NORTHEAST UTILITIES

The Connecticut Light And Power Company
Western Massachusetts Electric Company
Holyoke Water Power Company
Northeast Utilities Service Company
Northeast Nuclear Energy Company

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Re: 10CFR50.73(a)(2)(ii)
January 17, 1992
MP-92-68

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Reference: Facility Operating License No. DPR-65
Docket No. 50-336
Licensee Event Report 91-010-01

Gentlemen:

This letter forwards update Licensee Event Report 91-010-01.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen Scace

Stephen E. Scace
Director, Millstone Station

SES/RAB:ijs

Attachment: LER 91-010-01

cc: T. T. Martin, Region I Administrator
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2

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NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Millstone Nuclear Power Station Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 3 3 6

PAGE (3): 1 OF 0 6

TITLE (4): Reanalysis of Main Steam Line Break Exceed Containment Design 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 NAMES	DOCKET NUMBER(S)
1 0	1 8	9 1	9 1	0 1 0	0 1	0 1	1 7	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME: Robert A. Borchert, Unit 2 Reactor Engineer, Ext. 4418

TELEPHONE NUMBER: AREA CODE 2 0 3 | 4 4 7 - 1 7 9 1

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

X YES (If yes, complete EXPECTED SUBMISSION DATE) — NO

EXPECTED SUBMISSION DATE (15): MONTH 1 2 DAY 3 1 YEAR 9 2

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

On October 18, 1991, at 1305 hours, with the plant in Mode 1 at 100% power, a reportability determination was made concerning a reanalysis of the main steam line break event inside the containment. The reanalysis has confirmed that the assumptions made for the existing (1979) main steam line breaker analysis were non-conservative with respect to power level, break size, and single active failure. Using more restrictive assumptions, design limits for containment pressure and temperature could be exceeded. NNECo determined that this condition was reportable as a condition outside the design basis of the plant. An Immediate Report was made to the NRC, and the unit immediately commenced an orderly downpower to approximately 3% power (Mode 2). The existing main steam line breaker analysis is acceptable for Mode 2 operation.

A Justification for Continued Operation (JCO) was developed to allow the unit to return to power operation by stationing a dedicated reactor operator to close the main feedwater block valves following any reactor trip. This JCO documents the basis for reasonable assurance that, with the actions of a dedicated operator, containment pressure will remain below the design basis value for all postulated main steam line break events. The unit was returned to power operation on October 22, 1991.

Short term corrective hardware changes to automatically close the main feedwater block valves given a Containment Isolation Actuation Signal (CIAS) have been installed and tested. As stated in the JCO, these hardware changes eliminate the necessity to station a dedicated reactor operator to close the main feedwater block valves following a reactor trip. Due to the unavailability of several long lead time qualified replacement components, it will not be possible to perform all of the previously planned permanent hardware changes during the 1992 refueling outage. Changes for which hardware is available will be installed during the 1992 refueling outage. The remaining long term hardware changes will be completed during the 1994 refueling outage.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. Description of Event

On October 18, 1991, at 1305 hours, with the plant in Mode 1 at 100% power, a reportability determination was made concerning a reanalysis of the main steam line break event inside the containment. These re-analyses has shown that the assumptions made for the existing (1979) main steam line break analysis were non-conservative with respect to power level, break size, and single active failure. Using more restrictive assumptions, design limits for containment pressure and temperature could be exceeded.

The existing (1979) main steam line break analysis assumes a postulated double-ended (6.3 ft^2) break of the main steam line between the steam generator outlet and the steam line flow restrictor at hot zero power, with the worst single active failure being a failure of a diesel generator and the resultant loss of one-half of the emergency safeguards features which reduce containment pressure (1 containment spray pump and 2 containment air recirculation fans). The peak containment pressure and temperature for this analysis is predicted to be 47 psig and 274°F.

It has been determined that the limiting containment pressure and temperature are attained by postulating a double-ended break of the main steam line between the steam generator outlet and the steam line flow restrictor at full power, with the single active failure being a failure of the main feedwater regulating valve of the affected steam generator to close. This analysis also assumed operator actions to secure feedwater to the affected steam generator at 10 minutes following the reactor trip. The peak containment pressure and temperature for this analysis is predicted to be 92 psig and 427°F. These results are beyond the containment design pressure and temperature of 54 psig and 289°F.

