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MAY 1 5 2002

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

Gentlemen:

LER 354/02-001-00
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-354

This Licensee Event Report, "'B' Residual Heat Removal Inoperability due to Mispositioned Minimum Flow Manual Isolation Valve", is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73(a)(2)(i)(B).

The attached LER contains no commitments.

Sincerely,

D.F. Garchow

D. F. Garchow
Vice President - Operations

Attachment

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C Distribution
LER File 3.7

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NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
HOPE CREEK GENERATING STATION	05000354	1 OF 4

4. TITLE
"B" Residual Heat Removal Inoperability due to Mispositioned Minimum Flow Manual Isolation Valve

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	20	02	02	– 001	– 00	05	15	02	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

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

	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)		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
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	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)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Robin A. Ritzman, Licensing Engineer	856-339-1445

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

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

On March 20, 2002, the "B" Residual Heat Removal (RHR) minimum flow manual maintenance isolation valve was discovered in the closed position. The valve was discovered closed by two non-licensed operators who were preparing to close and tag the valve in preparation for a planned system outage. The cause of the inoperability of "B" RHR was a mispositioned valve. The cause of the mispositioning could not be identified. The "B" RHR pump has been evaluated for its response to a large break Loss of Coolant Accident (LOCA) and a small break LOCA, and for its ability to provide long-term post event cooling. An engineering evaluation based on industry operating experience determined that the pump would be available for at least 18 minutes without minimum flow protection. This exceeds the amount of time that the "B" RHR pump is required to respond to a LOCA. Based on the above, there was no impact to the health and safety of the public. Corrective actions included position verification of all accessible manual isolation valves, "lessons learned" discussions, and a review of the use of valve line-ups. There are no commitments associated with this LER.

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) {BO/-}
Reactor Core Isolation Cooling (RCIC) {BN/-}
High Pressure Coolant Injection (HPCI) {BJ/-}

Energy Industry Identification System {EIIS} codes and component function identifier codes appear as (SS/CCC)

CONDITIONS PRIOR TO OCCURRENCE

The plant was in OPERATIONAL CONDITION 1 (POWER OPERATION). No other structures, systems or components were inoperable at the start of this event that contributed to the event.

DESCRIPTION OF OCCURRENCE

On March 20, 2002, at approximately 0530, the “B” Residual Heat Removal (RHR) minimum flow manual maintenance isolation valve was discovered in the closed position. This is a normally locked open 4” rising stem gate valve (BO/ISV). The valve was discovered closed by two non-licensed operators who were preparing to close and tag the valve in preparation for a planned system outage. An engineering evaluation determined that “B” RHR was capable of performing its safety function in this configuration.

Technical Specifications state that a “system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s)”. “B” RHR was considered to be INOPERABLE because the minimum flow line was not capable of performing its design function. Its design function is to provide a small flow to the suppression chamber if no pump discharge valve is open, or if reactor vessel pressure is higher than the pump shutoff pressure. The investigation could not determine the length of time that the valve was mispositioned; therefore, it is assumed that the system was inoperable for longer than allowed by Technical Specifications. This LER is being submitted in accordance with 10CFR50.73(a)(2)(i)(B) as a Technical Specification prohibited condition.

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

CAUSE OF OCCURRENCE

The cause of the inoperability of "B" RHR was a mispositioned valve. The cause of the mispositioning could not be identified. System parameter and lineup datasheets, shift logs, pump performance procedures, and plant historian data points were reviewed. Interviews were conducted with individuals associated with the manipulation of the system including operations, Inservice Inspection, and engineering groups. Neither the time nor the reason for the valve mispositioning could be determined. Tampering was considered and ruled out as a cause of this event.

PRIOR SIMILAR OCCURRENCES

A review of LERs for Salem Unit 1, Salem Unit 2, and Hope Creek over the past three years identified two previous reportable occurrences of mispositioned components resulting in a Technical Specification prohibited condition. LER 354/01-004 reported that a reactor building pressure failed to meet the acceptance criteria during the reactor building integrity functional test due to the reactor building differential pressure controllers being set incorrectly during a maintenance or surveillance testing activity. LER 354/00-009 reported the INOPERABILITY of the Filtration, Recirculation, and Ventilation System Recirculation Subsystem caused by and Improperly secured manual damper. The corrective actions taken were specific to the events and systems involved and could not have prevented this event.

SAFETY CONSEQUENCES AND IMPLICATIONS

The "B" RHR pump (BO/P) has been evaluated for its response to a large break Loss of Coolant Accident (LOCA) and a small break LOCA, and for its ability to provide long-term post event cooling. An engineering evaluation based on industry operating experience determined that the pump would be available for at least 18 minutes without minimum flow protection. This exceeds the amount of time that the "B" RHR pump is required to respond to a LOCA. The worst case postulated event is a Small Break LOCA Without High-Pressure Injection and no operator action. In this event, operation of the pump without minimum flow protection would be approximately 12 minutes per facility design. This is based on pump auto-start from a reactor vessel level 1 signal to the time the RHR system injects in the LPCI mode per the UFSAR.

Based on the above, there was no impact to the health and safety of the public. A review of this condition determined that a Safety System Functional Failure (SSFF) has not occurred as defined in Nuclear Energy Institute (NEI) 99-02.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
HOPE CREEK	05000354	02	0 0 1	00	4 OF 4

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

CORRECTIVE ACTIONS:

1. Upon discovery of the event, the operating crew performed an alignment verification for all of the ECCS minimum flow manual isolation valves. This lineup found no valves outside of their required positions.

2. This initial verification was followed by a position verification of all accessible manual isolation valves for the ECCS and important safety systems. This verification of 918 valves yielded no discrepancies.

3. An information packet and speaking guide was provided to the Operating Superintendents to facilitate a “lessons learned” discussion with the crews. It contained information related to the “B” RHR event and additional information and guidance on other recent issues with status control.

4. A review of the use of valve line-ups is underway to determine methods of better assuring correct component status.

COMMITMENTS

The corrective actions cited in this LER are voluntary enhancements and do not constitute commitments.