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10 CFR 50.73

February 15, 2001

PSLTR: #01-0022

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

Dresden Nuclear Power Station, Unit 3
Facility Operating License No. DPR-25
NRC Docket No. 50-249

Subject: Licensee Event Report 2000-004-01, "Technical Specification Non Compliance due to Primary Containment Inboard and Outboard Feed Water Isolation Valves Exceeding Local Leak Rate Test Allowable Limits"

Enclosed is Licensee Event Report (LER) 2000-004-01, "Technical Specification Non Compliance due to Primary Containment Inboard and Outboard Feed Water Isolation Valves Exceeding Local Leak Rate Test Allowable Limits," for the Dresden Nuclear Power Station (DNPS). This condition is being reported pursuant to 10 CFR 50.73 (a)(2)(ii)(B), which requires the reporting of any operation or condition prohibited by the plant's Technical Specifications. This is a supplement to Revision 0 of LER 2000-004.

The following actions were taken:

The 3-0220-58A valve was repaired utilizing a modification to install a new 20-bolt assembly with an o-ring style seat. The repaired valve successfully passed the LLRT.

The 3-0220-62A valve seat assembly was replaced and tested satisfactorily.

This correspondence contains the following new commitment:

Create and/or revise PMs to replace the seat assembly at least every four refueling outages.

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Any other actions described in the submittal represent intended or planned actions by DNPS. They are described for the NRC's information and are not regulatory commitments.

If you have any questions, please contact Dale Ambler, Dresden Regulatory Assurance Manager at (815) 942-2920 extension, 3800.

Respectfully,

Preston Swafford

Preston Swafford
Site Vice President
Dresden Nuclear Power Station

Enclosure

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the information and Records Management Branch (t-6 f33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office Of Management And Budget, Washington, DC 20503. If 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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TITLE (4)

Technical Specification Non Compliance due to Primary Containment Inboard and Outboard Feed Water Isolation Valves Exceeding Local Leak Rate Test Allowable Limits

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	N/A
09	**18**	**2000**	**2000**	**004**	**01**	**02**	**15**	**2001**	FACILITY NAME N/A	DOCKET NUMBER N/A

OPERATING MODE (9)	**5**	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	**0**		20.2203(a)(i)		20.2203(a)(3)(I)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i))		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)	X	50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard A. Kelly, Regulatory Assurance	**(815) 942-2920 Ext. 2924**

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SJ	ISV	C665	Y					

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

ABSTRACT (Limit to 1400 spaces, i. e., approximately 15 single-spaced typewritten lines) **(16)**

On September 18, 2000, at 2230 hours, with Unit 3 shutdown for Refuel Outage D3R16, the Primary Containment Outboard Feedwater Check Valve 3-0220-62A failed the as found local leak rate test (LLRT) during performance of Dresden Operating Surveillance (DOS) 7000-26, “Local Leak Rate Testing Of Unit 2(3) Feedwater System Valves.” Since this valve is paired in series with Inboard Feedwater Check Valve 3-0220-58A that also failed its LLRT, this placed the unit in a condition prohibited by the Technical Specifications. The safety significance of the leakage through the primary containment isolation valves is considered to be minimal. Both valves were inspected, repaired and tested satisfactorily prior to the completion of D3R16.

The investigation resulted in two root causes. For valve 3-0220-58A, the 18-bolt seat assembly installed was determined to be a poor design for this application. For valve 3-0220-62A, the valve had normal wear creating a gap between the seats. Corrective actions taken included the installation of a new 20-bolt assembly with an o-ring style seat for the 3-0220-58A valve. The repaired valve successfully passed the LLRT. The 3-0220-62A valve seat assembly was replaced and tested satisfactorily. In addition, preventive maintenance tasks will be created or revised to replace the seat assembly at least every four refueling outages.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

PLANT AND SYSTEM IDENTIFICATION:

General Electric – Boiling Water Reactor - 2527 MWt rated core thermal power

Energy Industry Identification System (EIIS) Codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Technical Specification Non Compliance due to Primary Containment Inboard and Outboard Feed Water Isolation Valves Exceeding Local Leak Rate Test Allowable Limits

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3	Event Date: 09-18-2000	Event Time: 22:30
Reactor Mode: 5	Mode Name: Refuel	Power Level: 0
Reactor Coolant System Pressure: 0 psig		

B. DESCRIPTION OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73 (a)(2)(I)(B), which requires the reporting of any operation or condition prohibited by the plant's Technical Specifications (TS). In addition, this LER is also being submitted pursuant to 10 CFR 50.73(a)(2)(ii), 10 CFR 50.73(a)(2)(v), and 10 CFR 50.73(a)(2)(vii).

