower level in the suppression pool. During this activity, motor operated valve [20] 1E12-F024B (RHR discharge to suppression pool) would not open when Operations personnel attempted to open the valve in accordance with Clinton Power Station procedure 3312.01 "RESIDUAL HEAT REMOVAL (RHR)". Review of the electrical drawings for the valve indicated that 1E12-F024B will not open when the circuit breaker for motor operated valve [20] 1E12-F006B (RHR shutdown cooling injection valve) is open. This is due to the use of an interposing relay (powered from the 1E12-F006B electrical breaker) that provides a logic permissive signal to allow opening valve 1E12-F024B when 1E12-F006B is closed. Review of the configuration of the plant at the time of this event disclosed that the circuit breaker for 1E12-F006B was open. The interposing relay was installed in the plant in order to prevent inadvertent reactor vessel draining during plant shutdown. This is accomplished by not allowing valve 1E12-F024B to open unless 1E12-F006B is fully closed. The design change required circuit breaker for 1E12-F006B to be opened (as specified in plant operating procedure CPS No. 3312.01). In this configuration local manual action is required to place the "B" train of the RHR system in the suppression pool cooling mode of operation.

The circuit breaker for 1E12-F006B was opened as part of a design change to resolve concerns related to 10CFR50 Appendix R compliance. The specific concern that the design change addressed was that a hot short in the control cables for valve 1E12-F006B could cause it to go open during a fire. If valve 1E12-F006B is open, then valve 1E12-F024B would not open from the main control room because of the interlock installed to prevent inadvertent reactor vessel drain down. Method two of the safe shutdown analysis requires 1E12-F024B to open from the closed position. Because the hot short in the control circuitry would not allow 1E12-F024B to open remotely from the control room, it would have to be opened manually. However, eight hour emergency minimum battery [BTRY] supplied lighting [FH] was not provided near the valve or its egress path as required by 10CFR50 Appendix R for manual operation. This condition was previously reported in Licensee Event Report 97-016.

In order to address the 10CFR50 Appendix R concern, a design change was prepared (Engineering Change Notice 30211) to require that the circuit breaker for 1E12-F006B be open in Modes 1, 2, and 3 with reactor pressure greater than 97 pounds per square inch gauge (PSIG). The design change included installation of an annunciator bypass to allow the associated loss of power annunciator in the control room to be bypassed when the circuit breaker for 1E12-F006B was open. The focus of the engineers developing and reviewing the design change was the electrical circuitry necessary for the annunciator bypass switch. The engineers involved in the design change were aware that an interlock for the motor operated valve existed. The engineers assumed the configuration of the interlock was different than the actual design configuration based on previous experience with valve interlocks. The engineers believed that operation of the interlocked valve would not be affected by opening the circuit breaker for 1E12-F006B based on this past experience.

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

This design change was released for operation on July 16, 1997. During the period this design change was installed, suppression pool cooling was not required to be operable by the plant Technical Specifications or 10CFR50 Appendix R, as the plant was already in Mode 4 (Cold Shutdown). During the design change process neither the design engineer or reviewers identified that opening the circuit breaker for 1E12-F006B would prevent 1E12-F024B from opening.

On October 2, 1998, plant personnel determined that this event was reportable as operation of the plant outside its design basis. At that time the plant was in cold shutdown (Mode 4), reactor [RCT] temperature was being maintained between 95 and 115 degrees Fahrenheit, and reactor pressure was atmospheric.

No automatic or manually initiate safety system responses were necessary to place the plant in a safe and stable condition. No equipment or components were inoperable at the start of this event to the extent that their inoperable condition contributed to this event.

CAUSE OF THE EVENT

The cause of this event was determined to be human error. The engineers involved with the design change made an inappropriate assumption based on previous experience of the interlock configuration. This resulted in the failure to successfully implement the design change control procedures.

CORRECTIVE ACTIONS

A design change to allow the circuit breaker for valve 1E12-F006B to be open without affecting the operation of valve 1E12-F024B was installed so that the design basis requirements could be restored.

Training Seminar XZ61687, which discusses this event in detail, was prepared and presented to engineering personnel. Also, the engineering department conducted briefings that emphasized the need for strict procedural compliance and the need to broaden the thought process beyond the task at hand and consider post change operational impacts during normal, abnormal, and accident/post-accident conditions.

In order to establish the extent of condition for this event, a review of other selected design changes was conducted. The review did not identify any other design changes in the reviewed population that resulted in placing the plant outside of its design bases and supporting analysis. The selected design changes reviewed were selected based on the following criteria: individuals involved in the same role in performing the design change that caused this event, design changes approved during periods when plant startup was believed to be imminent, and design changes with potential functional application (could affect operation of equipment). This review encompassed eleven design change packages and over forty design change documents. The results of the review of selected design packages disclosed that there were no significant functional issues with the design change packages examined during this assessment.

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ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73 (a)(2)(ii)(B) as a condition that resulted in the plant operating outside its design basis. The opening of 1E12-F006B as part of Engineering Change Notice (ECN) 30211 placed the plant outside its design basis by requiring local operator action to open valve 1E12-F024B for method two of the safe shutdown analysis.

During the time period that ECN 30211 was installed, the plant was in Mode 4 (Cold Shutdown). In Mode 4, the Plant Technical Specifications do not require suppression pool cooling to be operable. The redundant RHR "A" train was not affected by this condition. Had the plant been in a mode that required suppression pool cooling to be operable, local action operators would have been required to manually open 1E12-F024B to place the RHR system "B" train in suppression pool cooling. This action, if necessary under design basis accident loss of coolant accident conditions, would have required either operating 1E12-F024B manually or closing the circuit breaker for 1E12-F006B. This action could have been precluded or delayed based on in-plant radiation levels. The analysis for the design basis accident assumes that suppression pool cooling is started about thirty minutes after the start of the accident.

This condition was originally determined to not be reportable per 10CFR50.73(a)(2)(ii)(B) (outside of design basis). This determination was made based on the information that, since installation of the modification the plant had not operated in a mode which required this design feature to be operable. However, after discussion with the NRC on October 2, 1997, Illinois Power determined that application of this reporting criteria is not plant operating mode dependent and therefore, this condition is reportable per 10CFR50.73(a)(2)(ii)(B).

This condition had existed from July 16, 1997, when ECN 30211 was installed in the plant and was corrected when modification RH-048 was installed and released for operation on December 16, 1998.

ADDITIONAL INFORMATION

No equipment or components failed during this event.

Review of Licensee Event Reports submitted in the last two years did not reveal other instances of an inappropriate assumption in recently implemented design changes causing the plant to be outside of its design basis.

For further information on this event contact Mike Walther at (217) 935-8881 extension 4024.