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Dominion®

JUN 21 2012

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Serial No. 12-439
LIC/NW/R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSEE EVENT REPORT 2012-004-00

Pursuant to 10 CFR 50.73, Dominion Energy Kewaunee, Inc., hereby submits the following Licensee Event Report applicable to Kewaunee Power Station.

Report No. 50-305/2012-004-00

This report has been reviewed by the Facility Safety Review Committee and will be forwarded to the Management Safety Review Committee for its review.

If you have any further questions, please contact Mr. Jack Gadzala at (920) 388-8604.

Very truly yours,

A. J. Jordan
Site Vice President, Kewaunee Power Station

Attachment

Commitments made by this letter: NONE

IE22
NRR

cc: Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Mr. K. D. Feintuch
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop O8-H4A
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Kewaunee Power Station

NRC FORM 366 U.S. NUCLEAR REGULATORYCOMMISSION
(10-2010)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

Estimated burden per response to comply with this mandatory collection request: 80 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Service (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Kewaunee Power Station	**05000305**	**1** OF **3**

4. TITLE

Pressure Boundary Leakage from Socket Weld on ¾-Inch Pipe to Sample Valve RHR-600

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
										05000
04	27	2012	2012	-- 004 --	00	06	21	2012	FACILITY NAME	05000

9. OPERATING MODE: **5**

10. POWER LEVEL: **0**

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	TELEPHONE NUMBER (*include Area Cod*)
Tim Hanna	**920-388-8414**

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	BP	PSF	Custom	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 April 27, 2012, during follow up inspection of a previously identified inactive boric acid leak, active leakage was identified in the upstream piping between containment isolation valve RHR-600 and the A train 10-inch residual decay heat removal piping. This leakage was determined to be pressure boundary leakage.

In the residual heat removal system cooldown lineup, this A train piping is the common flow path for both A and B train residual heat removal loops that maintain flow to the reactor coolant system cold leg and provide core cooling. Although the residual heat removal system remained in service providing decay heat removal, both trains were declared inoperable due to operational leakage from an ASME Code Class 2 component.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(ii)(A), for the plant, including its principal safety barriers, being seriously degraded; 10 CFR 50.73(a)(2)(v), for any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition and (B) Remove residual heat; and 10 CFR 50.73(a)(2)(vii), a single cause where two independent trains became inoperable in a single system designed to: (A) Shut down the reactor and maintain it in a safe shutdown condition and (B) Remove residual heat.

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NARRATIVE

Event Description:

On April 27, 2012, during follow up inspection of a previously identified inactive boric acid leak, active leakage was identified in the upstream piping between sample valve RHR-600 [SMV] and the A train 10-inch residual decay heat removal piping while the unit was in MODE 5, Cold Shutdown, following core reload and reactor [RCT] reassembly.

Dry white boric acid accumulation was identified two days prior (April 25, 2012) at the socket [PSF] weld on the ¾-inch sample line that taps off the A train residual heat removal (RHR) injection line. In the residual heat removal system [BP] cooldown lineup, this A train piping is the common flow path for both A and B train residual heat removal loops that maintain flow to the reactor coolant system (RCS) [AB] cold leg and provide decay heat removal capability to cool the reactor core.

With operational leakage identified in the ASME Code Class Piping while in Mode 5, Cold Shutdown, both trains of residual heat removal were declared inoperable. Although quantitative volumetric examination of the leaking weld area could not be performed, the residual heat removal system was functionally available and remained in service to provide core cooling.

Per Technical Specifications (TS) 3.4.7, RCS Loops – MODE 5, Loops Filled, both RHR trains are required to be operable. Action was initiated in accordance with TS 3.4.7 Required Action C.2 to restore one RHR loop to an operable status.

Approval by the Nuclear Regulatory Commission for relief from ASME Boiler and Pressure Vessel Code, Section XI requirements was requested. Pursuant to the provisions of 10 CFR 50.55a(a)(3)(ii), a request was made for a temporary deviation from the requirements of ASME Section XI, Appendix IX, Article IX-1000, Paragraph (c)(2), which prohibits the use of clamping devices on "...portions of a piping system that forms the containment boundary" and ASME Section XI, Appendix IX, Article IX-6000(a) which states that the area immediately adjacent to the clamping device shall be examined using a volumetric method.

Specifically, a temporary alternate repair of the RHR piping was proposed by installing a leak-limiting device to preserve containment integrity and structural integrity in the area of the leaking socket weld to regain system operability prior to making a MODE change from MODE 5 to MODE 4, Hot Shutdown. The temporary alternate repair would remain in place until the unit achieved MODE 4. After the unit would reach MODE 4, core cooling would be provided by reactor coolant pumps [P] circulating water through the core and to the steam generators [SG]. In this condition, with core cooling being provided by the steam generators, the system could be isolated, and the defect could be repaired by welding.

During the installation process of the leak-limiting device (after it was clamped to the 10-inch RHR pipe), a second through-wall leak was created downstream of the original leak location while initiating welding to attach the device to the ¾-inch schedule 40 pipe. The welder inadvertently created a through-wall leak while striking an arc to initiate a 3/16-inch weld. Subsequent ultrasonic examination of the exposed ¾-inch pipe confirmed sound base metal adjacent to the new through-wall leak.

Subsequently, a second leak-limiting device was manufactured and successfully installed, the unit was taken to MODE 4, Hot Shutdown, and the RHR system was isolated and drained. Permanent weld repairs were completed and the RHR system was returned to an operable status.

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This event is being reported pursuant to 10 CFR 50.73(a)(2)(ii)(A), for the plant, including its principal safety barriers, being seriously degraded; 10 CFR 50.73(a)(2)(v), for any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition and (B) Remove residual heat; and 10 CFR 50.73(a)(2)(vii), a single cause where two independent trains became inoperable in a single system designed to: (A) Shut down the reactor and maintain it in a safe shutdown condition and (B) Remove residual heat.

Event and Safety Consequence Analysis:

When the original leak was identified, the RHR system was in service and remained in service until the system was isolated to complete permanent weld repairs. Although both trains of RHR were declared inoperable, they maintained functionality to provide decay heat removal. If the RHR system would have been required to be isolated as a result of increased leakage, decay heat removal could have been provided by a safety injection pump taking suction from the refueling water storage tank.

Therefore, there is minimal safety consequence associated with the socket weld leak that was identified on April 27, 2012 and the second through-wall leak that was created on April 30, 2012.

Cause:

The cause of the defect in the original socket weld is not known. Due to the repair activities performed, no definitive cause analysis could be completed.

The second through-wall leak in the ¾-inch pipe was caused by improper welding. While initiating the welding process to attach the Hub Clamp to the pipe, the welder created a through-wall leak while striking an arc to initiate the weld. Contributing to this error was the development of the weld plan and the pre-job briefing for this welding activity.

Corrective Actions:

Immediate actions were to develop a repair plan, obtain regulatory approval, make the leak repairs, and return the RHR system to an operable status on May 6, 2012. The qualifications for the welder were removed and the previous welding activities performed by the welder were reviewed for adequacy.

Apparent cause evaluations were initiated to analyze the welding activities and implementation of the welding program.

Similar Events:

A review of LERs at Kewaunee Power Station over the past three years did not identify any similar events.