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AUG 10 2015

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

Serial No.	15-403
MPS Lic/LES	R0
Docket No.	50-336
License No.	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
LICENSEE EVENT REPORT 2015-002-00
DEGRADED EMERGENCY CORE COOLING SYSTEM CHECK VALVE

This letter forwards Licensee Event Report (LER) 2015-002-00 documenting an event at Millstone Power Station Unit 2 on June 11, 2015. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(v)(C). A supplemental LER is planned to be submitted by December 11, 2015.

If you have any questions or require additional information, please contact Mr. Thomas G. Cleary at (860) 447-1791 x3232.

Sincerely,

Matt Adams
Plant Manager – Millstone

Attachments: 1

Commitments made in this letter: None

IE22

cc: U.S. Nuclear Regulatory Commission
Region I
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Millstone Power Station

ATTACHMENT

LICENSEE EVENT REPORT 2015-002-00
DEGRADED EMERGENCY CORE COOLING SYSTEM CHECK VALVE

MILLSTONE POWER STATION UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC.

NRC FORM 366 (02-2014)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

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4. TITLE

Degraded Emergency Core Cooling System Check Valve

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
06	11	2015	2015 -	002 -	00	08	10	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 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)
100	☐ 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
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Thomas G. Cleary, Supervisor Nuclear Station Licensing	(860) 444-3232

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

CAUSE					CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☒ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☐ NO		12	11	2015

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

On June 11, 2015, while Millstone Power Station Unit 2 (MPS2) was in MODE 1 operating at 100 percent power, Engineering identified that due to a degraded check valve, the post-accident radioactivity release rates assumed in the FSAR could be affected. While performing 'B' High Pressure Safety Injection (HPSI) pump in-service testing, the measured flow was lower than expected. Because both trains of HPSI, Low Pressure Safety Injection (LPSI) pumps, and Containment Spray (CS) pumps share a common minimum flow recirculation line back to the Refueling Water Storage Tank, back-leakage through one of the idle pump's recirculation check valves was postulated as the cause of the observed drop in recirculation flow. Troubleshooting was performed, and it was determined that the minimum flow check valve associated with the 'A' CS pump, 2-CS-6A, was back-leaking. The associated minimum flow isolation valve was closed to eliminate the flow path. Back-leakage of the minimum flow recirculation check valves on the HPSI, LPSI, and CS pumps was not previously considered in radiological release analysis. This leakage has the potential to adversely affect calculated post–Loss of Coolant Accident recirculation phase radioactivity release rates under some postulated scenarios.

vent is being reported as an event or condition that could have prevented fulfillment of a safety function to control the release of radioactive material under 10 CFR 50.73(a)(2)(v)(C). The cause of the event was a failed open check valve. As a corrective action, the failed check valve 2-CS-6A has been repaired.

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NARRATIVE

1. EVENT DESCRIPTION:

On June 11, 2015, while Millstone Power Station Unit 2 (MPS2) was in MODE 1 operating at 100 percent power, Engineering identified that due to a degraded check valve, the post-accident radioactivity release rates assumed in the FSAR could be affected. While performing ‘B’ High Pressure Safety Injection (HPSI) pump in-service testing, the measured flow was lower than expected. Because both trains of HPSI, Low Pressure Safety Injection (LPSI) pumps, and Containment Spray (CS) pumps share a common minimum flow recirculation line back to the Refueling Water Storage Tank (RWST), back-leakage through one of the idle pumps recirculation check valves was postulated as the cause of the observed drop in flow. Troubleshooting was performed, and it was determined that a degraded minimum flow check valve associated with the ‘A’ CS pump (2-CS-6A) was back-leaking into the “A” train suction line and flowing to the RWST through 2-CS-14A (RWST suction check valve) and normally open 2-CS-13.1A (RWST suction line isolation). The associated minimum flow isolation valve for 2-CS-6A was closed to eliminate the path and the valve was repaired. 2-CS-14A permitted back flow because the conditions during the test provided insufficient back flow to adequately seat it.

Subsequently, engineering evaluation determined that during a loss of power to one facility or train of the Emergency Core Cooling System (ECCS), this back-leakage flow path could result in more leakage to the RWST than had been previously considered in the accident analysis. This leakage has the potential to adversely affect calculated post–Loss of Coolant Accident recirculation phase radioactivity release rates under some postulated scenarios. Specifically, without power on the affected (i.e.: back-leaking) train, the CS discharge flowpath would not automatically open resulting in leakage to the RWST.

Prior to this discovery, the last time that the ‘A’ containment spray pump was run was on April 15, 2015, which would have opened 2-CS-6A. On this basis, it was concluded that this condition existed from April 15, 2015 to June 11, 2015. During this period, the ‘A’ Emergency Diesel Generator (EDG) was inoperable 4 times for a total of approximately 25 hours. However, all 4 times were for surveillances. At no time during this period was any maintenance done on the ‘A’ EDG. It is judged that, between accident initiation and commencement of sump recirculation (approximately 2 hours with only one train available), the EDG could have been made available.

