Northeast
Nuclear Energy

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The Northeast Utilities System

Donald B. Miller Jr.,
Senior Vice President - Millstone

Re: 10CFR50.73(a)(2)(ii)
May 31, 1994
MP-94-371

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 94-002-02

Gentlemen:

This letter forwards update Licensee Event Report 94-002-02.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Donald B. Miller, Jr.
Senior Vice President - Millstone Station

BY: Harry F. Haynes
Director - Millstone Unit 1

HFH/VRJ:dlr

Attachment: LER 94-002-02

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

IE22
1/1

NRC Form 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES: 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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)
Failure to Meet Acceptable Isolation For Class 1E Protection Instrument Channels

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 NAME	DOCKET NUMBER
										05000
02	04	94	94	– 002 –	02	05	31	94	FACILITY NAME	DOCKET NUMBER 05000

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

OPERATING MODE (9)	1		20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100%		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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	TELEPHONE NUMBER (Include Area Code)
Phil Lutzi, Site Licensing	(203) 437-6585

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)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

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

On February 4, 1994, at approximately 1420 hours, an ongoing engineering review revealed that safety grade (1E) electrical isolation requirements were not satisfied. Two non-safety related, inter-connected, pressure control channels powered from separate Class 1E vital instrument buses via Foxboro Spec 200 instrument rack power supplies were not adequately isolated.

On March 10, 1994, at 0900 hours the review identified inadequate isolation between 1E Spec 200 powered reactor coolant cold leg temperature detector control loops and the non-1E main feedwater control loops.

On April 29, 1994 at 1826, the review identified inadequate isolation between the non-1E reactor regulating system nuclear instrumentation control loops and the 1E Spec 200 Anticipated Transient Without Scram Mitigating System Actuation Circuitry.

The root cause of the events is personnel error due to inadequate interface of design and equipment condition. Design interpretation errors during previous design change processes resulted in the compromising of safety related instrument channels.

Design changes to provide acceptable isolation have been completed.

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

I. Description of Event

At approximately 1420 hours, o February 4, 1994, while in Mode 1 at 100% power, an engineering review of the pressurizer pressure control loop, P-100X & P-100Y wiring, revealed that safety grade channel isolation requirements of IEEE Std 384-1981, "Standard Criteria for Independence of (Class) 1E Equipment and Circuits," were not satisfied. On February 15, 1994, the lack of a qualified isolation device was determined to be a condition outside the design basis of the plant. The two (2) non-Class 1E (non-safety related) pressure control channels are powered from the Class 1E vital instrument buses via the Foxboro Spec 200 instrument rack power supplies. Due to the lack of a qualified isolator between the two pressure control channels and the lack of an in depth analysis to demonstrate that the existing configuration provided acceptable electrical isolation, it was postulated that a fault could potentially propagate between the 2 channels of safety related instruments and compromise their independence from a single failure.

The wiring permits the pressurizer (0-10Vdc) low voltage control signal to pass from cabinet RC30A to RC30B via an isolated current to voltage (H2V) converter card. The common leg of both cabinets became connected due to the wiring configuration of the H2V card (refer to the attached figure). If a catastrophic fault in the form of a voltage surge from the 120VAC feed or +15/-15 Vdc power supply, or a ground fault were to occur, the fault could affect both cabinets. Each cabinet power supply also feeds an independent channel of safety related (Class 1E) reactor protection system (RPS)/Engineered Safety Features Actuation System (ESFAS) instrument loop. Therefore, the wiring deficiency could affect the safety related instrument loops if the fault were to cause failures of the +15/-15 Vdc power supply units in each cabinet.

At approximately 0900 hours on March 10, 1994, during the follow-up instrument channel isolation review, it was determined that inadequate isolation exists between the 1E cold leg temperature (Tcold) inputs into the non-1E feedwater regulating system single element control. The concern is that a fault in the non-safety grade Spec 200 instrument cabinets, RC31A or RC31B, could propagate to, and compromise the safety grade Spec 200 protection instrument cabinets of RC30A-1 or RC30B-1 as a result of the inadequate isolation scheme (refer to the attached figure).

The wiring permits the (0-10Vdc) low voltage Tcold control signal to pass from cabinet RC30A-1 (and RC30B-1) to non-1E cabinet RC31A (and RC31B) via an isolated "EMF to voltage" converter (T2V) card. The common leg of the 1E and non 1E cabinets become connected due to the output wiring of the T2V card (which only provides input wiring isolation).

