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LER 354/2005-002-00
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

This Licensee Event Report entitled, "Through-Wall Leak on 'B' Reactor Recirculation System Decontamination Port," is being submitted pursuant to the requirement of 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

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

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.

LICENSEE EVENT REPORT (LER)

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

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4. TITLE
Through-Wall Leak on 'B' Reactor Recirculation System Decontamination Port

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
03	27	2005	2005	- 002 -	00	05	26	2005	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE
2

10. POWER LEVEL
6

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

- ☐ 20.2201(b)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☒ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☐ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

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
X	AD	PSP	-	Y					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

On March 27, 2005, reactor power was reduced to approximately 6% power (Operational Condition 2) to perform a primary containment entry to identify the source of increasing unidentified reactor coolant system (RCS) leakage. Beginning on February 8, 2005, unidentified RCS leakage had been trending upward and was at approximately 0.73 gpm on March 27, 2005. Hope Creek personnel identified a steam leak coming from the insulated decontamination port that is connected to the 'B' reactor recirculation system between the suction of the pump and the suction isolation valve. At 2329 hours on March 27, 2005, the reactor mode switch was taken to shutdown to perform a controlled shutdown and cooldown of the plant to precisely identify and repair the leakage source. At 1035 hours on March 28, 2005, after removal of the insulation from the decontamination port, the steam leakage was determined to be coming from a crack located on the 4-inch nominal pipe size (NPS) portion of the decontamination port.

The through-wall crack in the decontamination connection was determined to be fatigue initiation and propagation as caused by the following: presence of an original subsurface weld defect and presence of a decontamination port geometry that experienced damaging high cycle vibration during certain recirculation system operating conditions. Corrective actions consisted of the modification of the A and B reactor recirculation decontamination ports and an extent of condition review to determine if other connections to the reactor recirculation system were vulnerable to the decontamination port failure.

This event is being reported in accordance with 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."

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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)

Reactor Coolant System {AB}
Reactor Recirculation System {AD}*

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

IDENTIFICATION OF OCCURRENCE

Event Date: March 27, 2005
Discovery Date: March 27, 2005

CONDITIONS PRIOR TO OCCURRENCE

A voluntary controlled plant power reduction was performed to determine the source of increasing reactor coolant system (RCS) {AB} unidentified leakage. On March 27, 2005, while in Operational Condition 2 with reactor power at approximately 6%, a primary containment entry identified a steam leak from the insulated decontamination port that is sealed by a bolted flange within the isolable boundary of the 'B' recirculation system. Hope Creek then proceeded to cold shutdown to precisely identify and repair the leakage source. With the plant in Operational Condition 3 (HOT SHUTDOWN), a through-wall leak at the welded junction of the 4-inch decontamination port and the 'B' reactor recirculation system suction piping was identified. Unidentified reactor coolant system leakage was at approximately 0.73 gpm prior to the plant shutdown, which was well below the Technical Specification limit of 5 gpm. There was no equipment out of service that impacted this event.

DESCRIPTION OF OCCURRENCE

On March 27, 2005, reactor power was reduced to approximately 6% power (Operational Condition 2) to perform a primary containment entry to identify the source of increasing unidentified reactor coolant system (RCS) {AB} leakage. Beginning on February 8, 2005, unidentified RCS leakage had been trending upward and was at approximately 0.73 gpm on March 27, 2005. Hope Creek personnel identified a steam leak coming from the insulated decontamination port that is connected to the 'B' reactor recirculation system {AD} between the suction of the pump and the suction isolation valve. At 2329 hours on March 27, 2005, the reactor mode switch was taken to shutdown to perform a controlled shutdown and cooldown of the plant to precisely identify and repair the leakage source.

At 1035 hours on March 28, 2005 while in Operational Condition 3 after removal of the insulation from the decontamination port, the steam leakage was determined to be coming from a crack located on the 4-inch NPS portion of the decontamination port. The decontamination port is 4-inch diameter SA-376 Type 304 stainless steel Schedule 80 ASME Code Class 1 piping.

Hope Creek entered cold shutdown at 1452 hours on March 28, 2005. The RCS unidentified leakage was well below the Technical Specification limit of 5 gpm.

This event is being reported in accordance with 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."