An immediate report was made to the NRC and the unit immediately commenced an orderly downpower to approximately 3% power (Mode 2) by plant operators. The existing main steam line break analysis is acceptable for Mode 2 operation. No automatic or manual safety systems were required to respond during this event.

II. Cause of Event

The cause of the event has been determined to be an incorrect assumption, made in the FSAR analysis, that the limiting condition, for the containment response due to a Main Steam Line Break (MSLB), was hot zero power. This incorrect assumption was based upon the judgement that at hot zero power the steam generators contain the largest inventory of hot water and thus, resulted in the largest discharge to the containment. However, recent MSLB sensitivity studies have shown that this assumption is not limiting.

As a result of the planned steam generator replacement, the containment response due to a MSLB was being reviewed. In order to assess the impact of the new steam generators, the current analysis results were used to benchmark new steam generator and containment models. During the benchmarking of the current analysis modeling the current steam generators, it was discovered that the hot zero power condition was not limiting.

The peak containment temperature is highly dependent on the moisture carryover that occurs from the break. Moisture carryover is important in that with a significant moisture carryover the containment temperature will be limited to the saturation temperature corresponding to the containment pressure. If no moisture carryover occurs and pure steam is discharged, superheating will occur and containment temperature will not be limited to the saturation temperature.

At hot zero power, the large steam generator inventory will assure that moisture carryover will occur for most break sizes. However, at full power, with the reduced inventory, moisture carryover is not predicted to occur. Thus, for peak containment temperature, the limiting condition would be full power.

Further, the limiting single failure is dependent upon the power level. At hot zero power, the main feedwater regulating valve will be closed at the initiation of the event and would remain closed throughout the transient. Thus, at zero power, the main feedwater regulating valve is not subject to a failed open condition. However, at full power, the main feedwater regulating valve would be open at the initiation of the event and thus would be subject to a failed open condition. With a main feedwater regulating valve failure, the feedwater addition would more than offset the difference in initial inventory between full power and hot full power. Thus, the limiting condition for maximum mass discharge to the containment would be full power with a failure of the main feedwater regulating valve.

These factors were not taken into account in performing the FSAR analysis. Further, they were not explicitly taken into account in the MSLB analysis performed to support the TMI action plan item to implement an automatic system for initiation of auxiliary feedwater nor the response the NRC Inspection and Enforcement Bulletin 80-04 where additional MSLB spectrum studies were requested.

In response to an NRC request for information on automatic initiation of the auxiliary feedwater system made on December 21, 1979, the design basis steamline break analysis was reevaluated. In the analysis the additional mass releases to the containment due to auxiliary feedwater addition were added to the FSAR case and shown to have no impact on the peak containment pressure and temperature. Since this study was aimed at only assessing the impact of the new automatic initiation system, the original FSAR assumptions were not reevaluated. This was supported by evaluations done by the NSSS vendor Combustion Engineering. This analysis was submitted to the NRC on January 27, 1980. Since the information requested in the I&E Bulletin 80-04 issued in February 1980 was very similar to the request made in December 1979, it was assumed that this analysis was also sufficient to response to the Bulletin. Therefore, no new analysis was performed for the Bulletin. A Safety Evaluation Report was received from the NRC on October 7, 1982. The non-conservative assumptions were not discovered until the MSLB was reviewed to evaluate the impact of the planned steam generator replacement.

It should be recognized that in determining the cause of this event, reliance has been placed upon the available documentation for the analysis and evaluations performed in the 1974-1980 time period. Because these evaluations were performed over ten years ago, not all of the documentation has been retrieved. However, from the documentation that we have retrieved, we believe we have been able to reconstruct the logic used to justify the previous submittal and have determined the root cause of the event.

III. Analysis of Event

This event is being reported in accordance with 10CFR50.73(a)(2)(ii)B), which requires the reporting of any event or condition that results in the nuclear power plant being in a condition outside the design basis of the plant.

The safety consequences of this event are the potential overpressurization of the containment with subsequent damage to the containment structure and safety related equipment required for safe shutdown of the plant from a postulated MSLB event. The safety consequences are minimal, however, upon consideration of containment design margins, safety related equipment qualifications, and standard post trip operator actions.

