PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

APR 18 1996

LR-N96109

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

Dear Sir:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
UNIT NO. 1
LICENSEE EVENT REPORT NO. 96-010-00

This Licensee Event Report entitled "Engineered Safety Feature Actuation - Reactor Core Isolation Cooling System Isolation" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(iv).

Sincerely,

Mark E. Reddemann
General Manager -
Hope Creek Operations

JPP
SORC Mtg. 96-046

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LER File

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NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)

Engineered Safety Feature Actuation - Reactor Core Isolation Cooling System Isolation

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
										05000
3	19	96	96	-- 010 --	00	4	18	96	FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	2	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
POWER LEVEL (10)	1%		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	x	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	TELEPHONE NUMBER (Include Area Code)
James Priest, Licensing and Regulation	(609) 339-5434

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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)

On March 19, 1996, the Hope Creek Generating Station was preparing to enter Operational Condition 1, POWER OPERATION, following the sixth refueling outage. During warm-up of the Reactor Core Isolation Cooling (RCIC) system, an automatic isolation signal was received from a high steam flow signal at 0430 hours. The RCIC system was not required to be operable at the time the isolation signal was received since reactor pressure was approximately 137 psig (the Technical Specifications require RCIC to be operable with reactor pressure above 150 psig). Operators determined that the isolation signal resulted from condensate flashing in the RCIC steam lines during the warm-up procedure. The operators re-set the isolation signal, completed the warm-up of the RCIC steam lines and successfully placed RCIC into service. At 0653 hours, a four hour report was made to the NRC for this Engineered Safety Feature (ESF) actuation in accordance with the requirements of 10CFR50.72(b)(2)(ii). The apparent cause of the event was deficient RCIC warm-up procedures that caused high steam flow conditions. Corrective action includes a revision to the procedure to eliminate the conditions causing the high flow signal.

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

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)

Reactor Core Isolation Cooling (RCIC) System - EIIS Identifier {BN}

IDENTIFICATION OF OCCURRENCE

Event date:	3/19/96
Discovery date:	3/19/96
Date determined to be reportable:	3/19/96

Problem Report 960319086

This event is reportable under the provisions of 10CFR50.73(a)(2)(iv).

CONDITIONS PRIOR TO OCCURRENCE

Plant was in OPERATIONAL CONDITION 2 (STARTUP).
Reactor was at 1% of rated thermal power.

DESCRIPTION OF OCCURRENCE

On March 19, 1996, the Hope Creek Generating Station was preparing to enter Operational Condition 1, POWER OPERATION, following the sixth refueling outage. In order to place the RCIC system into service, operators initiated a warm-up of the RCIC steam line in accordance with procedure HC.OP-SO.BD-0001(Q), RCIC System Operations. Shortly after a reactor operator cracked open the RCIC steam line warm-up valve, FC-HV-F076, an automatic isolation signal was received due to a sensed high steam flow condition at 0430 hours. The FC-HV-F076 valve closed as a result of the isolation signal. The RCIC system was not required to be operable at the time of the isolation signal since reactor pressure was approximately 137 psig (the Technical Specifications require RCIC to be operable with reactor pressure above 150 psig). Operators immediately initiated an investigation of the RCIC isolation signal and determined that a steam line break did not occur. This initial investigation determined that the isolation signal resulted from condensate flashing in the RCIC steam lines during the warm-up procedure. The operators reset the RCIC system isolation signal, completed the warm-up of the RCIC steam lines and successfully placed RCIC into standby service. At 0653 hours, a four hour report was made to the NRC for this Engineered Safety Feature (ESF) actuation in accordance with the requirements of 10CFR50.72(b)(2)(ii).

