FENOC

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March 11, 2011

L-11-061

10 CFR 50.73

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

SUBJECT:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
LER 2010-002-01

Enclosed is Licensee Event Report (LER) 2010-002-01, "270 Degree Circumferential Flaw Found on Residual Heat Removal System Drain Valve Socket Weld." This event was previously reported in accordance with 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(B) on November 29, 2010. This LER Supplement updates the direct cause of event based on the results of the metallurgical examination of residual heat removal system drain valve socket weld failure.

There are no regulatory commitments contained in this submittal. Any actions discussed in this document that represent intended or planned actions are described for the NRC's information, and are not regulatory commitments.

If there are any questions or if additional information is required, please contact Mr. Brian T. Tuite, Manager, Regulatory Compliance at 724-682-4284.

Sincerely,

Paul A. Harden

Attachment

c: Mr. W. M. Dean, NRC Region I Administrator
Mr. D. L. Werkheiser, NRC Senior Resident Inspector
Ms. N. S. Morgan, NRR Project Manager
INPO Records Center (via electronic image)
Mr. L. E. Ryan (BRP/DEP)

IE22
NRR

NRC FORM 366 (10-2010)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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4. TITLE

270 Degree Circumferential Flaw Found on Residual Heat Removal System Drain Valve Socket Weld

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: None	DOCKET NUMBER
10	02	2010	2010	- 002 -	01	03	11	2011	FACILITY NAME	DOCKET NUMBER

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 Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Brian T. Tuite, Manager, Regulatory Compliance	(724) 682-4284

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	V	W120	Y					

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

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

Shortly after the Beaver Valley Power Station Unit No.1 (BVPS-1) shut down for a refueling outage, an active boric acid leak was identified on a drain valve (1RH-200) located on the common suction piping for the Residual Heat Removal (RHR) system. Following initial identification of the active leak, a Non-Destructive Examination (VT-1 examination) was performed on the valve and its associated piping. A circumferential flaw (crack) of 270 degrees in length with water seeping from the toe of the weld was discovered. Based on discovery of the weld crack, both trains of the RHR system were then declared inoperable. With no trains of the RHR System operable in Mode 5, this condition is reportable as an event that could have prevented the fulfillment of a safety function for systems needed to remove decay heat per 10CFR50.73(a)(2)(v)(B). Technical Specification (TS) 3.4.7 Actions were appropriately entered when the VT-1 inspection determined the presence of a weld flaw on 1RH-200. However, since the adverse condition was identified prior to the completion of the VT-1 examination and no action was immediately initiated per TS 3.4.7 for no operable RHR trains (since the significance of the noted leakage could not be recognized until the VT-1 examination was completed), this was an (inadvertent) operation/condition prohibited by plant's TS, and is also reportable per 10 CFR 50.73(a)(2)(i)(B).

The direct cause was determined to be a fatigue failure of the socket weld. A metallurgical examination of the 1RH-200 socket weld failure determined that the mechanism responsible for the cracks in the pipe and socket weld is fatigue. The most probable root cause is less than adequate (LTA) consideration of vibration induced fatigue in the design process. The safety significance associated with the BVPS Unit 1 cracked weld on the RHR drain line valve is considered to be very low.

NARRATIVE

There were no structures, components, or systems that were inoperable at the start of the event that contributed to the event. Energy Industry Identification System (EIIS) codes are identified in the text using the format [XX].

DESCRIPTION OF EVENT

On October 2, 2010, the Beaver Valley Power Station (BVPS) Unit No. 1 was shut down at 0011 hours to enter a scheduled refueling outage (1R20). At 0325 hours, an active boric acid leak (approximately 5 drips per minute) was identified on valve 1RH-200, a 3/4 inch manual drain valve on the Residual Heat Removal (RHR) System [BP] located inside containment. It was reported at 0512 hours that this drain valve leak may be from a weld, although the leak's exact location was not known. This drain valve is on a 3/4 inch ASME (American Society of Mechanical Engineers) Class 2 drain pipe located just downstream of the two RHR inlet isolation valves which separate the RHR System from the Reactor Coolant System [AB] and is located on the inlet piping (14 inch) common to both trains of RHR. The first RHR system pump had been started at 0414 hours and Mode 5 was entered at 0453 hours.

Following identification that the leak may be on a weld, a Non-Destruction Examination (NDE) VT-1 inspection was immediately requested on the valve and its associated piping. This inspection subsequently reported a circumferential flaw (crack) of 270 degrees in length with water seeping from the toe of the weld at 1011 hours. At this time, both trains of RHR system were declared inoperable based upon the lack of reasonable assurance that both trains of RHR system remain operable with a cracked weld on 1RH-200. Technical Specification (TS) 3.4.7 Actions were appropriately entered when the VT-1 inspection determined the presence of a weld flaw on 1RH-200.

