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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September 19, 1985

Docket No. 50-423
B11731

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Technical Specifications - Proof and Review

Representatives from Northeast Nuclear Energy Company (NNECO) met with the Staff on September 6, 1985 to discuss the Technical Specifications for Millstone Unit No. 3. NNECO was requested to submit additional information concerning certain draft technical specifications by September 20, 1985. Enclosed please find NNECO's response to the questions raised.

We trust the attached will resolve the Staff's concerns. If there are additional questions, please contact our licensing representative directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

J. F. Opeka

J. F. Opeka
Senior Vice President

E. J. Mroczka

By: E. J. Mroczka
Vice President

B001
3/40

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me E. J. Mroczka, who being duly sworn, did state that he is Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Sheila M. Oates
Notary Public
My Commission Expires March 31, 1986

TECHNICAL SPECIFICATIONS ADDITIONAL REVIEW AND INFORMATION ITEMS

Item No. 8:

Limiting Safety System Settings, Bases pages B2-7 and B2-8, Reactor Trip System Interlocks, P-6 and P-10.

NNECO's Response:

For P-6 Remove the statement "...provides a backup block for Source Range Neutron Flux doubling,...". There is no such feature in the Source Range Nuclear Instrumentation System.

P-7 - Remove the word "turbine trip". This permissive is addressed by P-9 and is not incorporated into P-7 in the Millstone 3 design.

The P-10 statement should stand as it is in the Proof and Review copy.

Item - 8

Tech Spec Additional Review/Information Items.

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Low Shaft Speed - Reactor Coolant Pumps (Continued)

pump speed (with resulting decrease in flow) on two reactor coolant pumps in any two operating reactor coolant loops. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above 1.30 following a 4-pump loss of flow event.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), ~~provides a backup block for Source Range Neutron Flux doubling~~, and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump low shaft speed, ~~[illegible]~~ pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant ...

TECHNICAL SPECIFICATIONS ADDITIONAL REVIEW AND INFORMATION ITEMS

Item 20: Table 3.3-6, Radiation Monitoring Instrumentation for Plant Operations.

Resubmit correct design information.

NNECO's Proposed Table 3.3-6

1) MP3 does not purge during operation, hence containment air monitors only safety function during modes 1-4 is for leakage detection - Item 1b. Other safety function for Containment Rad Monitors is for automatic purge isolation during Mode 6 Fuel Handling Accidents. The proper name, mode and alarm for this monitor has been indicated as Item 1a.

2) Item 1a provides purge isolation - This is an area monitor, not a particulate/gas - hence Item 2) is deleted.

3) Fuel Storage Pool Monitors do not provide automatic purge isolation -hence 3a is deleted.

4) The only control room monitor with a safety function is the Air Intake Rad Monitor which provides auto isolation. 1 SWCH Monitor is sufficient as there are also 1 air borne and 1 area monitor in the control room to indicate need for manual action.

   Item 4b deleted. Serves no safety function.

5) Action 28 should be replaced with our Action 25. These monitors must function at all times. Not just receiving. No need to suspend refueling operations as long as portable monitors are used.

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TABLE 3.3-6 Tech. Spec. Item 20

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment AREA ~~Atmosphere-Radioactivity High~~ PURGE AND EXHAUST ISOLATION	1	~~2~~ 2	All	≤ 1R/h ~~[2] mR/h~~	26
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2) Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
~~2. Purge and Exhaust Ventilation~~					
~~a. Particulate Radioactivity~~	~~1~~	~~2~~	~~All~~	~~*~~	~~26~~
~~b. Gaseous Radioactivity~~	~~1~~	~~2~~	~~All~~	~~*~~	~~26~~
2. ~~3.~~ Fuel Storage Pool Areas					
~~a. Radioactivity High~~					
~~Gaseous Radioactivity~~	~~1~~	~~2~~	~~**~~	~~≤ [2] mR/h~~	~~27~~
2 ~~b.~~ Criticality-Radiation Level	1	2	***	≤ 15 mR/h	28
3 ~~4.~~ Control Room					
a. Air Intake-Radiation Level	1/intake	2/intake	All	≤ 1.5×10^{-5} μc/cc ~~[2] mR/h~~	27
~~b. Control Room Atmosphere Radiation-High~~	~~1~~	~~2~~	~~All~~	~~≤ [2] mR/h~~	~~27~~

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TABLE 3.3-6 (Continued)

TABLE NOTATIONS

~~* Must satisfy Specification 3.11.2.1 requirements.~~

~~** With irradiated fuel in the fuel storage pool areas.~~

*** With fuel in the fuel storage pool areas.

ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.