In considering the safety consequences of this event the following items were addressed:

(a) **Containment Structural Integrity**. The Millstone 2 containment consists of a prestressed, reinforced concrete cylinder and dome connected to an supported by a massive reinforced concrete foundation slab. The containment was designed for an internal pressure of 54 psig, and was tested to 62 psig during the structural integrity test. The working stress design method was used to design the containment structure for various load cases, including the case of a design pressure of 54 psig. The containment structure was checked for factored loads and load combinations, including the case with a 1.5 load factor on the design pressure, which corresponds to 81 psig. The code requires that "strength be adequate to support the factored loads and that servicability of the structure at the service load level be assured." (ACI-318-71 Commentary Section 9.1.1)

The ultimate capacity of containments has been studied and documented by many sources recently. In general, the anticipated ultimate capacity of a containment structure has been found to be 2 to 2.5 times design pressure. NUREG-1150 entitled "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," studies the ultimate capacity of typical containment structures. Included in this study was Zion, which is a prestressed containment, with a design pressure of 47 psig. The evaluation determined that a lower bound on the ultimate capacity was around 100 psig (a factor of 2). These detailed studies have taken into account material strengths being higher than assumed, code allowables being conservative, as well as a detailed evaluation of structural behavior during beyond design basis events. A similar detailed study has not been performed for Millstone 2, but the same factors which contribute to a lower bound ultimate capacity of 2 to 2.5 times design exist in the Millstone 2 containment structure. This evaluation of studies relative to ultimate capacity further substantiates that the containment can support the factored load case and beyond. These postulated load cases are beyond the design basis of the containment structure, but within the overall load carrying capability of the structure.

(b) **Equipment Qualification**. The electrical equipment in the containment required for safe shutdown following all MSLB events has been qualified to 10CFR50.49 requirements with temperatures ranging from 324°F to 448°F and pressures ranging from 70 to 127 psig. The original containment qualification profile for this equipment was based on a LOCA event with a maximum temperature of 280°F and pressure of 54 psig.

For a full power MLSB with no automatic feedwater isolation and no operator actions to isolate feedwater for 10 minutes, the predicted peak containment pressure and temperature is 93 psig and 427°F. Although this temperature and pressure would have exceeded the qualification of required equipment, we have determined that equipment required to be available to mitigate this event would have remained operable.

Thermal analysis has shown for the predicted short duration temperature peak at superheated conditions that the surface temperature of safety related equipment will not rise above the saturated temperature of the partial steam pressure in containment. This method of analysis has been presented by the NRC in NREG 0588 paragraph 1.2(5), NUREG 0510 and NUREG 1511. There have been a number of vendor test results which have also demonstrated this phenomenon. The postulated partial steam pressure of the containment during the accident is estimated by subtracting 14.7 psi, i.e., the initial partial atmospheric air pressure, from the absolute pressure of the accident analysis. The resulting maximum steam temperature during the above MSLB accident is thus predicted to be 322°F. At this temperature, all of the required safe shutdown equipment in containment would be qualifiable.

The predicted MSLB accident pressure of 93 psig is slightly higher than the qualification pressure of some of the safe shutdown equipment in containment. The lowest qualification pressure of this equipment is 70 psig. In general, electrical equipment is more sensitive to higher temperature and humidity than higher pressure. The following equipment has been analyzed for operability with this pressure condition:

(1) Westinghouse Containment Air Recirculation (CAR) Fan motors would have been operating prior to the pressure peak. The increase in the CAR fans brake horsepower, due to the higher density air during the accident, has been determined to be within the qualification requirements of the CAR fan motors.

(2) Conax Electrical Penetrations are qualified to pressures in excess of 110 psig. They would have been operable and qualifiable.

(3) ASCO Solenoid valves control various containment isolation valves. Following containment isolation, these valves would be deenergized and would remain deenergized through the pressure peak.

(4) Foxboro and Rosemount pressure transmitters are used to transmit pressure and level signals to the reactor protection system. Although the accuracy of level transmitters may be slightly affected by higher static pressure, they would remain operable.

