PSEG

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Nuclear Business Unit

FEB 27 1996

LR-N96058

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 96-005-00

This Licensee Event Report entitled "Inadequate Surveillance Testing for RHR Suppression Pool and Spray Modes Due to Unaccounted RHR Heat Exchanger Bypass Valve Leakage" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i).

Sincerely,

M. E. Reddemann
General Manager -
Hope Creek Operations

JWK/tcp

Attachment
SORC Meeting 96-023
c Distribution

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

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

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)		
Hope Creek Generating Station	05000354	1 OF 5

TITLE (4)

Inadequate Surveillance Testing for RHR Suppression Pool Cooling and Spray Modes Due to Unaccounted RHR Heat Exchanger Bypass Valve Leakage.

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
01	31	96	96	-- 005	-- 00	02	27	96		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	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)		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)
John Karrick, Hope Creek LER Coordinator	609-339-5298

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)

A review of recently performed Residual Heat Removal pump surveillance data at the Hope Creek Generating Station revealed that the surveillance requirements of Technical Specifications (TS) 4.6.2.2.b and 4.6.2.3.b have not been satisfied during past performances. These surveillance test requirements each specify a flow rate "through" the Residual Heat Removal (RHR) heat exchanger for verifying operability of the suppression chamber spray and suppression chamber cooling modes of RHR. RHR heat exchanger bypass valve leakage was not accounted for when crediting previous surveillance tests in that the recorded flow rates were total system flow. This represents a condition prohibited by the plant's Technical Specifications and is being reported pursuant to 10CFR50.73(a)(2)(i)(b).

The root cause of this event was a lack of rigorous application of engineering principles and design review when developing the TS. A contributor to the event was a missed opportunity to incorporate prior operating experience information. Corrective actions include a revision to the TS surveillance requirements, determinations of bypass valve leakage quantity, and changes to the Operating Experience process. There was minimal impact on plant safety as a result of this event.

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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)
Residual Heat Removal (RHR)-EIIS Identifier: {BO}

IDENTIFICATION OF OCCURRENCE:

Discovery Date: January 30, 1996
Report Date: February 28, 1996

CONDITIONS PRIOR TO OCCURRENCE:

The plant was in Operational Condition 5, Refueling, at the time of discovery.

The B loop of RHR Shutdown Cooling was in service at the time. There were no systems, structures, or components that were known to be inoperable at the start of the event that contributed to the event.

DESCRIPTION OF EVENT:

On January 30, 1996, it was determined that the surveillance requirements for Technical Specifications (TS) 4.6.2.2.b and 4.6.2.3.b have not been met in the past. The surveillance for 4.6.2.2.b states, "The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE: By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and the suppression pool spray sparger when tested pursuant to Specification 4.0.5". The surveillance for 4.6.2.3.b states, "The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE: By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5".

The RHR system design includes the application of an 18" Fisher Type 7620A Butterfly valve without a seat in the heat exchanger bypass lines. The designed leakage for these valves per vendor specifications is 100 gpm minimum. The bypass valves automatically open on a Low Pressure Coolant Injection (LPCI) signal to provide a direct injection path to the vessel, control plant cool down rates through manual operation, and provide maximum heat exchanger cooling when closed.

Previous surveillance testing performed to comply with the above specifications did not account for the minimum design leakage nor the actual leakage through the bypass valves. Actual bypass valve leakage was determined to be 203 (+/- 24) gpm and 230 (+/- 32) gpm for the A and B RHR loops respectively. Additionally, flow tests performed on the heat exchangers indicated an actual flow of 9968 (+/-199.6) gpm and 9648 (+/-194.8) through the A and B RHR heat exchangers respectively.

The surveillance test procedure for 4.6.2.3.b only requires RHR pump flow to be greater than or equal to 10,000 gpm which has resulted in instances which the recorded flow readings for the suppression pool cooling test were at or only slightly above the required 10,000 gpm. Given the actual bypass valve leakage rates and the heat exchanger flow tests' results, the actual flow through the RHR heat exchangers was less than the required 10,000 gpm for those tests and as such represents inadequate surveillance testing and a condition prohibited by Technical Specifications.

