Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUN 6 2005

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

Serial No. 05-335
MPS Lic/RWM R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION, UNIT 2
LICENSEE EVENT REPORT 2005-002-00
REACTOR COOLANT SYSTEM PRESSURE BOUNDARY LEAKAGE FROM PWSCC IN A PRESSURIZER HEATER SLEEVE

This letter forwards Licensee Event Report (LER) 2005-002-00, documenting a condition that was discovered at Millstone Power Station Unit 2 on April 10, 2005. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications.

If you have any questions or require additional information, please contact Mr. David W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

J. Alan Price
Site Vice President – Millstone

Attachments: (1)

Commitments made in this letter: None.

IE22

cc: U.S. Nuclear Regulatory Commission
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Mr. S. M. Schneider
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Millstone Power Station

Attachment 1

Millstone Power Station, Unit No. 2

LER 2005-002-00

Dominion Nuclear Connecticut, Inc.

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Reactor Coolant System Pressure Boundary Leakage from PWSCC in a Pressurizer Heater Sleeve

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	10	2005	2005	– 002	– 00	06	06	2005	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David W. Dodson, Supervisor Nuclear Station Licensing	860-447-1791

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

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

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

On April 10, 2005, with the plant in Mode 5 (Cold Shutdown) for refueling outage 16 (2R16), an in-service visual inspection of the pressurizer heater penetrations was being performed. One heater sleeve penetration was found to show indications of minor leakage as evidenced by water and a tacky build up on the outside of the penetration. It is conservatively assumed that the leakage could have existed in Modes 1 through 4. Technical Specification 3.4.6.2 states that in Modes 1 through 4 "Reactor Coolant Leakage shall be limited to no Pressure Boundary Leakage." This event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications.

The cause of this event was a through-wall crack in one pressurizer heater sleeve at nozzle penetration location B-2 from Primary Water Stress Corrosion Cracking (PWSCC) of the Alloy 600 nozzle.

The leakage was identified during visual examinations prior to any significant degradation of the reactor coolant system pressure boundary. The eddy current testing identified one axial through wall indication about 1/2 inch long from the bottom of the J-groove weld. The axial cracking of the extent shown in this heater sleeve would not have created a susceptibility to a catastrophic failure of the nozzle. The overall safety significance of this condition was determined to be minimal. There was no actual impact on the public health and safety due to this condition.

The leaking heater sleeve was repaired during the refueling outage by the use of a Mechanical Nozzle Seal Assembly (MNSA). Long term resolution of PWSCC of Alloy 600 in the pressurizer will be resolved with the replacement of the pressurizer during the next refueling outage.

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

1. Event Description

On April 10, 2005 with the plant in Mode 5 (Cold Shutdown) for refueling outage 16 (2R16), an in-service visual inspection of the Millstone Power Station Unit 2 pressurizer [PZR] heater [EHTR] penetrations [PEN], pressurizer instrument nozzle [NZL] and pressurizer Alloy 82/182 pipe buttwelds was being performed. One heater sleeve [SLV] penetration was found to show indications of minor leakage as evidenced by water and a tacky build up on the outside of the penetration. This leakage was too small to have been detected via normal means (containment particulate radiation monitors [RI] or other leakage monitoring [MON] systems) during cycle 16 operations.

The heater penetrations in Combustion Engineering (CE) designed Nuclear Steam Supply Systems (NSSS) are fabricated from Alloy 600 (Inconel 600) and are joined to the pressurizer using partial penetration J-groove welds. The weld metal is Alloy 82/182, which is the weld metal for Alloy 600. Industry experience for other CE designed NSSSs has shown that these welds and the nozzles are susceptible to Primary Water Stress Corrosion Cracking (PWSCC). The inspection activities are a result of NRC Bulletin 2004-01 and previous plant experience with PWSCC. The inspections were also a part of the Boric Acid Corrosion Control Program that incorporated NRC Bulletins 2002-01 and 2002-02, addressing RCS pressure boundary integrity.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications. Technical Specification 3.4.6.2 states that in Modes 1 through 4 "Reactor Coolant leakage shall be limited to no Pressure Boundary Leakage". From the amount of wetness and tacky build up on the outside of the pressurizer, it is conservatively assumed that the leakage could have existed in Modes 1 through 4.

2. Cause

The cause of this event was a through-wall crack in one pressurizer heater sleeve at nozzle penetration location B-2 from PWSCC of the Alloy 600 nozzle and/or J-weld. This PWSCC allowed primary coolant to leak into the annulus between the pressurizer lower head and the heater sleeve, creating a breach of the reactor coolant pressure boundary. The eddy current inspections performed on this nozzle, and both plant specific and industry experience support the conclusion of PWSCC.