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ANALYSIS OF OCCURRENCE

The RCIC system is designed to ensure adequate core cooling following a reactor main steam isolation and subsequent loss of feedwater flow. RCIC is used to control reactor pressure, temperature and water level. The RCIC system steam line contains an inboard containment isolation valve (FC-HV-F007) and an outboard containment isolation valve (FC-HV-F008). A warm-up line around the inboard isolation valve contains a one inch containment isolation valve (FC-HV-F076). During the RCIC system warm-up, all manual steam drains were closed prior to throttling open the FC-HV-F076 valve to admit steam into the line. The drain valves remained closed while steam was admitted through the warm-up line and the condensed steam was drained through the normal steam trap bypass line to the main condenser. Prior to the warm-up, the RCIC steam supply line was drained and pressurized to protect structural components of the RCIC system by minimizing the possibility of a water hammer.

The RCIC system is designed to automatically isolate upon various indications of a steam line break, including a sensed high steam flow condition. The inlet steam line flow is monitored by two redundant differential pressure switches, which generate an isolation signal on two different logic channels. The high flow condition is sensed by a flow nozzle upstream of the steam line isolation valves and is set at a differential pressure equal to 300% of rated steam flow with a three to thirteen second time delay. Isolation logic channel "D", when actuated, closes the FC-HV-F007, FC-HV-F076 and the RCIC turbine trip throttle valves.

During this event, the FC-HV-F076 was partially opened. Approximately one minute later, an isolation signal was received and the FC-HV-F076 valve closed. In accordance with procedure, the FC-HV-F007 valve was already closed and the FC-HV-F008 valve remained open since it does not isolate on a "D" logic channel signal.

Follow-up investigation of the event revealed a fundamental difference in the methods used to warm-up the High Pressure Coolant Injection (HPCI) and RCIC systems. When performing a warm-up of the HPCI system, the manual drain valves remain open after steam is admitted to the HPCI steam line. The procedure ensures that all condensed steam is drained and also prevents a vacuum from being drawn in the HPCI steam line prior to steam admission. When the HPCI steam line pressure reaches 100 psig, the manual drain valves are closed. When performing a warm-up of the RCIC system, the manual drain valves are closed prior to admitting steam into the RCIC steam line. The condensed steam is drained through the normal steam trap bypass line to the main condenser. When the RCIC manual drain valves are closed, there is a potential for the RCIC steam line pressure, downstream of the isolation valves, to drop to main condenser vacuum pressure (approximately 5" Hg absolute). When the FC-HV-F076 valve is throttled open during the RCIC warm-up, the main condenser vacuum could cause upstream RCIC steam line pressure

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ANALYSIS OF OCCURRENCE (Continued)

to drop, causing a high flow condition. This has been determined to be the apparent cause for the RCIC system isolation; however, the conditions which caused this isolation could not be positively confirmed since instrumentation is not normally used to record these parameters during the RCIC warm-up.

Since automatic isolation of the RCIC system due to a high steam line flow signal is designated as an ESF actuation, this event is reportable under the provisions of 10CFR50.73(a)(2)(iv).

APPARENT CAUSE OF OCCURRENCE

As discussed in the previous section, the apparent cause of the event was a deficient RCIC warm-up procedure, which resulted in actuation of the RCIC steam line high flow logic.

ASSESSMENT OF SAFETY CONSEQUENCES

The RCIC system is required to be operable by the Technical Specifications when reactor pressure is above 150 psig. When the RCIC isolation signal was generated during system warm-up, the reactor pressure was approximately 137 psig. Although the isolation signal represented an unnecessary challenge to an ESF system, the status of RCIC (in terms of Technical Specification required operability) was not affected. Therefore, there were no adverse potential safety consequences associated with this event.

PREVIOUS OCCURRENCES

LER 91-020-00 documented an occurrence where the High Pressure Coolant Injection (HPCI) system received an isolation signal when the HPCI steam line warm-up was in progress. The cause of that event was attributed to mis-wiring of the NUMAC module for the "A" channel isolation system. The corrective actions for that event would not have prevented the RCIC isolation.

CORRECTIVE ACTIONS

On 3/19/96, operators initiated an investigation of the RCIC isolation signal and determined that a steam line break did not occur. The isolation signal was re-set, the warm-up of the RCIC steam lines was completed and the RCIC system was successfully placed into standby service.

The RCIC warm-up procedure was revised to eliminate the aforementioned RCIC steam line conditions which could cause a high flow signal.