This event is being reported as an event or condition that could have prevented fulfillment of a safety function to control the release of radioactive material under 10 CFR 50.73(a)(2)(v)(C). Note: The initial non-emergency report (#51149) of this issue on June 11, 2015 was subsequently updated on July 10, 2015. An additional unidentified release path from the original design of the plant was reported to the NRC as a follow-up notification on July 10, 2015.

BACKGROUND

The MPS2 Containment Spray (CS) system functions as an engineered safety feature to limit containment pressure and temperature after a loss-of-coolant accident (LOCA) and Main Steam Line Break (MSLB) accident, and thereby reduce the potential for leakage of airborne radioactivity to the outside environment. A minimum flow recirculation line is included in the design for recirculating water from the outlet of the pump to the RWST. With the CS system operating during a design basis accident, a small portion of the CS pump discharge flow recirculates to the RWST during the injection

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NARRATIVE

phase; however, the recirculation line is isolated from the RWST when transferring to sump recirculation. All seven ECCS/CS pumps (2 LPSI, 2 CS, and 3 HPSI) have minimum flow recirculation lines that tie into one common header to the RWST. Exhibit A attached to this LER is a sketch depicting the configuration.

Evaluation of this failure identified that during a Small Break Loss of Coolant Accident (SBLOCA), concurrent with a loss of power to the "A" train associated with the leaking 2-CS-6A, the operating (opposite) train HPSI pump would pressurize the common recirculation header during the recirculation phase of the accident as was described above. In the operator response to this postulated event when the 2-CS-6A leaks to the RWST, suction 2-CS-13.1A is manually isolated terminating the release to the RWST. However, it was additionally identified that the "A" train suction header would pressurize because of the continued back leakage and none of the "A" train pumps flowing due to loss of power and no open flow path (to containment spray or RWST). This pressurization would continue, exceeding the design pressure of the suction header (60 psig) ultimately reaching 500 psig, the lift setpoint for the "A" train shutdown cooling heat exchanger relief valve downstream of the "A" CS pump and the relief valve on the common LPSI discharge header. These relief valves discharge to the EDST (Emergency Drain Sump Tank) which is located and vented outside the filtered ventilation boundary. Upon emptying of the RWST and initiation of sump recirculation, the described back-leakage flow path would contain sump fluid. This additional unidentified release path from the original design of the plant was reported to the NRC as a follow-up notification on July 10, 2015.

For this condition (2-CS-6A back-leakage) to result in significant radiological consequences, the following accident sequence must occur:

- 2-CS-6A back-leakage
- SBLOCA
- Loss of Offsite Power (LOOP)
- Loss of the "A" train of onsite electrical power
- SBLOCA break size that results in RCS re-pressurization and sustained RCS pressures above 500 psig, such that HPSI would continue to pressurize the recirculation header above the relief valve setpoints
- Fuel damage

While the discussion above relates to 2-CS-6A and the "A" train, back-leakage from any of the minimum flow check valves and a concurrent loss of power to that train would have a similar outcome.

2. CAUSE:

The cause of the event was a failed-open check valve. Further analysis indicates a leaking check valve could also cause the problem described in the Assessment of Safety Consequences. Leakage through these check valves was not considered in the FSAR.

3. ASSESSMENT OF SAFETY CONSEQUENCES:

A preliminary assessment reveals that if a recirculation line check valve fails open or leaks, the following conditions could occur:

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NARRATIVE

1. An additional recirculation flow path exists for the running pumps back to the non-running pumps' respective suction lines via the common minimum flow recirculation line.
2. During a loss of power to one facility or train of the ECCS/CS systems, more leakage to the RWST than had been previously analyzed could result.
3. Under certain small break LOCA conditions, ECCS/CS non-operating train suction piping could be pressurized significantly above its design pressure, and relief valves could lift, sending sump recirculation water to the Equipment Drains Sump Tank (EDST), and result in an unfiltered release of radioactivity, and associated reduction in sump inventory.
4. Low pressure piping portions of ECCS systems could be caused to yield for the failed open check valve case, but not for the leaking check valve case, although would be unlikely to rupture.

This scenario must be evaluated using modified methods. Millstone typically evaluates design basis radiological consequences based on a LBLOCA and associated fuel damage and a SBLOCA model does not exist. Therefore, a preliminary sensitivity model has been developed using best-estimate assumptions (i.e., a gap release source term, 1% iodine partitioning in the EDST, and measured control room filter efficiency and unfiltered leakage) for application in the small break LOCA scenario. Preliminary results show that control room operator dose would remain within the regulatory limits of 10 CFR 50.67 if the release from 2-CS-6A is terminated within 24 hours, and offsite doses would also remain within the regulatory limits of 10 CFR 50.67 (25 REM TEDE offsite dose and 5 REM TEDE Control Room dose) under these assumptions.

The evaluation of the dose consequences is continuing for this event, and a supplement to the LER is planned once the evaluation is completed.

4. CORRECTIVE ACTION:

The failed check valve 2-CS-6A has been repaired. Additional corrective actions are being taken in accordance with the station's corrective action program.

5. PREVIOUS OCCURRENCES:

- None

6. Energy Industry Identification System (EIIS) codes:

- Pump – P
- Valve – V
- Tank – T
- Containment Spray – CS
- Safety Injection – SI

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