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September 30, 2010
L-10-258

10 CFR 50.73

ATTN: Document Control Desk
United States Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station
Docket Number 50-346, License Number NPF-3
Licensee Event Report 2010-002 Revision 01

Enclosed is Revision 01 to Licensee Event Report (LER) 2010-002, "Control Rod Drive Nozzle Primary Water Stress Corrosion Cracking and Pressure Boundary Leakage." This LER is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B), operation in a condition prohibited by the Technical Specifications, and 10 CFR 50.73(a)(2)(ii)(A), condition of the plant, including its principal safety barriers, being seriously degraded. This LER is being revised to provide results of the completed Root Cause evaluation.

There are no new regulatory commitments contained in this letter or its enclosure. The actions described represent intended or planned actions and are described for information only. If there are any questions or if additional information is required, please contact Mr. Patrick J. McCloskey, Manager, Site Regulatory Compliance, at (419) 321-7274.

Sincerely,

Barry S. Allen

Barry S. Allen

GMW

Enclosure: LER 2010-002-01

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board

IE22
NRR

NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104	EXPIRES 8/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		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 Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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

Control Rod Drive Nozzle Primary Water Stress Corrosion Cracking and Pressure Boundary Leakage

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
03	12	2010	2010	002	01	09	30	2010	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
6	☐ 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)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
000	☐ 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)
Gerald M. Wolf, Supervisor, Nuclear Compliance	(419) 321-8001

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	NZL	B105	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)

On March 12, 2010, during the Davis-Besse Nuclear Power Station (DBNPS) Refueling Outage, results of planned ultrasonic examinations identified Control Rod Drive Mechanism (CRDM) nozzles that did not meet applicable acceptance criteria. Additionally, results of the planned bare metal visual examination of the outer surface of the Reactor Vessel Closure Head (RVCH) identified boric acid deposits indicative of Reactor Coolant System leakage. No discernable wastage of the RVCH was identified. These conditions are being reported in accordance with 10 CFR 50.73(a)(2)(i)(B), operation in a condition prohibited by the Technical Specifications, and 10 CFR 50.73(a)(2)(ii)(A), condition of the plant, including its principal safety barriers, being seriously degraded.

The direct cause of this event is primary water stress corrosion cracking (PWSCC) of the CRDM nozzles. A total of 24 CRDM nozzles were repaired, and these degraded CRDM nozzles were modified utilizing the inside diameter temper bead welding method to restore the pressure boundary of the degraded nozzles. The Root Cause of this event was a less than adequate perception of the risk of PWSCC susceptibility with the replacement RVCH resulting in inadequate identification, development, and implementation of interim actions to mitigate degradation prior to replacement with a PWSCC resistant Alloy 690 RVCH.

NARRATIVE

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

DESCRIPTION OF EVENT:

On February 28, 2010, the Davis-Besse Nuclear Power Station (DBNPS) commenced the Sixteenth Refueling Outage. Planned activities during this outage included ultrasonic (UT) examinations of the Control Rod Drive Mechanism (CRDM) nozzles [AB-NZL] penetrating the reactor vessel closure head (RVCH) and bare metal visual examination of the outer surface of the RVCH to meet the requirements of 10 CFR 50.55a(g)(6)(ii)(D).

On March 12, 2010, with the station in Mode 6, Refueling, results of UT examinations performed on the CRDM nozzles began to identify nozzles that did not meet the applicable acceptance criteria. Additionally, on March 13, 2010, results of the bare metal visual examination of the outer surface of the RVCH identified boric acid deposits indicative of Reactor Coolant System (RCS)/primary water leakage. One nozzle (nozzle #4) was confirmed to have evidence of leakage, and several other nozzles (some with acceptable UT examination results) were identified with potential leakage due to tightly adhering boric acid deposits. These nozzles required additional examination to determine if degradation of the J-groove welds could have caused the leakage. Supplemental liquid dye penetrant testing identified more unacceptable indications, and a decision was made to perform eddy current examinations on all CRDM J-groove welds that had not already been dispositioned as requiring modification due to unsatisfactory UT or penetrant testing results. These eddy current examinations, which were beyond the code required minimum inspections, utilized equipment that allowed the eddy current probe to follow the contours of the CRDM nozzle J-groove weld. Through UT, liquid dye penetrant, visual examinations, and eddy current testing, a total of 24 CRDM nozzles were repaired due to nozzle or J-groove weld indications, with two of these nozzles (nozzles #4 and #67) confirmed to have pressure boundary leakage.

Samples of the boric acid on the RVCH were collected from CRDM nozzle #4 and the east side of the RVCH. Analysis of these samples determined that they had characteristics of leakage during startup or shutdown periods of operation in which significant lithium was present in the RCS. Additionally, short-lived isotopes were identified which could only be present from leakage occurring in the last month of operation. An additional boric acid sample collected from the RVCH flange had characteristics of leakage from earlier in the operating cycle (first half of 2009) due to the absence of short-lived isotopes. All three samples had evidence of thermal conditioning, indicating the leakage occurred during operation, but contained minimal corrosion products. The estimated leak rate to form the deposits found on the RVCH were below the level of detection of the RCS inventory balance conducted to meet Technical Specification Requirements with the plant in operation.

CAUSE OF EVENT:

The direct cause of the unacceptable CRDM nozzle and J-groove weld flaws with RCS pressure boundary leakage is primary water stress corrosion cracking (PWSCC) that resulted in flaws propagating through the CRDM nozzle or through the length of the weld with boric acid leakage onto the reactor vessel closure head. This cause is supported by an independent technical review of the UT examinations, destructive examinations of samples from two CRDM nozzles, and aligns with industry experience for reactor vessel closure heads and CRDM nozzles constructed of Alloy 600 material.