At approximately 1826 hours on April 29, 1994, during the continuing instrument channel isolation review, it was determined that inadequate isolation existed between the 1E Foxboro Spec 200 Diverse Scram System (DSS) and Automatic Auxiliary Feedwater Initiation (AAFI)-Anticipated Transient Without Scram (ATWS) Mitigating System Actuation Circuitry (AMSAC) and the inputs from the non-1E Reactor Regulating System (RRS) control channel Nuclear Instrumentation (NI) loops. The concern for this event is a fault propagation from the non-safety grade RRS cabinets which could compromise the safety grade Spec 200 instrument cabinets of RC30A or RC30B as a result of the inadequate isolation scheme.

The wiring permits the low voltage RRS cabinet NI signal for nominal 20% power permissive for AMSAC actuation in the event of a high pressurizer pressure signal to pass to 1E cabinet RC30A (and RC30B) via an alarm relay (ALM) card. The common leg of the 1E and non-1E cabinets become connected due to the wiring of the NI input to the coil side of the ALM card. The coil of the ALM card is powered by the Spec 200 power supply.

Cabinets RC30A and RC30A-1 are powered from safety train (Facility Z1) 120 Volt Vital Instrument AC (VIAC) bus VA10. Cabinets RC30B and RC30B-1 are powered from safety train (Facility Z2) 120 Volt VIAC bus VA20.

A failure modes evaluation was conducted which concluded that no credible failures exist which preclude actuation of the protection channels, and thus safe operation of the plant.

II. Cause of Event

The root cause of the events is personnel error due to inadequate interface of design and equipment condition. A design interpretation error during previous design changes resulted in the compromising of safety related instrument channels.

A pressurizer pressure control design change installed QA, Class 1E components, with consideration given to maintaining acceptable isolation, increasing reliability and minimizing the likelihood of exposure to a single failure. However, an error was made in the interpretation of what constitutes a qualified isolator. The H2V card was misapplied as an output isolation device for the pressurizer pressure control channels. The lack of qualified isolation was not identified during the design change review process.

The cold leg reactor coolant system resistance temperature detector (RTD) replacement design change installed QA, Class 1E RTDs with consideration given to maintaining acceptable isolation between the non 1E interface with the single element feedwater control. The lack of qualified output wiring isolation was not identified during the design change review process.

The AMSAC design change installed QA, Class 1E Spec 200 instrumentation with consideration given to maintaining acceptable isolation between the non-1E interface with the NI inputs from RRS. The lack of qualified NI input wiring isolation was not identified during the design change review process.

ATWS is considered a beyond design basis event. A failure as a result of the isolation scheme which would prevent the actuation of both channels of AMSAC is considered a low probability event.

As a result of the mis-applied isolation devices, the design changes potentially compromised reactor protection system (RPS) and engineered safety features (ESF) equipment (refer to Attachment 1 for list). No in depth analysis was performed to justify adequacy of the isolation configuration as required to meet the requirements of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

The isolation concerns were not identified during the design change reviews due to a lack of familiarity with the Spec 200 isolation/design scheme. The design change reviews indicated that isolation was reviewed and the isolation schemes were determined to be acceptable.

The most recent reviews, however, concluded there was no analysis to substantiate the claim that the isolation schemes are adequate. Pursuant to the requirements of IEEE Std. 279-1971, adequate isolation must be demonstrated by either testing or analysis. The Millstone Unit 2 FSAR indicates that the Class 1E reactor protection system instrument channels meet the (physical separation as well as electrical isolation) requirements of IEEE Std. 279-1971.

III. Analysis of Event

The events are reportable pursuant to 10CFR50.73(a)(2)(ii), as conditions outside the design basis of the plant. The February 4th event was initially assessed to be not reportable. The event was determined to be reportable on February 15, 1994.

Immediate notifications were completed pursuant to 10CFR50.72(b)(1)(ii) on February 15, 1994, for the first event, on March 10, 1994, for the second event, and on April 29, 1994 for the third event.

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

The pressurizer pressure channels P-100X & P-100Y are non safety related, non 1E control circuits used to maintain programmed pressurizer pressure. The loops are powered from vital AC buses Facility Z1-VA10 for P-100X and Facility Z2-VA20 for P-100Y. The output of one channel is selected to control pressurizer/reactor coolant pressure. Both channels are recorded in the Control Room at main control board CO3. Outputs are indicated on CO3 and at the hot shutdown panel (C21). High and low pressure alarms are also provided. The associated transmitters, PT-100X & PT-100Y, are environmentally and seismically qualified, and the control loop (Spec 200) components were purchased and installed to meet Class 1E Quality Assurance standards.

Because the design of the P-100X & P-100Y channels resulted in the tying together of two channels of safety related circuits, the condition resulted in the potential reduction of protection channel independence, if a fault were to occur.