CAUSE OF OCCURRENCE

The cause of the through-wall crack in the decontamination connection was fatigue initiation and propagation as a result of the following:

- original subsurface weld defect, and
- presence of a decontamination port geometry that experienced damaging high cycle vibration during certain recirculation system operating conditions.

The fatigue crack was initiated at a subsurface (near the outside diameter (OD) surface) weld defect in combination with the presence of the decontamination spool geometry that experienced damaging high cycle vibration during certain recirculation system operating conditions.

The metallurgical examinations identified fatigue initiation and propagation. The as-found 'B' decontamination port contained an approximately 4-inch long circumferential crack. A significant portion of the fracture surface was covered with a heavy oxide scale suggesting that this portion existed for an extended period. A minor portion of the fracture surface nearest the inside surface was much less oxidized suggesting a more recent fracture that was responsible for the through-wall leak.

Analysis revealed that the natural frequencies of the 'B' decontamination port were coincident with the 5X vane passing frequencies of the reactor recirculation pump at normal operating speeds. The natural frequencies of the decontamination port ranged from 122 to 135 Hz. The 5X vane passing frequency of the reactor recirculation pump at normal operating speeds is 120 to 125 Hz.

An analysis of pump operating data indicated that on February 8, 2005, the pump experienced a transient 5X vibration acceleration at a speed of 1501 rpm that was high in amplitude and short in duration, apparently due to the convergence of the 5X vane passing frequency with acoustic resonance of the recirculation piping system. An analysis of the drywell floor leakage demonstrated an increasing trend in unidentified RCS leakage following this event.

PREVIOUS OCCURRENCES

A review of LERs at Salem and Hope Creek generating stations for the previous two years was performed to identify prior similar occurrences. LER 354/04-010-00 identified an October 10, 2004, pipe failure of the A moisture separator drain line that led to a manual reactor scram. Although the October 10, 2004, event identifies a pipe failure, the mechanism of this pipe failure was different from the current event. The October 10, 2004 pipe rupture was the result of fatigue failure due to 25 days of high dynamic loading caused by the flow through a failed open valve. The corrective actions associated with the October 10, 2004 event would not have prevented this current event from occurring.

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SAFETY CONSEQUENCES AND IMPLICATIONS

An analysis of pump operating data indicated that on February 8, 2005, the 'B' recirculation pump experienced a transient 5X vibration acceleration at a speed of 1501 rpm that was high in amplitude and short in duration. An analysis of the drywell floor leakage demonstrated an increasing trend in unidentified RCS leakage following this event. After the February 8, 2005, transient 5X vibration acceleration, an administrative limit for the 'B' recirculation pump speed was put in place to prevent entering this region.

As the result of monitoring the increasing trend in RCS unidentified leakage, Hope Creek performed a controlled power reduction of the plant on March 27, 2005, to perform a primary containment entry and identified the source of the leakage as the insulated 'B' decontamination port. Hope Creek then proceeded to cold shutdown to precisely identify and repair the leakage source. After removal of the insulation from the decontamination port, with the plant in Operational Condition 3 (hot shutdown), a through-wall leak at the welded junction of the 4-inch decontamination port and the 'B' reactor recirculation system suction piping was identified. The 'B' decontamination port is located between the suction of the 'B' recirculation pump and the suction isolation valve. Therefore, in the unlikely event that the crack would have propagated, the decontamination port could have been isolated by the closure of the 'B' recirculation loop suction and discharge isolation valves. RCS unidentified leakage was approximately 0.73 gpm, which was well below the Technical Specification limit of 5 gpm. The crack on the decontamination port was repaired prior to restart of the plant.

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 controlled safe cold shutdown condition following the identification of the 'B' decontamination port through-wall leak.

CORRECTIVE ACTION

1. The decontamination ports for both the A and B recirculation loops were modified. The 4-inch decontamination port was shorted to increase the natural frequencies so they are not coincidental with the normal recirculation pump vane passing frequencies. These modifications were completed prior to the plant startup.
2. A review of connections to the reactor recirculation loops was performed to determine if other similar connections would be susceptible to the failure experienced with the decontamination port. This review consisted of finite element analysis, vibration analysis, modal analysis, isometric review and previous ISI inspections. A total of twenty-three (23) welds were inspected during the plant shutdown using a combination of nondestructive examination (NDE) techniques. These NDE inspections were satisfactory.

COMMITMENTS

This LER contains no commitments.