# Exelon Generation.

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10 CFR 50.73

November 23, 2016
BW160095

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

Braidwood Station, Unit 1
Renewed Facility Operating License Nos. NPF-72
NRC Docket No. STN 50-456

Subject: Licensee Event Report 2016-003-00 – Indication in Control Rod Drive Mechanism Nozzle Weld due to Embedded Flaws Opening Up from Thermal and Pressure Stresses during Operation

The enclosed Licensee Event Report (LER) is being submitted in accordance with 10 CFR 50.73, "Licensee Event Report System."

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Steven Reynolds, Regulatory Assurance Manager, at (815) 417-2800.

Respectfully,

Marri Marchionda-Palmer
Site Vice President
Braidwood Station

Enclosure: LER 2016-003-00

cc: NRR Project Manager – Braidwood Station
Illinois Emergency Management Agency – Division of Nuclear Safety
US NRC Regional Administrator, Region III
US NRC Senior Resident Inspector (Braidwood Station)
Illinois Emergency Management Agency – Braidwood Representative

| NRC FORM 366 (11-2015) U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018 |
|---|---|
| **LICENSEE EVENT REPORT (LER)**<br>(See Page 2 for required number of digits/characters for each block) | 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 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. |

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
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| Braidwood Station, Unit 1 | 05000456 | 1 OF 3 |

**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 | DOCKET NUMBER |
| | | | | | | | | | N/A | N/A |
| 10 | 02 | 2016 | 2016 | - 003 | - 00 | 11 | 23 | 2016 | N/A | N/A |

| 9. OPERATING MODE | 11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)* | | | |
|---|---|---|---|---|
| N/A – Defueled | ☐ 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) |
| 10. POWER LEVEL | ☐ 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) |
| 000 | ☐ 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 NUMBER *(Include Area Code)* |
|---|---|
| Steven Reynolds, 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 *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On October 2, 2016, during the liquid penetrant examination on the weld build up for control rod drive mechanism (CRDM) Penetration 69 during refueling outage A1R19, two rejectable rounded indications were documented. The first was a 7/32 inch rounded indication on the reactor head portion of the weld build up which was 4 inches from the transition of the head to penetration. The second was a 1/4 inch rounded indication located at the transition of the head to penetration. The transition is the point where the vertical portion of the penetration meets the horizontal area of the reactor head. This LER is being submitted in follow-up to ENS 52275.

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 indications in penetration 69 were reduced to an acceptable dimension by manual buffing.

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.

NRC FORM 366A (11-2015) U.S. NUCLEAR REGULATORY COMMISSION

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

# LICENSEE EVENT REPORT (LER) CONTINUATION SHEET



| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. LER NUMBER | | |
|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REV NO. |
| Braidwood Station, Unit 1 | 05000456 | 2016 | - 003 | - 00 |

NARRATIVE

## A. Plant Operating Conditions Before the Event:

Event Date: October 2, 2016

Unit: 1 MODE: Not Applicable – Defueled

Unit 1 Reactor Coolant System [AB]: Not Applicable

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

## B. Description of Event:

On October 2, 2016, during the liquid penetrant examination on the weld build up for control rod drive mechanism (CRDM) Penetration 69 during refueling outage A1R19, two rejectable rounded indications were documented. The first was a 7/32 inch rounded indication on the reactor head portion of the weld build up which was 4 inches from the transition of the head to penetration. The second was a 1/4 inch rounded indication located at the transition of the head to penetration. The transition is the point where the vertical portion of the penetration meets the horizontal area of the reactor head.

The A1R19 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 third in service examination of the repair weld since it was applied in A1R16 (April 2012). The weld was also repaired in A1R17 (September 2013) and A1R18 (April 2015). 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 ASM E 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 A1R19 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. Actions to reduce both indications to an acceptable dimension were completed on October 9, 2016. No other CRDM penetration repairs were required in A1R19.

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 52275 made on October 2, 2016 at 2302 CDT.

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NARRATIVE

C. **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. One of these flaws could be seen (below recordable dimension) during the previous PT inspection of Penetration 69 in A1R18 (April 2015). The other flaw was rejected in A1R18 and was reduced to an acceptable dimension through manual buffing under a planned contingency A1R18 repair package.

D. **Safety Consequences:**

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

Both flaws were identified in a timely manner and repaired. The flaws were identified as part of a required periodic inspection and neither flaw penetrated through the embedded flaw repair weld. Potentially, if either of the flaws remained undetected, they could have over time propagated through the embedded flaw repair to form a leak path through the reactor coolant pressure boundary, but the frequency of the required inspections (every refuel outage) would likely detect degradation before it reached any level of significance.

Subsequent bare metal visual and NDE volumetric examinations did not identify any evidence of through-wall pressure boundary leakage. Both indications were reduced to an acceptable dimension on October 9, 2016. Neither indication penetrated through the existing embedded flaw repair which confirms that the primary coolant pressure boundary was maintained and capable of preventing the release of radioactive material. Based on the A1R19 documented characteristics and dimensions of the observed PT indications, there was no loss of safety function as a result of these indications.

E. **Corrective Actions:**

The identified indications were reduced to an acceptable dimension by manual buffing.

F. **Previous Occurrences:**

Previous Licensee Event Reports were made in June 2012, November 2013 and June 2015 at Braidwood Station Unit 1 for indications on CRDM penetration 69 (LER 2012-002-00, LER 2013-002-00 and LER 2015-002-00).

G. **Component Failure Data:**

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