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

Dominion

OCT 21 2002

Docket No. 50-423
B18780

RE: 10 CFR 50.73(a)(2)(ii)(B)

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

Millstone Power Station, Unit No. 3
Licensee Event Report 2002-003-01
Inadequate Validation of Fire Safe Shutdown Analysis Assumptions

This letter forwards Licensee Event Report (LER) 2002-003-01, documenting an event that occurred at Millstone Power Station, Unit No. 3, on August 22, 2002. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B).

There are no regulatory commitments contained within this letter.

Should you have any questions regarding this submittal, please contact Mr. David W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Attachment (1): LER 2002-003-01

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
NRC Senior Resident Inspector, Millstone

IE22

Docket No. 50-423
B18780

Attachment 1

Millstone Power Station, Unit No. 3

LER 2002-003-01

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

LICENSEE EVENT REPORT (LER)

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

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Power Station - Unit 3	05000423	1 OF 3

TITLE (4)
Inadequate Validation of Fire Safe Shutdown Analysis Assumptions

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
08	22	2002	2002	– 003 –	01	10	21	2002	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	95

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

	20 2201(b)		20 2203(a)(3)(ii)	X	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
	20 2203(a)(2)(iv)		50 73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		in NRC Form 366A
	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)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David W. Dodson, Supervisor-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 26, 2002, with the Unit in Mode 1 at 100% power, it was determined that emergency operating procedure (EOP) directed actions did not provide adequate assurance that power operated relief valve (PORV) function would be disabled within a timeframe that would prevent a fire-induced spurious PORV actuation from depressurizing the reactor coolant system (RCS). During subsequent review, on August 22, 2002, it was determined that the assumed timeframe for mitigation of spurious opening of an atmospheric dump valve (ADV) bypass valve would not maintain pressurizer within acceptable levels as required by fire safe shutdown design basis. A plant response to spurious operation of a PORV is similar to a small break loss of coolant accident and spurious operation of the ADV bypass valve is similar to an excess steam demand event. The ability to effectively diagnose and manage events like these coincident with a design basis fire was not consistent with the fire safe shutdown design basis. These circumstances represent an unanalyzed condition that could significantly degrade plant safety and is being reported pursuant to 10CFR50.73(a)(2)(ii)(B).

This condition is historical in nature dating to the original development of the assumptions used to support the fire safe shutdown analysis. Consequently a root cause evaluation was not performed. The apparent cause is attributed to a failure to properly validate fire safe shutdown analysis assumptions.

Upon discovery, compensatory actions were implemented to minimize risk of fire in the areas of concern. These actions included increased surveillances to verify operability of detection and suppression systems, and to confirm control of transient combustibles and ignition sources in the affected areas.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Power Station - Unit 3	05000423	2002	003	01	2 OF 3

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

1. Event Description

On April 26, 2002, with the Unit in Mode 1 at 100% power, it was determined that emergency operating procedure (EOP) directed actions did not provide adequate assurance that power operated relief valve (PORV) [PCV] function would be disabled within a timeframe that would prevent a fire-induced spurious PORV actuation from depressurizing the reactor coolant system (RCS) [AB]. During subsequent review, on August 22, 2002, with the unit in Mode 1 at 95% power, it was determined that the assumed timeframe for mitigation of fire-induced spurious opening of a main steam [SB] atmospheric dump valve (ADV) bypass valve would not maintain pressurizer [PZR] within acceptable levels as required by fire safe shutdown design basis. A plant response to spurious operation of a PORV is similar to a small break loss of coolant accident and spurious operation of the ADV bypass valve is similar to an excess steam demand event. The ability to effectively diagnose and manage events like these coincident with a design basis fire was not consistent with the fire safe shutdown design basis. These circumstances represent an unanalyzed condition that could significantly degrade plant safety which is reportable pursuant to 10CFR50.73(a)(2)(ii)(B).

In 1985, it was assumed that 15 minutes would be available to isolate the PORVs in order to establish control of the reactor coolant system pressure boundary as part of a sequence of actions required to establish stable hot standby conditions following a fire in the control room, cable spreading room, or instrument rack room. Local manual operations necessary to achieve hot standby were considered acceptable for the alternate fire safe shutdown design and EOP, provided that the operating staff could support those actions in the prescribed timeframe. Note that no question exists as to the ability of the operating staff to meet the 15 minute PORV isolation criteria. Recent results from a Unit 2 fire safe shutdown analysis prompted review of the Unit 3 control room fire EOP operator actions and response times. The conclusions of this preliminary review determined that the application of the 15 minute timeframe in the EOP would not assure isolation of the PORV prior to RCS depressurization. The subsequent loss of cooling would challenge performance of a safe shutdown at the alternate plant location.

As a result of on-going efforts to validate assumptions used in the fire safe shutdown analysis, it was determined that the early analysis did not model the pressurizer level response to a fire-induced operation of an ADV bypass valve. For this event, 30 minutes was assumed to be sufficient to manually close the valve, however, the pressurizer level would actually decrease below acceptable levels required by the licensing basis in less than 30 minutes.

2. Cause

This condition is historical in nature dating to the original development of the assumptions used to support the fire safe shutdown analysis. Consequently a root cause evaluation was not performed. The apparent cause is attributed to a failure to properly validate fire safe shutdown analysis assumptions. Specifically, the original licensing basis assumption of 15 minutes to disable the PORVs and the 30 minute assumption to close the ADV bypass valve following a reactor trip are invalid.

3. Assessment of Safety Consequences

At the onset of a fire in the control room, cable spreading area, or instrument rack room, the potential exists for hot shorts to cause a PORV to open. Fires in any of the areas listed also have the potential to degrade shutdown capability from the control room. If control room functions became substantially degraded, the control room would be abandoned and plant shutdown would be accomplished at various alternate shutdown locations. The limited set of plant indications that are available at these alternate locations may not support timely diagnosis and mitigation of a PORV actuation. Additionally, procedures for shutdown from these alternate control locations do not currently anticipate and address the significant challenges associated with maintaining adequate core cooling with a depressurized RCS. Similarly, a hot short could cause the motor operated ADV bypass valve to open and create a plant response that is not in accordance with the licensing basis which requires pressurizer level to be maintained within the indicating range. In this event, the allowed time to manually close the ADV bypass valve is too long, and significant RCS shrinkage (decreasing pressurizer level) would occur before the valve could be fully closed.

The safety significance of this condition is considered low. It should be noted that the original assumptions (15 minutes, 30 minutes) are judged to be acceptable for slowly developing fires where the event continues to be managed from the control room for an extended period of time. The fire scenario of concern in this case is a rapidly developing fire of significant magnitude which forces an evacuation of the control room shortly after detection and commencement of safe shutdown from alternate plant locations. Based on the very low probability of occurrence of this type of fire and based on the availability of fire detection and suppression systems in the specified areas, the safety significance of this condition is considered low.

4. Corrective Action

Upon discovery of this condition, compensatory actions to minimize risk of fire in the areas of concern were implemented including increased surveillances to verify operability of detection and suppression systems, and to confirm control of transient combustibles and ignition sources in affected areas traversed by PORV and ADV bypass valve control circuits.

Investigation and resolution of the issues presented in this LER are being addressed in accordance with the Millstone Corrective Action Program. Review of additional fire safe shutdown analysis assumptions is ongoing.

5. Previous Occurrences

No previous similar events/conditions were identified.

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