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10 CFR 50.73

1CAN111603

November 29, 2016

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

Subject: Licensee Event Report 50-313/2016-004-00
Decay Heat Removal System Socket Weld Leak due to a Vibration Induced Fatigue Crack
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to the reporting requirements of 10 CFR 50.73, attached is the subject Licensee Event Report concerning the inoperability of both trains of decay heat removal at Arkansas Nuclear One, Unit 1, which resulted in a loss of safety function.

There are no new commitments contained in this submittal.

Should you have any questions concerning this issue, please contact Stephenie Pyle, Manager, Regulatory Assurance, at 479-858-4704.

Sincerely,

ORIGINAL SIGNED BY RICHARD L. ANDERSON

RLA/ble

Attachment: Licensee Event Report 50-313/2016-004-00

cc: Mr. Kriss Kennedy
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
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Arlington, TX 76011-4511

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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form
http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

APPROVED BY OMB: NO. 3150-0104 **EXPIRES: 10/31/2018**

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by 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

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4. TITLE
Decay Heat Removal System Socket Weld Leak due to a Vibration-Induced Fatigue Crack

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 N/A	DOCKET NUMBER N/A
09	30	2016	2016	004	000	11	29	2016	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE: 5

10. POWER LEVEL: 000

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)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☒ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☒ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMER *(Include Area Code)*
Stephenie L. Pyle, Manager, Regulatory Assurance	(479) 858-4704

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	BP	VTV	CFC	N/A	N/A	N/A	N/A	N/A	N/A

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

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

On September 29, 2016, at Arkansas Nuclear One, Unit 1 (ANO-1), during refueling outage 1R26 with both trains of Decay Heat Removal (DHR) in service, a 0.125 gpm leak was identified in the DHR system at a one-inch drain line. This leak was on a section of cross-connect piping shared by both trains of the DHR system. The consequence of the leak was that both trains of the DHR system were declared inoperable.

The leakage was due to a fatigue crack caused by vibration of the drain line due to a pipe support that was not designed for system vibration.

Other systems and components in ANO-1 and ANO, Unit 2 (ANO-2) exposed to elevated system vibration were evaluated with respect to this condition. As a result of this evaluation, socket welds on other drains and vents in the ANO-1 DHR system were cut out and replaced, and pipe supports were modified where needed to withstand system vibration.

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NARRATIVE

A. Plant Status

At the time this condition was identified, ANO-1 was shut down and in a refueling outage with the core fully loaded. No structures, systems, or components were out of service at the time of this event that contributed to this event.

B. Event Description

At approximately 20:47 on September 29, 2016, the ANO-1 Control Room received a report of a leak in the ANO-1 Reactor Building (RB). Investigation revealed a 0.125 gpm leak in a class 1 pipe from a socket weld [VTV] where the drain pipe connects to the main DHR system injection line. This leak was on a section of cross-connect piping shared by both trains of the DHR system.

Operations closed RB penetrations providing direct access from the RB atmosphere to the outside atmosphere in accordance with required actions of DHR Technical Specification (TS) 3.9.5.

To reduce the potential for propagation of the weld crack, Operations maintained DHR flow at less than 2000 gpm and monitored data from vibration probes affixed to the affected piping to maintain low vibration levels.

To remove the reliance on the DHR system, actions were taken to defuel the reactor vessel, which was accomplished at 12:00 on Thursday, October 6, 2016. The leak remained less than or equal to 0.125 gpm from the time of discovery until the piping was removed from service for repair.

C. Background – System Design

The DHR /Low Pressure Injection (LPI) [BP] pumps have three major functions. The first is to remove decay heat from the fuel and Reactor Coolant System (RCS) once the plant is shut down and cooled down. To do this, the DHR pump takes suction from the RCS, discharges through a cooler which is cooled by Service Water, and injects the cooled water back into the RCS. A second function is to provide required net positive suction head from the RB Sump to the High Pressure Injection (HPI) pumps in the event of an intermediate size loss-of-coolant accident (LOCA). The third function is LPI of borated water from the Borated Water Storage Tank or RB Sump into the RCS in the event of a large-break LOCA or after cooling down and depressurizing sufficiently following an intermediate break LOCA.

D. Event Cause

This condition was self-revealing. The direct cause of the leak was a fatigue crack caused by vibration of the drain.

