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U. S. Nuclear Regulatory Commission
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SUPPLEMENTAL LER 354/2005-003-01
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

This Supplemental Licensee Event Report entitled, "Reactor Coolant System Leak from Check Valve Position Indicator," is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i)(A), 10CFR50.73(a)(2)(iv)(A), and 10CFR50.73(a)(2)(ii)(A).

Sincerely,

Michael J. Massaro

Michael J. Massaro
Plant Manager - Hope Creek

Attachment

BJT

C Distribution
LER File 3.7

JE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
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: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Reactor Coolant System Leak from Check Valve Position Indicator

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
06	07	2005	2005	- 003 -	01	09	29	2005	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)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Brian Thomas, Licensing Engineer	856-339-2022

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
B	BO	V	A585	N					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

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

On June 7, 2005, at 1413 hours, the reactor recirculation pumps were taken to minimum speed and the reactor mode switch was locked in shutdown to scram the reactor due to increasing drywell pressure and drywell floor drain leakage. An Unusual Event (UE) was declared at 1437 hours due to drywell floor drain leakage exceeding 10 gpm. The operating crew took action to bring the plant to a cold shutdown condition and a drywell entry was performed. At 0240 hours on June 8, 2005, the source of the leak was determined to be from the F050A residual heat removal (RHR) check valve position indicator. On June 8, 2005, at 0336 hours, the B RHR loop was placed in shutdown cooling and Operational Condition 4 was entered at 0456 hours. The UE was terminated at 0515 hours on June 8, 2005.

The cause of the reactor coolant system leak was the result of an approximately 280 degree circumferential crack in the position indicating tube for the F050A RHR check valve. The through wall leak was caused by the vibration of the attraction sleeve (which is on the end of actuator rod) in the presence of the switch magnetic force resulting in the attraction sleeve fretting/wearing through the position indicating tube. Corrective actions consist of modification of the F050A and F050B RHR check valves, ultrasonic testing of the six other check valves with the same position indicating tubes, development of preventative maintenance activities and an engineering review of components and sub-components communicating with the pressure boundary of the loop 'A' and 'B' reactor recirculation piping.

This event is being reported in accordance with 10CFR50.73(a)(2)(i)(A), "the completion of any nuclear plant shutdown required by the plant's Technical Specifications," 10CFR50.73(a)(2)(iv)(A), "any event or condition that resulted in manual or automatic actuation of ...reactor protection system (RPS) including: reactor scram or reactor trip," and 10CFR50.73(a)(2)(ii)(A), "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded."

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor (BWR/4)

Reactor Coolant System {AB}*
Residual Heat Removal {BO}*

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

IDENTIFICATION OF OCCURRENCE

Event Date: June 7, 2005
Discovery Date: June 7, 2005

CONDITIONS PRIOR TO OCCURRENCE

Hope Creek was in operational condition 1 at 100% power prior to the event. The B residual heat removal (RHR) pump was out of service for maintenance at the start of the event. There was no other equipment out of service that impacted this event.

DESCRIPTION OF OCCURRENCE

On June 7, 2005, at 1351 hours, the operating crew entered the abnormal operating procedure for drywell leakage. At 1411 hours with drywell pressure at 0.18 psig, the drywell leak detection system alarmed. At 1412 hours with drywell pressure at 0.27 psig and continuing to rise, reactor power was reduced to 80% at the direction of the control room supervisor. At 1413 hours with drywell pressure at 0.40 psig, the reactor recirculation pumps were reduced to minimum speed and the reactor mode switch was locked in shutdown to scram the reactor. Technical Specification (TS) 3.4.3.2 was entered for unidentified leakage greater than 5 gpm. TS 3.4.3.2 requires the reduction of leakage to within limits in 4 hours or to be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

At 1421 hours on June 7, 2005 with drywell pressure at 0.47 psig, the operating crew commenced the lowering of reactor pressure to 600 psig. At 1426 hours, drywell pressure was at 0.46 psig and drywell leakage was determined to be greater than 10 gpm. At 1427 hours with reactor vessel pressure control being provided by the electro-hydraulic control (EHC) system, reactor water level 8 was reached. At 1429 hours, drywell floor drain leakage was determined to be approximately 10.5 gpm with drywell pressure at 0.43 psig. At 1430 hours, reactor vessel water level 3 scram level was reached due to the reactor vessel pressure reduction. Reactor vessel pressure was at 600 psig. The operating crew further reduced pressure to 550 psig. During the pressure reduction, reactor vessel water level reached level 8. At 1434 hours with reactor vessel pressure at 550 psig, drywell floor drain leakage was approximately 11.3 gpm and drywell pressure was 0.40 psig. At 1437 hours, an Unusual Event (UE) was declared due to drywell floor drain leakage being greater the 10 gpm. At 1438 hours, the operating crew commenced the cool down of the reactor in accordance with procedures. Reactor vessel level was being maintained at +35". At 1442 hours, drywell floor drain leakage was approximately 14.6 gpm with drywell pressure at 0.37 psig. At 1458 hours, drywell floor drain leakage was approximately 11.2 gpm with reactor pressure at 500 psig. At 1522 hours, drywell floor drain leakage was steady at 9.9 gpm. At 1618 hours, reactor pressure was reduced to 300 psig. At 1651 hours, the operating crew reset the scram signal. The position indication for the F050A residual heat removal (RHR) {BO/V} check valve was lost at 1802 hours. At 1823 hours the B RHR pump was restored from maintenance and retested satisfactorily. At 2344 hours with reactor pressure less than 82 psig, the A RHR train was declared inoperable for shutdown cooling due to questions with the failure of the valve position indication. Preparations were made for drywell entry and placing B RHR in shutdown cooling.

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SAFETY CONSEQUENCES AND IMPLICATIONS

On June 7, 2005, drywell floor drain leakage increased to greater than 10 gpm. As a result an Unusual Event (UE) was declared. The operating crew took appropriate actions to bring the plant to a controlled cold shutdown condition. Reactor water level was maintained above level 2 (initiation signal for emergency core cooling injection) and reactor pressure control was maintained throughout the event. Even if a complete severance of the F050A RHR check valve position indicating tube were to occur, this failure would not result in an uncontrolled reactor depressurization. Based on the above, there was no impact to the health and safety of the public.

A review of this event determined that a Safety System Functional Failure (SSFF) has not occurred as defined in Nuclear Energy Institute (NEI) 99-02. Hope Creek was brought to a cold shutdown condition following the identification of the reactor coolant system leak from the position indicating tube of the F050A RHR check valve.

CORRECTIVE ACTION

1. The F050A and F050B RHR check valves were modified to remove the position indicator tubes. These modifications were completed prior to the plant startup.
2. Six other check valves contain the same position indicator tube as the F050A and F050B RHR check valves. The position indicating tubes for these six check valves underwent ultrasonic testing with no indications of wear as experienced on the F050A RHR check valve. These inspections were completed prior to the plant startup.
3. An engineering review was performed for components and sub-components communicating with the pressure boundary of the loop 'A' and 'B' reactor recirculation piping or connected piping systems that were susceptible to system vibration and fretting/wear vulnerabilities. The identified components were assessed as adequate with respect to the pressure integrity function.

COMMITMENTS

This LER contains no commitments.