On September 18, 2000, at 2230 hours, with Unit 3 shutdown for Refuel Outage D3R16, the Primary Containment Outboard Feedwater [SJ] Check Valve 3-0220-62A failed the as found local leak rate test (LLRT) during performance of Dresden Operating Surveillance (DOS) 7000-26, "Local Leak Rate Testing Of Unit 2(3) Feedwater System Valves." The amount of leakage for the 62A valve was found to be undetermined based upon the amount of leakage identified exceeding the capabilities of the LLRT equipment utilized during the performance of the test. Since this valve is paired in series with Inboard Feedwater Check Valve 3-0220-58A that also failed its LLRT with an undetermined leakage rate, this placed the unit in a condition prohibited by the Technical Specifications. The total leakage between the two valves was undetermined resulting in the minimum path leakage exceeding the 0.6La limit in TS 3/4.7.A, "Primary Containment Integrity."

Upon the discovery of the failures, the valves were disassembled and inspected. The results of these inspections were as follows:

The 3-0220-58A valve has had a history of failure during previous LLRTs. The valve was previously changed from a four clamp o-ring style assembly to an 18-bolt gasket style assembly to improve performance. When the valve was opened, it was discovered that there was a minimum 0.002-inch gap between the seats from the 9:00 to 3:00 positions with a maximum 0.003-inch gap identified from about the 12:00 to 1:00 positions.

The 3-0220-62A has not had a history of LLRT failures. Upon disassembly, the gap between the valve seats was found to be less than 0.0015 inches in all areas. Gaps this small have typically resulted in a successful LLRT. There were some scratches noticed on the seats. It was determined that these scratches were not of significant size to have caused leakage in the amount required to have caused an LLRT failure. This valve is an o-ring style assembly with four clamps. The clamps at the 9:00 and 3:00 positions were found to be loose. The clamps at the 12:00 and 6:00 positions were found to be tight. Upon removal of the seat assembly, the o-ring was found in good condition with no sign of leakage.

The valves were repaired and tested with satisfactory results.

C. CAUSE OF EVENT:

The 3-0220-58A valve has had a history of LLRT failures. During D3R14, in an effort to improve performance, an 18-bolt seat assembly with a nickel gasket was installed. It was determined that the 18-bolt design has two weaknesses. First, the gasket creates an uneven surface if not compressed evenly. The hinge ears do not allow for bolting in that area which can cause uneven compression and distortion of the valve seat resulting in a gap between the seats. Second, there is very little material between the bolt hole and the inner diameter of the gasket. Thus, the gasket can blow through at this location. As a result, the root cause for the failure of the 3-0220-58A valve was determined to be that the 18-bolt seat assembly is a poor design that often can result in gaps between the seats. (NRC Cause Code B)

The 3-0220-62A valve did show gaps between the seats of 0.001" to 0.0015". Although two of the four seat clamps were found to be loose during disassembly inspections, there was no sign of leakage through the o-ring sealing area. Therefore, the gaps identified were the only leak paths detected during the disassembly and inspection. Since no failed components were identified that would have caused a gap like this to be created, the root cause for the failure of the 3-0220-62A valve was determined to be normal wear of the seats. (NRC Cause Code X)

D. SAFETY ANALYSIS

Based on the results found during the internal inspections of these valves, the safety significance of the LLRT failures is considered to be minimal based upon the following discussion.

Both the inboard and outboard primary containment feed water check valves were internally inspected after the as-found LLRTs were found to be undetermined. The LLRT equipment is capable of reading up to 100 scfh. Thus, an undetermined amount of leakage is any leakage greater than 100 scfh. Upon disassembly, both of the check valves were found to be in the closed position. Each valve was exercised by lifting the disc by hand. Both check valves stroked smoothly with no indication of binding. The inboard check valve did have a 0.003-inch gap between the seats at the 12:00 to 1:00 positions. This size of gap typically allows a significant amount of leakage. The outboard check valve had a gap between the valve seats that was found to be less than 0.0015 inches in all areas. Based on prior testing at Dresden Nuclear Power Station, a 0.001-inch leak may exceed 100 scfh but is not expected to exceed La leakage. Calculations performed on a gap of this size also support a leakage of approximately one-half of La. The outboard check valve did have scratches on the seating surfaces but they were small and are not believed to add significantly to the estimated leakage. No other leakage paths were found for either of these check valves.

E. CORRECTIVE ACTIONS:

The 3-0220-58A valve was repaired utilizing a modification to install a new 20-bolt assembly with an o-ring style seat. The repaired valve successfully passed the LLRT. (Complete)

The 3-0220-62A valve seat assembly was replaced and tested satisfactorily. (Complete)

Create and/or revise PMs to replace the seat assembly at least every four refueling outages. (ATI 35169 – 29)

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TEXT *(If more space is required, use additional copies of NRC Form 366A) (17)*

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
98-004-00/05000237	Outboard Main Steam Line Isolation Valves 2-203-2B And 2-203-2D As-found Leakage Rates Exceeded Technical Specification Limit
98-004-01/05000237	Supplement to Outboard Main Steam Line Isolation Valves 2-203-2B And 2-203-2D As-found Leakage Rates Exceeded Technical Specification Limit

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Crane Co,	3A FW Header Inboard Drywell Check Valve	973	N/A
Crane Co.	3A FW Header Outboard Drywell Check Valve	973	N/A