NRC FORM 366 (02-2014)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 1/31/2017
LICENSEE EVENT REPORT (LER)		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 Section (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.

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4. TITLE
Indication in Control Rod Drive Mechanism Nozzle Weld due to Embedded Flaws Opening Up from Thermal and Pressure Stresses during Operation

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
04	03	2015	2015	- 002 -	00	06	02	2015	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE: 6

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)(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

LICENSEE CONTACT	TELEPHONE NUMBER
Phillip J. Raush, Regulatory Assurance Manager	(815) 417-2800

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	AB	1718E72	W120	Y	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

On April 3, 2015, during the Braidwood Station Unit 1 refueling outage (A1R18), an in service liquid penetration (PT) examination was performed on the previously repaired control rod drive mechanism (CRDM) penetration 69. During the examination of the repair for CRDM penetration 69, one 3/8 inch rounded indication was documented exceeding the acceptance criterion (ASME Section III 1971 Edition through the Summer 1973 Addenda) of dimensions greater than 3/16 inch. This LER is being submitted in follow-up to ENS 50953 made on April 3, 2015.

Based on industry experience, the cause of this event was determined to be mechanical discontinuities/minor subsurface voids opening up to the weld surface due to thermal and/or pressure stresses during plant operation.

The indication in penetration 69 was reduced to acceptable size as approved by the NRC in Braidwood Relief Request I3R-09.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(A), "any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded" since the as found indication did not meet the applicable acceptance criterion referenced in ASME Code Case N-729-1 to remain in-service without repair.

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NARRATIVE

A. Plant Operating Conditions Before the Event:

Event Date: April 3, 2015

Unit: 1 Mode: 6 Reactor Power: 000 percent

Unit 1 Reactor Coolant System [AB]: Shutdown for refueling outage, fuel movement in progress

No structures, systems or components were inoperable at the start of this event that contributed to the event.

Description of Event:

On April 3, 2015 at 1631 hours, a total of five recordable indications located within the embedded flaw repair weld were documented, with one exceeding the acceptance criterion (ASME Section III 1971 Edition through the Summer 1973 Addenda) of dimensions greater than 3/16 inch. The 3/8 rounded indication exceeding the acceptance criterion was discovered in the stainless steel buffering weld layer of the embedded flaw repair of control rod drive mechanism (CRDM) Penetration 69. The A1R18 examination of the embedded flaw repair in penetration 69 was performed in accordance with Braidwood Third Interval Relief Request I3R-09 which requires liquid penetrant (PT) examination of embedded flaw weld repairs every refuel outage.

This was the second in service examination of the repair weld since it was applied in A1R16 (April 2012). The weld was also repaired in A1R17 (September 2013). There was a total of one rounded recordable indication exceeding the acceptance criterion of 3/16 inch. Per the original Construction Code (ASME Section III 1971 Edition through the Summer 1973 Addenda), unacceptable indications include "Rounded indications with dimensions greater than 3/16 inch." In addition to the PT examination of the embedded flaw weld repair on Penetration 69, all penetrations were examined by ultrasonic and eddy current methods using procedures and personnel qualified in accordance with the EPRI Performance Demonstration Program. The EPRI program is implemented by 10 CFR 50.55a, "Codes and standards", which includes the use of ASME Section XI Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1". No indications of Primary Water Stress Corrosion Cracking (PWSCC) or through wall leakage were observed on any of the remaining penetrations. A bare metal visual inspection of the exterior surfaces of the reactor head and penetrations was also performed during A1R18 in accordance with ASME Section XI Code Case N-729-1. There was no indication of through wall leakage observed during the bare metal visual examination. Contingency plans to reduce this indication to an acceptable dimension were implemented. No other CRDM penetration repairs were required in A1R18.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(A), "any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded" since the as found indication did not meet the applicable acceptance criterion referenced in ASME Code Case N-729-1 to remain in-service without repair. This LER is being submitted in follow-up to ENS 50953 made on April 3, 2015.

B. Cause of Event

Based on industry experience, the cause of the flaw is attributed to existing mechanical discontinuities/minor subsurface voids opening up the weld surface due to thermal and/or pressure stresses during plant operation. An acceptable rounded indication was previously detected during A1R17 at this location.

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D. **Safety Consequences:**

This condition had no actual safety consequences impacting plant or public safety.

The flaw was identified in a timely manner and repaired. The flaw was identified as part of a required periodic inspection. Potentially, if the flaw remained undetected, it could have over time, propagated through the embedded flaw repair to form a leak path through the reactor coolant pressure boundary.

Based on the A1R18 documented characteristics and dimensions of the observed PT indication, there was no Safety Significant functional failure (i.e., loss of safety function) as a result of these indications. The primary coolant pressure boundary was maintained and capable of preventing the release of radioactive material.

The Rod Drive system remained functional and there was no impact to structural integrity.

E. **Corrective Actions:**

The identified indication was reduced to an acceptable dimension by mechanical means (manual grinding) and verified to be within the acceptance standard by liquid penetrant.

Develop and implement a corrective action plan that includes consideration of a long term Alloy 600 mitigation and other acceptable alternatives for the remaining acceptable, but unrepaired penetrations on the Braidwood Units' 1 and 2 RPV closure heads.

F. **Previous Occurrences:**

Previous Licensee Event Reports were made in April 2012 and November 2013 at Braidwood Station Unit 1 for indications on CRDM penetration 69 (LER 2012-002-00 and LER 2013-002-00).

G. **Component Failure Data:**

Manufacturer	Nomenclature	Model	Mfg. Part Number
Westinghouse	Reactor Vessel Integrated Head Package Termination	1718E72	N/A