ACTION 29 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours

Tech. Spec. Item 20

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment AREA ~~Atmosphere Radioactivity-High~~	S	R	M	All
b. RCS Leakage Detection PURGE AND EXHAUST				
1) Particulate Radioactivity	S	R	M	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	M	1, 2, 3, 4
~~2. Purge and Exhaust Ventilation~~				
~~a. Particulate Radioactivity~~	~~S~~	~~R~~	~~M~~	~~All~~
~~b. Gaseous Radioactivity~~	~~S~~	~~R~~	~~M~~	~~All~~
2 ~~3~~. Fuel Storage Pool Areas				
~~a. Radioactivity High Gaseous Radioactivity~~	~~S~~	~~R~~	~~M~~	~~**~~
2 ~~b~~. Criticality-Radiation Level	S	R	M	*
3 ~~4~~. Control Room				
a. Air Intake Radiation Level	S	R	M	All
~~b. Control Room Atmosphere Radiation-High~~	~~S~~	~~R~~	~~M~~	~~All~~

TABLE NOTATIONS

* With fuel in the fuel storage pool area.
** With irradiated fuel in the fuel storage pool areas.

Tech Spec Item 20

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TECHNICAL SPECIFICATION ADDITIONAL REVIEW AND INFORMATION ITEMS

Item 22: Table 4.3-4, Seismic Monitoring Instrumentation Surveillance Requirements.

Provide justification for the deletion of the monthly "CHANNEL CHECK" of the seismic triggers.

NNECO's Response:

NNECo is committed to the surveillance requirements of ANSI/ANS 2.2, Earthquake Instrumentation Criteria for Nuclear Power Plants. Even though the monthly surveillance requirements do not conform to the Technical Specification definition of CHANNEL CHECK, NNECo will accept Table 4.3-4, as written for the seismic triggers.

TECHNICAL SPECIFICATION ADDITIONAL REVIEW AND INFORMATION ITEMS

Item 28: 3/4.3.4, Turbine Overspeed Prot ction

Justify deleting turbine overspeed protection from Millstone Unit 3 technical specifications.

NNECO's Response:

Section 3/4.3.4 of Standard Technical Specifications incorporates arbitrary Limiting Conditions of Operation and Surveillance Requirements having no firm bases in maintaining a low probability of a turbine overspeed event. Several inservice tests and inspections which are essential in assuring a low probability of turbine missile generation not only as a consequence of excessive turbine overspeed but also as a result of a low pressure turbine rotor failure due to flaws that can initiate and propagate in service have not been addressed.

To assure a low turbine missile generation probability and to meet NNECO's more inclusive requirement of high overall turbine generator reliability, NNECO will implement a comprehensive turbine maintenance program based on the turbine manufacturer's (General Electric Co.) recommendations and NNECO's extensive operating experience with similar General Electric turbine generators on Millstone Units One and Two. Under the turbine maintenance program, inservice testing, periodic inspections, and other essential maintenance are performed in an integrated manner to assure that the high reliability inherent to the Millstone Unit Three turbine design will be maintained throughout plant life. Consistent with NNECO's past practice, the maintenance program will include extensive surveillance procedures (non-technical specification) for inservice testing.

The very favorable results from inservice testing at Millstone Units One and Two demonstrate the high reliability of General Electric turbine steam valves and overspeed protection systems and provides the justification for continuation of NNECO's past practices on Millstone Unit Three. For example, in over 24 unit years of periodic testing of main and intermediate steam valves and overspeed trip systems at MP-1 and MP-2, no failures have occurred of a valve to close or of an overspeed trip system to function properly. Also, all main and intermediate steam valves have been disassembled and inspected numerous times without a single problem detected that could have caused a valve to fail to close.

The turbine maintenance program will include but not be limited to:

a. Periodic stroke testing of the main stop and control valves and intermediate valves.

b. Periodic testing of extraction non-return valves.

c. Periodic performance of Mechanical Overspeed Trip Test, Mechanical Trip Piston Test, and Back-up Overspeed Logic Test.

d. Periodic demonstration of the complete trip system by an intentional overspeeding of the turbine up to the trip speed.

Item 28 continued

e. Periodic valve tightness tests.

f. Periodic turbine inspections including volumetric examinations of the LP rotor on a staggered interval.

g. Periodic steam admission valve inspections.