The heater element was removed from the one identified location and an eddy current examination characterized the nature of the flaw. The eddy current testing identified one axial through wall indication of about 1/2 inch long from the bottom of the J-groove weld. The defect was determined to be axial and not cirumferential. The axial cracking of the extent found in this pressurizer heater sleeve would not have created a susceptibility to a catastrophic failure of the nozzle. Additional Non-destructive Examination has measured the thickness of pressurizer lower head adjacent to the leaking nozzle and other non-leaking nozzles and no differences were noted.

3. Assessment of Safety Consequences

The design function of the pressurizer is to maintain reactor coolant system pressure. This is done with a combination of heaters to raise the temperature in the pressurizer and spray valves [V] to lower the temperature in the pressurizer. The heater sleeves are part of the pressure boundary of the pressurizer. A through-wall crack in the heater sleeves creates a leak path in the reactor coolant pressure boundary that is not permitted by Technical Specifications.

This leakage did not represent a safety system functional failure and the actual safety significance of the cracking found in the heater sleeve is low. The crack from PWSCC is very tight. The leakage at this nozzle is too small to be quantified and the leakage rate is well below the allowable rate of 1 gpm for unidentified leakage. The leakage was identified during visual examinations prior to any significant degradation of the reactor coolant system

pressure boundary. The PWSCC found at Millstone Unit 2 is similar to other CE designed NSSSs. The defect was determined by eddy current examination to be axial rather than circumferential. The axial cracking of the extent shown in this pressurizer heater sleeve would not have created a susceptibility to a catastrophic failure of the nozzle. At Millstone Unit 2, the potential worst case situation of a circumferential crack that could result in the complete loss of the penetration would be well below the limiting hole size, less than 3 inches, for a Small Break Loss of Coolant Accident, and is therefore bounded by the current analysis. The overall safety significance of this condition was determined to be minimal. There was no actual impact on the public health and safety due to this condition.

4. Corrective Action

The leaking heater sleeve was repaired during the refueling outage by the use of a Mechanical Nozzle Seal Assembly (MNSA). The MNSA is a mechanical device with a mechanical seal that encapsulates the pressurizer heater sleeve to prevent leakage in the event cracks develop in the sleeve or J-groove weld due to PWSCC of the Alloy 600 material. The MNSA also acts as a restraint to prevent a heater sleeve nozzle from being ejected from the vessel in the event of a complete circumferential failure of the nozzle.

The use of the MNSA for this repair has been approved, as documented in the Nuclear Regulatory Commission's Safety Evaluation Report, "Safety Evaluation of Relief Request RR-89-43, Temporary Installation of Mechanical Nozzle Seal Assemblies on Pressurizer Heater Penetration Nozzles," dated October 28, 2003 (ADAMS Accession No. ML032690807).

Long term resolution of PWSCC of Alloy 600 in the pressurizer will be resolved with the replacement of the pressurizer during the next refueling outage (2R17). The replacement pressurizer will not use Alloy 600 for any pressure boundary component or weld. The corrective actions associated with this condition are being addressed in accordance with the Millstone Corrective Action Program.

5. Previous Occurrences

The PWSCC of the Alloy 600 heater sleeves at Millstone Unit 2 is similar to the PWSCC that was found in the pressurizer heater nozzles and Control Element/Rod Drive Mechanisms (CEDMS/CRDMS) at various plants including Millstone Unit 2. The materials of construction and the design with the partial penetration J-groove welds for the pressurizer heater sleeves are the same as for the CEDM/CRDM and Incore Instrumentation (ICI) nozzles. During refueling outage 2R14, Millstone Unit 2 found three CEDMs with shallow cracking below the J-groove weld with ultrasonic inspection techniques and two pressurizer heater nozzles with through wall cracks with visual inspection. During refueling outage 2R15, Millstone Unit 2 found 11 CEDMs with shallow cracking below the J-groove weld with ultrasonic inspection techniques and two pressurizer heater nozzles with through wall cracks with visual inspection. All the cracked pressurizer heater sleeve nozzles were repaired with MNSAs. The indications in three CEDMs were removed by grinding. The other CEDMs were repaired with a partial penetration nozzle replacement technique. In the most recent refueling outage (2R16), a new reactor vessel closure head was installed that uses materials less susceptible to PWSCC.

LER 2002-001-00, Two Reactor Coolant System Pressure Boundary Leakage Events, dated April 16, 2002:
- heater sleeve nozzle penetration locations A-1 and C-4.

LER 2003-004-00, Reactor Coolant System Pressure Boundary Leakage Event, dated December 3, 2003:
- heater sleeve nozzle penetration locations F-4 and C-3.

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