The Required Action for Condition C of TS 3.4.7 requires action immediately to restore one RHR loop to operable status and operation. The operational decision making process was immediately initiated to evaluate several options to restore the RHR system to operable status. A temporary strongback support was initially installed at 0526 hours on October 3, 2010 to promptly provide additional support of the flawed pipe section. Subsequently, the use of ASME, Section XI, Appendix IX "Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundary" was chosen as the method to return the RHR System to operable status. An ASME Code repair was installed on October 6, 2010. Both trains of RHR were then declared operable at 1156 hours on October 6, 2010. The Prompt Operability Determination (10-83533) associated with the ASME Code repair required that the system be permanently repaired at the first available opportunity pursuant to applicable ASME Code requirements.

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NARRATIVE

Following offload of the reactor core, the RHR System was no longer required to be in operation and was isolated to repair 1RH-200. 1RH-200 was permanently repaired by replacement on October 15, 2010. 1RH-200 was declared operable on October 18, 2010 following completion of required post-maintenance inspections/testing.

CAUSE OF EVENT

The direct cause is fatigue failure of the valve inlet socket weld on 1RH-200. The most probable root cause is less than adequate design considerations to accommodate vibration induced fatigue on small bore vents and drains in the BVPS Unit No. 1 RHR System piping.

ANALYSIS OF EVENT

In Mode 5 with Reactor Coolant System loops filled, one RHR loop shall be operable and in operation, and either one additional RHR loop shall be operable or the secondary side water level of at least one steam generator shall be operable pursuant to BVPS Unit 1 Technical Specification 3.4.7. Both trains of RHR were declared inoperable on October 2, 2010 based upon the lack of reasonable assurance that both trains of RHR system remained operable with a cracked weld on 1RH-200. Condition C of Technical Specification 3.4.7, for no required RHR loops operable, was entered at 1011 hours on October 2, 2010 and was not exited until 1156 hours on October 6, 2010. With no trains of the RHR System operable in Mode 5, this condition is reportable as an event that could have prevented the fulfillment of a safety function for systems needed to remove decay heat per 10CFR50.73(a)(2)(v)(B).

The Required Action for Condition C of Technical Specification (TS) 3.4.7 requires action immediately to restore one RHR loop to operable status and operation. Since the adverse RHR condition existed when TS 3.4.7 became applicable upon entering Mode 5 at 0453 hours on October 2, 2010 and no action was immediately initiated at 0453 hours (since the significance of the noted leakage was not recognized until 1011 hours), this was an (inadvertent) operation/condition prohibited by plant's Technical Specifications, and is also reportable per 10 CFR 50.73(a)(2)(i)(B).

The safety significance associated with the cracked weld on the BVPS Unit 1 RHR drain line valve 1RH-200 is considered to be very low. This is based on the fact that the indicated leakage was insignificant, contingencies were in place to limit any further crack propagation, and in the unlikely event that a pipe rupture occurred, the shutdown safety functions for decay heat removal and RCS inventory control could be maintained.

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CONTINUATION SHEET

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This event was previously reported as an event that could have prevented the fulfillment of a safety function of systems needed to remove decay heat, pursuant to 10 CFR 50.72(b)(3)(v)(B) at 1538 hours on October 2, 2010 (Event Notification No. 46304).

CORRECTIVE ACTIONS

1. Valve 1RH-200 was replaced and the pipe length shortened to minimize the susceptibility to natural frequency vibration induced fatigue on October 15, 2010. The new valve was declared operable following required post-maintenance inspection/testing.

2. The prior valve 1RH-200 with a piping segment attached was metallurgically examined to validate the probable cause of the flaw (crack). The results of this examination determined that the mechanism responsible for the cracks in the pipe and socket weld is fatigue.

3. The extent of condition at BVPS Unit 1 involved selecting at least two vents/drains for additional examinations during the 1R20 refueling outage from each of six susceptible systems. PT (penetrant test) and visual examinations were performed on the thirteen selected locations, all with acceptable results.

4. The thirteen BVPS Unit 1 potentially susceptible piping locations described in Corrective Action #3 above, along with the 1RH-200 location, will be evaluated for natural frequency and the potential need for further permanent modifications.

5. The engineering Design Interface Review Checklist Form will be revised to ensure that modifications to safety related system vent and drain lines adequately evaluate and minimize susceptibility to vibration induced fatigue weld failures.

6. A review will be performed of BVPS Unit No. 2 unsupported RHR System vents/drains along with a sample of other major systems unsupported vents/drains during the upcoming BVPS Unit No. 2 refueling outage scheduled for the Spring of 2011. Should the sample indicate a design weakness of vibration induced fatigue, an expanded scope may be considered.

7. A plant operating experience report was issued on this event on 10/29/2010 (OE 32192).

Completion of the above and other corrective actions are being tracked through the BVPS corrective action program.

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NARRATIVE

PREVIOUS SIMILAR EVENTS

BVPS Unit 1 has not experienced any significant primary coolant leaks from any vent or drain valves since prior to 1994. BVPS Unit 1 Licensee Event Report 2009-004 described an event involving two notable flaws in a Reactor Coolant System drain line; however these flaws were not through-wall nor did they involve a valve or weld.

BVPS Unit 2 identified a crack in a similar RHR drain valve weld in 2008. A separate condition report has been entered into the Corrective Action Program to evaluate why prior Operating Experience events did not initiate sufficient corrective action to minimize the probability of the BVPS Unit 1 1RH-200 event occurring in 2010.

CR 10-83533/10-84995