The surveillance test procedure for TS 4.6.2.2.b, suppression pool spray mode operability, has the operator first establish an overall system flow of greater than 10,000 gpm. Flow through the suppression pool spray line is then throttled open to a value greater than or equal to 500 gpm as indicated by instrumentation on the spray flow line. Typical recorded flow rates from previous tests were 500-700 gpm, however it cannot be assured that the recorded flow rates were indicative of flow that had actually passed through the RHR heat exchanger. This also represents inadequate surveillance testing and a condition prohibited by Technical Specifications.

ANALYSIS OF OCCURRENCE:

The flow rates recorded for the surveillance tests in question have been measured by flow instrumentation that is located in the common piping downstream of the bypass valve and heat exchanger. Therefore, the indicated flow rates included flow from both the heat exchanger outlet and the bypass valve leakage.

The design basis heat transfer requirement for the RHR heat exchangers has been calculated and is equivalent to a heat exchanger flow rate of greater than or equal to 8,985 gpm for the suppression pool cooling mode of operation. This was derived assuming post-LOCA conditions with a suppression pool temperature of 212 degrees F., which is the most limiting scenario applicable to this event. The worst case actual heat exchanger flow based on the flow test data is above the minimum design basis value. Consequently, it has been concluded that the safety function associated with the suppression pool cooling mode of RHR was preserved. By maintaining the suppression pool design temperature limits, the requirements for the containment spray modes are bounded.

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The completion of the safety function for the containment spray modes of RHR operation is dependent upon operator action in the event of an accident. Although the previous surveillance tests were inadequate to meet the technical specification requirement, the RHR system's ability to have provided adequate containment cooling and spray flow was maintained.

SAFETY SIGNIFICANCE:

Since the RHR system was capable of completing its safety functions of decay heat removal, suppression pool cooling and containment sprays, the significance of this event was minimal.

APPARENT CAUSE OF OCCURRENCE:

The root cause of this event was the lack of rigorous application of engineering principles and design review when developing the Technical Specifications.

A contributing factor was a missed opportunity within the Operating Experience (OE) Feedback process. The Limerick Generating station reported an identical concern in 1992 and OE 5512 was issued by the utility as a result. The screening of OE 5512 at Hope Creek identified this issue for informational interest to the Operations department rather than directly assigning action. At that time this screening was considered appropriate because of the hierarchy of the OE document received. No formal response was documented nor was one required by the OE Coordinators. As a result, this issue was reviewed but no actions were taken.

PREVIOUS OCCURRENCES:

A review of LERs over the last two years has shown that there have been no previous similar events at the Hope Creek Generating Station.

The valve design application was evaluated for potential generic concerns within other safety related systems. There were no generic concerns discovered as a result of this review.

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

CORRECTIVE ACTIONS:

1. A change to the Technical Specifications has been submitted. This change will allow credit to be taken for the heat exchanger bypass valve leakage for the surveillance requirements of 4.6.2.2.b and 4.6.2.3.b. This change is similar to the approved change that the Limerick Generating Station implemented in response to their event.

2. The flow through the bypass valves was measured using ultrasonic equipment. The as found leak rates were 203 (+/-24) and 230 (+/-32) gpm for the A and B RHR loops respectively.

3. RHR heat exchanger flow tests were performed. The as found flow rates were 9968 (+/- 199.6) and 9643 (+/-194.8) gpm for the A and B RHR heat exchangers respectively.

4. The OEF group Supervisor has discussed the details regarding the screening that was performed for OE 5512 with personnel assigned those duties and used it as an example of the need for maintaining sensitivity to the issue during their reviews. The OE screening process has since been changed to be issue driven rather than driven by hierarchy of the OE document received. Plant Status reports and Operating Experience Summaries (OES) from utilities are assigned action tasks if the item has a potential for significance and is applicable to the Hope Creek Generating Station.

5. A follow up review for the generic implications of heat exchanger or component bypass flow within TS required systems will be incorporated into the existing Technical Specification Surveillance Improvement Program (TSSIP). This review will be completed by December 31, 1996.