ANO's corrective actions for previous DHR/LPI system drain line socket weld cracks did not either adequately address worst case operating conditions or eliminate the source of the vibration. In addition, the DHR operating procedure contained inadequate administrative barriers to minimize DHR system vibration. Furthermore, previous corrective actions did not ensure that Operations training programs were updated regarding the basis for limiting DHR system flow to minimize system vibration.

Extent of Condition

The socket welds on other drains and vents in the ANO-1 DHR system were evaluated which resulted in cut out and replacement of other vents and drains. The drain line socket welds were replaced with enhanced welds. The tie-back supports for DH-1037 and DH-1450 were designed using as-measured vibration spectra to provide hardening against system vibration. The evaluation of other systems and components identified no additional corrective actions for vibration-induced damage.

E. Corrective Actions

The corrective actions taken and planned are:

- The cracked socket weld was repaired. The tie-back drain line support was redesigned and replaced. This hardens the system against vibration-induced fatigue cracking and component damage.
- The DHR system piping and supports inside the RB were inspected for damage. No other damage was identified. The stresses on the DHR system one-inch drain line isolation valves inside the RB were calculated using as-measured response spectra to ensure acceptable performance. The socket welds for several other drain valves were cut out and replaced using enhanced welds.
- Post-modification testing of the drain valve and support modifications confirmed that the corrective actions would result in acceptable vibration of the drain lines during high system flow rates.
- The DHR operating procedure is planned to be revised to provide specific administrative guidance to minimize DHR system vibration, consistent with results of post-modification testing.

F. Safety Significant Evaluation

For the function of the DHR system, the effect of the one-inch drain line completely separating from the eight-inch Core Flood Tank injection line is estimated to be a leak of 98 gpm with the backpressure that would exist at 4,000 gpm of DHR system flow. When the leak was discovered, both DHR pumps were in service. The second DHR pump was secured and the remaining pump flow was throttled back to approximately 2000 gpm (2000 gpm is adequate for core cooling at all expected decay heat levels). The maximum flow that can be provided by a DHR/LPI pump is 3547 gpm per CALC-92-E-0077-08. Therefore, the potential loss of approximately 175 gpm through the leak path would not have prevented either DHR train from providing adequate core cooling.

For the function of piggyback operation (suction from RB Sump through LPI Pump to HPI pump) the HPI pump has a maximum flowrate of approximately 525 gpm. Considering the aforementioned LPI flow capability, complete separation of the one-inch drain line would not have prevented the HPI pump from performing its specified safety function.

In the LPI mode of operation following a trip from full power, the mechanism which caused the flow-induced vibration fatigue failure would not have occurred until RCS pressure had lowered to approximately 30 psig. This is because venturi cavitation does not occur until the pressure downstream of the venturis reaches approximately 30 psig. The extent-of-condition review identified three drains that were susceptible to this vibration-induced fatigue failure. The total loss of the three drains equates to an area of about 1.56 sq. inches, which is bounded by supporting LOCA calculations. A review of the associated pump curves indicated that there is more than enough margin to account for the losses through the drain lines assuming these two failed during a LOCA.

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The significance of the leak is lessened by the fact that the leak is located inside the RB and any leaks are fully recoverable by aligning the suction of the DHR/LPI pump to the RB Sump.

Summary:

The actual consequences as stated in the problem statement were that both trains of DHR were declared inoperable. There were no other actual consequences to general safety of the public, nuclear safety, industrial safety, and radiological safety for this event.

The potential consequences to general safety of the public, nuclear safety, industrial safety, and radiological safety of this event are reduced flow to the core in the event of a large-break LOCA. However, adequate flow remains available to ensure the reactor can be maintained in a safe shutdown condition.

Based on the above, the significance of this condition is low.

G. Basis for Reportability

10 CFR 50.73(a)(2)(v)(B): Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat.

10 CFR 50.73(a)(2)(v)(D): Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

At the time of discovery, both trains of the DHR system were in service. Based on the location of the leak both trains of DHR were declared inoperable. With both trains being declared inoperable the associated safety function could not be met. In accordance with the NUREG-1022 guidelines, at the time of discovery ANO-1 made the 8-hour notification under EN 52271 and this LER is required.

H. Additional Information

10 CFR 50.73(b)(5) states that this report shall contain reference to "any previous similar events at the same plant that are known to the licensee." NUREG 1022 reporting guidance states that term "previous occurrences" should include previous events or conditions that involved the same underlying concern or reason as this event, such as the same root cause, failure, or sequence of events.

A review of the ANO corrective action program and LERs for the previous three years was performed. No relevant LER were similar to the failure mode.