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

Dominion

APR 21 2006

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

Serial No. 06-284
MPS Lic/WEB R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
LICENSEE EVENT REPORT 2006-002-00
MANUAL REACTOR TRIP DUE TO TRIP OF BOTH FEED PUMPS FOLLOWING A LOSS OF INSTRUMENT AIR

This letter forwards Licensee Event Report (LER) 2006-002-00, documenting an event that occurred at Millstone Power Station Unit 2 on February 23, 2006. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in manual actuation of the Reactor Protection System (RPS).

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

IE22

Attachments: (1)

Commitments made in this letter: None.

cc: U.S. Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. V. Nerses
Senior Project Manager
U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
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Rockville, MD 20852-2738

Mr. S. M. Schneider
NRC Senior Resident Inspector
Millstone Power Station

Attachment 1

Millstone Power Station Unit 2
LER 2006-002-00

Millstone Power Station Unit 2
Dominion Nuclear Connecticut, Inc. (DNC)

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	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 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 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
Manual Reactor Trip Due To Trip Of Both Feed Pumps Following A Loss Of Instrument Air

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
02	23	2006	2006	002	00	04	21	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	1
10. POWER LEVEL	100

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

	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)	X	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)		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)		

12. LICENSEE CONTACT FOR THIS LER

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

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

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

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

On February 23, 2006, with the plant in Mode 1 and 100% power, a Manual Reactor Trip was initiated due to a trip of both feed pumps following a loss of Instrument Air. Mechanics were attempting to replace a pipe clamp on a two-inch copper Instrument Air header on Millstone Power Station Unit 2 when an inadequately soldered joint failed on a ½ inch tee connection. The ½ inch line separated from the header and resulted in rapidly lowering Instrument Air pressure in a portion of the turbine building and shutting of an excess flow check valve, as designed. Numerous air operated valves shifted to their loss-of-air failure position. Feedwater heater high-level dump valves opened causing a reduction of heater drain flow and loss of suction pressure to the steam generator feed pumps (SGFP). Both SGFPs tripped and a manual reactor trip was initiated. The Auxiliary Feedwater System actuated as expected, however, the Number 1 Steam Generator AFW Regulating Valve went to its failed position (i.e. full open) allowing excessive flow to the Number 1 Steam Generator, which contributed to a greater than expected, cool down of the RCS. An operator was dispatched to take manual control of the regulating valve at which point RCS temperature was restored to the normal post-trip band of 530-535° F. All other safety systems functioned as designed. The plant was stabilized in Mode 3 at normal operating temperature and pressure.

This event/condition is being reported pursuant to 50.73(a)(2)(iv)(A) as an event that resulted in the manual actuation of the Reactor Protection System as well as the automatic actuation of the Auxiliary Feedwater System.

LICENSEE EVENT REPORT (LER)

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

1. Event Description

On February 23, 2006, with the plant in Mode 1 and 100% power, a Manual Reactor Trip was initiated following an Instrument Air (IA) [LD] leak that occurred while replacing a pipe clamp on a two-inch copper Instrument Air header in the Turbine Building. An inadequately soldered joint failed on a ½ inch tee connection from a two-inch Instrument Air line that resulted in rapidly lowering Instrument Air pressure that caused the excess flow check valve to shut. Numerous air operated valves shifted to their loss-of-air position. Feedwater [SJ] heater high-level dump valves opened causing a reduction of heater drain flow and a loss of suction pressure to the Steam Generator [SB] Feed Pumps (SGFP). Both SGFPs tripped and a manual reactor trip was initiated. Non-Vital 120VAC Regulated AC panels VR11 and VR21 shifted to backup power supplies as expected due to the transfer of station power from the Normal Station Service Transformer (NSST) to the Reserve Station Service Transformer (RSST) following a reactor trip. The momentary loss of power to VR11 during this transfer resulted in a loss of letdown and indication of Pressurizer Power Operated Relief Valve (PORV) and Main Steam Safety Valve (MSSV) position changes. Operators subsequently restored letdown and confirmed no actuation of either PORV's or MSSV's had occurred during the event.

Following the reactor trip, Control Element Assembly Position Display System (CEAPDS) indicated CEA 7 was not fully inserted and the Core Mimic indicated CEA 44 was not fully inserted. Upon further review, it was confirmed that both CEA 7 and 44 had fully inserted and the indication anomalies were due to reed switch indication behavior. Additionally following the trip, the plant experienced an abnormal cool down to 526° F in part due to excessive AFW flow to the Number 1 Steam Generator. An operator was dispatched to take manual control of the regulating valve at which point RCS temperature was restored to the normal post-trip band of 530-535° F. It was subsequently determined that the Number 1 Steam Generator AFW Regulating Valve regulator was incorrectly set. This resulted in the Auxiliary Feed Regulating Valve going to its failed position (i.e., full open). All other safety systems functioned as designed.

The plant was stabilized in Mode 3 at normal operating temperature and pressure.

This event/condition is being reported pursuant to 50.73(a)(2)(iv)(A) as an event that resulted in the manual actuation of the Reactor Protection System as well as the automatic actuation of the Auxiliary Feedwater System.

2. Cause

The cause of this event was determined to be an already weakened solder joint which was disturbed while attempting to repair an incorrectly installed clamp, not designed for Instrument Air piping. This in turn, resulted in a ½ inch copper line separating from a tee connection causing a partial loss of the Instrument Air System in the Turbine Building.

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

3. Assessment of Safety Consequences

There were no adverse consequences as a result of this event. The risk of the event was determined to be no greater than that of a manual reactor trip with a loss of the Main Feedwater. No loss of a safety function occurred. The Auxiliary Feedwater remained available but was initially supplying excessive flow to the steam generators. This was the result of the AFW Regulating Valve failing in the fully open position caused by an incorrect setting on the backup air regulator. After the setting was adjusted, the AFW flow control was restored and the RCS cool down rate returned to normal. In summary, the trip is considered to be of low safety significance.

4. Corrective Actions

An investigation was conducted and appropriate corrective actions are being addressed in accordance with the Millstone Corrective Action Program.

The corrective actions to prevent recurrence of this condition are:

- Training Review Board to determine necessary training for field workers;
- Instruct field workers to perform visual inspection of piping and to "snoop" (leak test) soldered joints within a close proximity prior to performing physical work on the Instrument Air system and;
- Inspect a sample of Instrument Air Piping and supports for proper components, installation and spacing.

5. Previous Occurrences

No previous similar events/conditions were identified.

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