The single element feedwater control utilizes steam generator level measurement to balance steam flow and feedwater flow at low power levels. Below 15% feedwater flow, single element control is employed due to the difficulty in measuring flows. Reactor coolant system Tcold measurements are used to provide dynamic compensation for smoother performance during utilization of automatic steam generator level control. The Tcold Spec 200 control loops were purchased and installed to meet Class 1E Quality Assurance standards.

The AMSAC circuitry utilizes the control channel NI signals from RRS for the 20% reactor power permissive to actuate the DSS and AAFI circuits if a high pressurizer pressure signal is present. The AMSAC Spec 200 control loops were purchased and installed to meet Class 1E Quality Assurance Standards.

These configurations are contrary to the Millstone Unit 2 FSAR stipulation that reactor protection system channels meet the isolation requirements of IEEE Std. 279-1971. The instruments affected are listed on Attachment 1.

The events have minimal safety consequences based on the following considerations. The likelihood of a fault which could prevent actuation of protection systems is not considered a credible single failure event. A failure modes evaluation concluded that the credible faults (i.e., a line to line fault, a short or open circuit) would in the worst case result in actuation of the protection channels (which is a conservative action), and would not prevent fulfillment of a safety function.

IV. Corrective Action

No immediate corrective action was required by plant operators in response to the three events. A failure modes evaluation was conducted which concluded that no credible failures exist which would preclude actuation of the protection channels and thus safe operation of the plant. A review of other Spec 200 instrument channels was performed to identify any further problems/common mode failure concerns. The March 10th and April 29th events were identified as a result of this review. The entire review has been completed.

As corrective action, design changes have been completed to provide isolation for the two pressurizer pressure channels, Tcold inputs to single element feedwater control, and NI inputs to AMSAC in order to maintain a design basis consistent with the FSAR description.

The present design change controls and enhanced engineering knowledge with respect to component isolation will prevent recurrence. A copy of this report will be routed to the design engineering groups to increase awareness.

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

V. Additional Information

There have been no similar events with the same root cause and/or sequence of events. The Spec 200 equipment is manufactured by Foxboro. This LER discusses design deficiencies resulting from misapplication of components. There were no component design deficiencies identified during the review of this event.

EIIS Codes

Systems

Reactor Regulating System
AMSAC/AAFI/DSS
Engineered Safety Features Actuation System – JE
Instrument and Uninterruptible Power System – Class 1E – EF
Panels System (Cabinets) – JL
Plant Protection System – JC
SPEC 200 Instrumentation and Controls
Feedwater Control System – SJ

Components

Annunciators – ANN
Auxiliary Relays – RLY
Converter (current to voltage) – CNV
(Current/Voltage) Isolator – IB/EB
Control Panels (Cabinet) – CAB
Resistance Temperature Detector – DET

Manufacturer

Foxboro Company – F180

NRC Form 366A (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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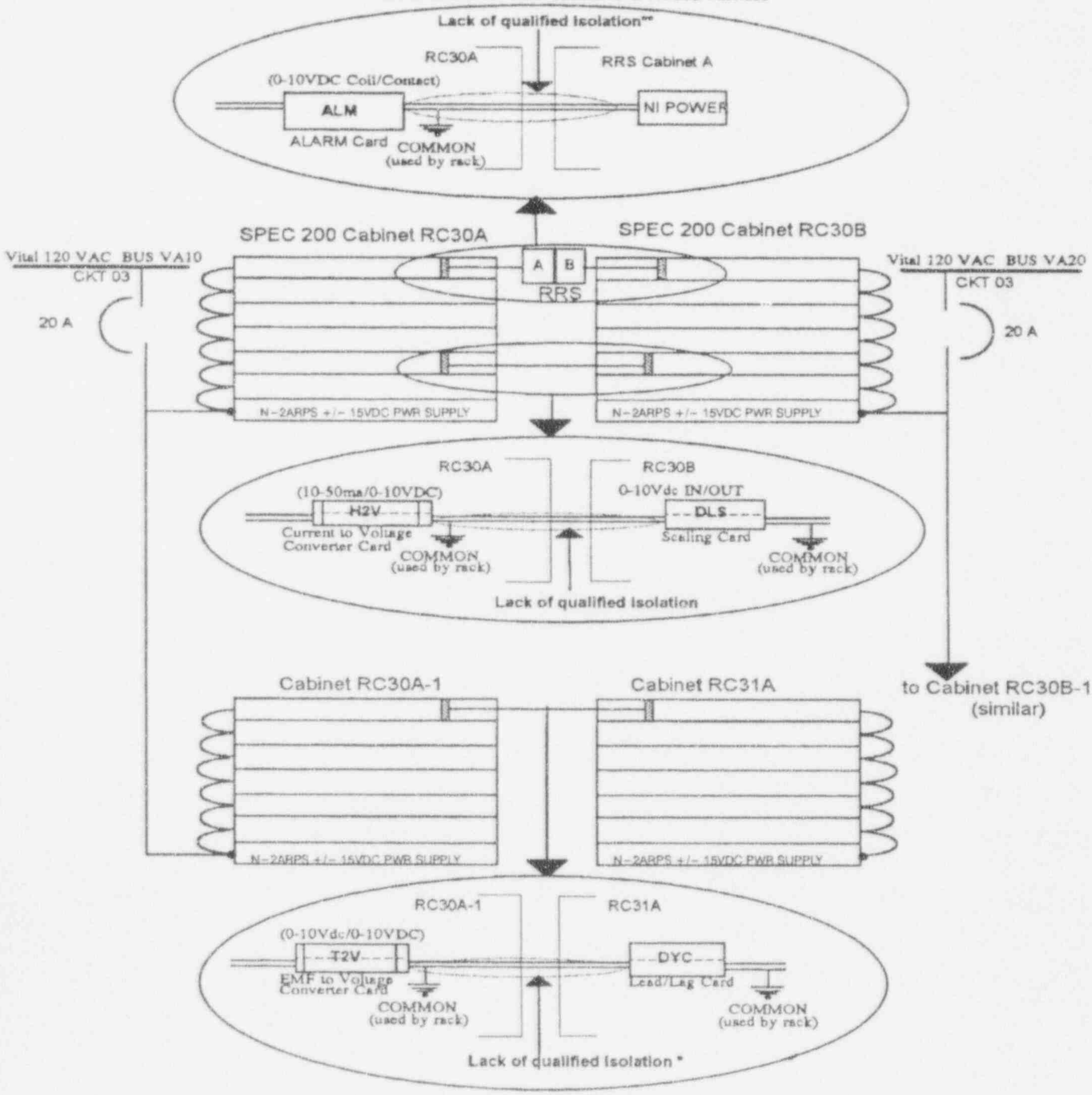