(5) Weed RTD's monitor RCS temperature. The operability of an RTD is not affected by pressure changes of the magnitude expected by the MSLB.

(6) Inadequate Core Cooling Monitoring System consists of in-core heated junction thermocouples to monitor reactor vessel water level and core exit thermocouples to monitor core outlet temperature. The system does not contain any pressure sensitive components.

In summary, all required safe shutdown equipment in containment has been determined to be operable for all MSLB events.

(c) **Main Feedwater Block Valves.** Evaluation of the original valve specification for the main feedwater block valves, 2-FW-42A and 2-FW-42B, indicated that the valves would have closed in the event of a main feedwater regulating valve failure coincident with a MSLB. Specifically, the worst case differential pressure was determined to be 991 psid, which is significantly less than the original valve specification of 1600 psid. Additionally, plant startup data demonstrated valve closure within 10 seconds as required by the original valve specification.

(d) **Operator Actions.** The Emergency Operating Procedure (EOP) for Standard Post Trip Actions provides instructions for establishing proper main feedwater system configuration following a reactor trip. This configuration requires verification that the main feed regulating valves are closed. The contingency actions state: "IF main feed flow is excessive, THEN (i) Manually close main feed regulating valves or blocking valves and control feedwater flow using bypass, OR (ii) Trip the main feed pumps."

The standard post trip actions are normally completed within 2 to 3 minutes following a reactor trip. While these actions would not have been performed quickly enough to prevent containment pressure from exceeding 54 psig, standard post trip operator actions would have isolated feedwater much faster than the 10 minutes assumed in the re-analyzed MSLB events.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

IV. Corrective Action

A Justification for Continued Operation (JCO) was developed to allow the plant to return to power operation by stationing a dedicated reactor operator to close the main feedwater block valves following any reactor trip. A main steam line break event was analyzed for a double-ended break at full power, with a failure of the main feedwater regulating valve to close on the affected steam generator. This analysis assumed operator action to close the main feedwater block valves within 15 seconds following the reactor trip with a 10 second closure time of the valve. The peak containment pressure and temperature for this case is predicted to be 54 psig and 413°F. This JCO documents NNECo evaluation of operator actions following a reactor trip, feedwater block valve operation under postulated accident conditions, containment structural integrity, and equipment environmental qualification. Diagnostic testing of the motor operated main feedwater block valves was performed in accordance with established procedures developed under the Northeast Utilities Generic Letter 89-10, "Motor Operated Valve Test Program." This JCO provides reasonable assurance that, with the actions of a dedicated operator, the containment pressure will remain below the design basis value for all main steam line break events, although the predicted MSLB temperature peak exceeds the containment qualification temperature of 289°F, all of the required safe shutdown equipment is qualifiable based on a maximum saturated steam temperature of 285°F. Given this justification for continued operation, the unit was returned to power operation on October 22, 1991.

As part of the JCO, short term corrective hardware changes to automatically close the main feedwater block valves given a Containment Isolation Actuation Signal (CIAS) have been installed and tested. As stated in the JCO, these hardware changes eliminate the necessity to station a dedicated reactor operator to close the main feedwater block valves following a reactor trip. Due to the unavailability of several long lead time qualified replacement components (main feedwater block valves and motor control center buckets), it will not be possible to perform all of the planned permanent hardware changes during the 1992 refueling outage. Changes for which hardware is available will be installed during the 1992 refueling outage. The remaining long term hardware changes will be installed during the 1994 refueling outage. Following these changes, the predicted MSLB peak containment pressure and temperature will be equal to or less than 54 psig and 413°F, therefore, the required safety shutdown electrical equipment will remain qualifiable.

A revised response to IE Bulletin 80-04 will be submitted in 1992 to update our previous submittal for containment response and return to power for MSLB events.

V. Additional Information

There were no failed components during this event.

Similar LERs: 77-23, 80-05, 83-07, 85-01 and 86-10

Main Feedwater Regulating Valves

Manufacturer:	Copes-Vulcan
Model:	P-200-12 Angle
Size:	14 inch 900#
EIIS Code:	SJ-LCV-C635

Main Feedwater Block Valves

Manufacturer:	Crane
Model:	L-900 Gate
Size:	18 inch 900#
EIIS Code:	SJ-ISV-C864