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NARRATIVE

CAUSE OF EVENT: (continued)

The DBNPS reactor vessel closure head was replaced due to the discovery of degradation as documented in DBNPS Licensee Event Report 2002-002. The current DBNPS reactor vessel closure head, which was purchased from the cancelled Midland Unit #2 and placed in service in September 2003, has been in service for three operating cycles. No RCS pressure boundary leakage had been identified during previous bare metal visual examinations of this RVCH.

Operating temperatures of the DBNPS RVCH were previously considered to be equal to the highest RCS hot leg temperature at approximately 606.4°F. Analysis performed following discovery of RCS pressure boundary leakage on March 13, 2010, determined the temperature of the water circulating through the upper region of the RVCH is approximately seven to eight degrees warmer than previously understood. This is due to channeling of warmer water leaving fuel assemblies located below the control rod guide tubes to the RVCH. The core fuel design determines the fuel assemblies' power and subsequent water exit temperature. The RVCH temperature is used to calculate Effective Degradation Years and Re-Inspection Years, which are used in the calculation to determine when the next inspection is required to be performed per regulations.

The Root Cause of the CRDM nozzle cracking and J-groove weld flaws with RCS pressure boundary leakage was a less than adequate perception of the risk of PWSCC susceptibility with the replacement RVCH resulting in inadequate identification, development, and implementation of interim actions to mitigate degradation prior to replacement with a PWSCC resistant Alloy 690 RVCH.

ANALYSIS OF EVENT:

The bare metal visual inspection did not identify any discernable areas of wastage of the RVCH. The majority of the indications/flaws identified on the nozzles from UT examinations were primarily axial in direction, and the few identified with circumferential aspects were no more than 15 degrees in length. Based on industry evaluation of CRDM nozzle indications, the critical size of a circumferential flaw that could result in a CRDM nozzle ejection is on the order of 330 degrees in length, and axial nozzle cracking is not a credible mechanism for CRDM nozzle ejection. Since the CRDM nozzle flaws observed at the DBNPS were identified by a planned inspection designed to detect those types of flaws, and since the flaws that were detected were well below flaw sizes required for nozzle ejection and there was no discernable head wastage, this issue was of very low safety significance.

Reportability Discussion:

Most of the indications found in the CRDM nozzles were determined to be unacceptable and require repair. Section 3.2.4 of NUREG-1022, Event Reporting Guidelines, identifies that defects in the RCS pressure boundary that cannot be dispositioned as acceptable per ASME Section XI represent a condition that results in the nuclear power plant, including its principal safety barriers, being seriously degraded. These conditions were initially reported to the Nuclear Regulatory Commission per 10 CFR 50.72(b)(3)(ii)(A) on March 13, 2010 at 0445 hours of this condition (reference Event #45764).

As a result of the bare metal visual inspection, evidence of leakage from a CRDM nozzle penetration was observed, which also indicates serious degradation of a principle safety barrier. These conditions were also reported to the Nuclear Regulatory Commission per 10 CFR 50.72(b)(3)(ii)(A) on March 13, 2010 at 1903 hours (also reference Event #45764).

NARRATIVE

ANALYSIS OF EVENT: (continued)

Technical Specification Limiting Condition for Operation 3.4.13 states that RCS operational Leakage shall be limited to no pressure boundary leakage. The evidence of leakage from the CRDM Nozzles indicates the plant operated in a condition prohibited by the Technical Specifications. Therefore, this issue is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B), operation in a condition prohibited by the Technical Specifications, and 10 CFR 50.73(a)(2)(ii)(A), serious degradation of a principle safety barrier. This event does not meet the definition of a Safety System Functional Failure.

CORRECTIVE ACTIONS:

The CRDM nozzles containing indications or evidence of leakage were modified utilizing the inside diameter temper bead (IDTB) welding method to restore the pressure boundary of the degraded nozzles. These activities were conducted in accordance with the 1995 Edition through the 1996 Addenda of ASME Code Section XI, Code Case N-638-1, Code Case N-729-1, and alternative requirements as requested via separate correspondence (letter L-10-099 dated April 1, 2010, and others) from the FirstEnergy Nuclear Operating Company (FENOC) to the NRC, and authorized by the NRC verbally on June 4, 2010 and via letter dated September 20, 2010.

In accordance with the NRC Confirmatory Action Letter issued to the DBNPS on June 23, 2010, the DBNPS will be shut down no later than October 1, 2011, to replace the RVCH. Until replacement of the RVCH, upon reaching Action Level 3 of procedure EN-DP-01171, "Engineering Implementation of the RCS Integrated Leakage Program," the DBNPS shall be shutdown in 30 days if RVCH leakage cannot be ruled out. During subsequent shutdown as part of the containment inspection for RCS leakage, if RVCH leakage cannot be ruled out a bare metal visual examination of the RVCH will be performed per the applicable ASME Code Case and 10 CFR 50.55a(g)(6)(ii)(D).

Lessons learned training on the cause of this event including the less than adequate perception of risk related to the event will be provided to Engineering Support Personnel and Site Supervisors.

PREVIOUS SIMILAR EVENTS

DBNPS LER 2002-002 documented a previous event where RCS pressure boundary leakage occurred due to primary water stress corrosion cracking of CRDM nozzles, which resulted in wastage and degradation of the RVCH. The root cause of the 2002 RVCH degradation was boric acid corrosion due to an inadequate Boric Acid Corrosion Control Program. No discernable wastage of the RVCH was identified for the current event. The RVCH was replaced as a result of the 2002 event with an unused RVCH from the cancelled Midland Unit #2.