*NOTE: Similar for RC30B-1 to RC31B interface

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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Attachment 1

List of Instruments Associated with RC30A & RC30B	
Instrument Channel	**Comments**
Pressurizer Pressure	(P)–Reactor Rx trip on High Pressurizer Pressure
Auxiliary Feedwater flow (AFW) to #1 & #2 Steam Generators/AFW flow control valve control.	(E)–AFW Valves fail full open allowing full flow to S/Gs and AFW pumps start.
#1 and #2 Steam Generator level – Channels A & B	(P)(E)–Provides Lo Steam Generator levels (Rx trip) and Lo–Lo S/G Level AFW actuation.
#1 and #2 Steam Generator pressure – Channels A & B	(P)/(E)–Provides Main Steam Line Isolation Signal and Rx trip.
Containment pressure – Channels A & B	(P)/(E)– Provides high Containment pressure ESF function: gives Containment Isolation Actuation Signal (CIAS) or Containment Spray Actuation Signal (CSAS) and Rx trip.
Containment pressure – wide range – Channels A & B	Provides no protection function
#1, #2, #3 & #4 Safety Injection Tank Level	Provides no protection function
#1, #2, #3 & #4 Safety Injection Tank Pressure	Provides no protection function
Unit No. 2 Stack Air Flow Indication and Control	Provides no protection function
Reactor Regulating System "Power Summer"	Blocks Hi pressure, at power, ATWS Mitigation Actuation Signals
List of Instruments Associated with RC30A–1 & RC30B–1	
Instrument Channel	**Comments**
CIAS to Hydrogen (H2) Purge Isolation Valves	(E) – Spurious signal to close valves
Containment High Range Radiation Monitor to H2 Purge Isolation Valves	(E) – Spurious signal to close valves
Reactor Coolant Loop 1 & 2 flow	(P) – Rx trip on Low Flow Rate
Loop 1 and 2 cold leg temperature (thermal margin/low pressure setpoint and delta temperature power reference)	(P) – Thermal/Margin low pressure Rx trip input
Loop 1 and 2 hot leg temperature (thermal margin/low pressure setpoint and delta temperature power reference)	(P) – Thermal/Margin low pressure Rx trip input
Loop 1 cold leg average temperature calculator and feedwater regulating system (single element control).	No protection function
Loop 1 and 2 cold leg average temperature calculator and Inadequate Core Cooling Indication.	No protection function
Pressurizer pressure input to Low Temperature Over Pressure Protection (LTOP).	Auto LTOP input to PORVs blocked

NOTES:

A "(P)" designates a Reactor Protection Circuit

An "(E)" designates an Engineered Safety Features (ESF) Actuation Circuit

There are four channels of protection systems cabinets RC30A–D, which provide signals to the Reactor Protection and Engineered Safety Features Actuation Systems.