As previously agreed upon and docketed in the Millstone Unit 3 FSAR, NNECO will submit for NRC approval, within three years of obtaining an operating license, a turbine system maintenance program based on the manufacturer's (General Electric Co.) calculations of missile generation probabilities. The NRC has officially accepted the G. E. probabilistic methodology which now provides the basis for nuclear low pressure rotor inspection recommendation.

Based on the above discussion and the FSAR docketed analysis showing acceptable results for postulated turbine missile damage, it is clear that the overall turbine test and maintenance program does not fall under Technical Specification as required by 10CFR50.36.

TECHNICAL SPECIFICATION ADDITIONAL REVIEW AND INFORMATION ITEMS

Item 41: 4.6.1.7.2, Containment Ventilation System

Justify surveillance to demonstrate operability of containment purge supply and exhaust isolation valves every refueling instead of every six months.

Containment Ventilation System (vs. tech. spec. 4.6.1.7.2)

Section III.D.3 of 10CFR50 Appendix J only requires that containment isolation valves be tested during refueling outages (but in no case at intervals greater than 2 years). Valves with resilient seals have been found capable of maintaining their leak tightness in this time interval at MP-1. For example, MP-1 atmospheric control butterfly valves with resilient seats have passed 3 App. J leakage tests without replacement. Recently, NNECO has been considering replacement of purge valve seats once every two core operating cycles. This maintenance program and test experience are considered to be justification for exemption of these valves from the 6 month surveillance requirements of 4.6.1.7.2 at MP-1. This approach should be an acceptable alternative for Millstone 3 as well because Millstone 3 purge isolation valves have resilient seal seats.

In addition, the difficulty in performing the 6 month surveillance during plant operations must be considered. For example, the containment purge inner isolation valves are exposed to subatmospheric pressure during operations. The outer valves are exposed to atmospheric pressure at this same time. A test connection between the valves could be used to pressurize them during operations, however, the correct differential pressure could not be achieved simultaneously across both containment valves. Successive pressurizations, to accomplish correct ΔP across each valves, would present difficulties in interpretations of test results. Testing of the purge valves can be achieved effectively during shutdowns. However, based on the information provided above, 6 month shutdowns instituted solely for the purpose of testing valves with resilient seal seats is unnecessary.

TECHNICAL SPECIFICATIONS ADDITIONAL REVIEW AND INFORMATION ITEMS

Item 55 and 59: 3.8.1.1.b(5) and 3.8.1.2.b(5), Capability to transfer lubricating oil from storage to the diesel generator unit.

Justification to delete the above.

NNECO's Response:

This item concerns a Technical Specification requirement dealing with the verification of the capability to transfer lubricating oil from storage to the diesel unit. This transfer and subsequent fill are manually performed. In our April 6, 1985 response (Docket No. 50-423), it was stated that a written station procedure for the addition of lube oil to an operating diesel generator will be developed. This procedure will address how and where to add lube oil and the type of lube oil to be used. Specification procedural steps will be posted locally in the diesel generator rooms. The use of this procedure will be demonstrated during preoperational testing which affords training to personnel in its use. Training will be conducted onsite for responsible maintenance personnel. Since this is a manual transfer and is adequately covered in these station procedures, the verification appearing in the TS should be deleted.

TECHNICAL SPECIFICATION REVIEW AND INFORMATION ITEMS

Item 66: Table 3.8-2a1 Motor-Operated Valves Thermal Overload Protection With Bypass Device Bypassed Only During Accident Conditions.

Table 3.8-2b, Motor-Operated Valves Thermal Overload Protection Not Bypassed.

NNECO's Response:

Tables attached.

TABLE 3.8-2a

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3CCP*MOV45A	Containment Isolation Valve
3CCP*MOV45B	Containment Isolation Valve
3CCP*MOV48A	Containment Isolation Valve
3CCP*MOV48B	Containment Isolation Valve
3CCP*MOV49A	Containment Isolation Valve
3CCP*MOV49B	Containment Isolation Valve
3CCP*MOV222	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV223	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV224	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV225	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV226	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV227	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV228	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV229	Containment Air Recirculation Cooling Coil Supply
3CHS*LCV112B	Volume Control Tank Outlet Isolation
3CHS*LCV112C	Volume Control Tank Outlet Isolation
3CHS*LCV112D	Volume Control Tank Outlet Isolation
3CHS*LCV112E	Volume Control Tank Outlet Isolation
3CHS*MV8100	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8104	Boric Acid Flter to Charging Pump Isolation
3CHS*MV8105	Charing Pump to Reactor Coolant Isolation
3CHS*MV8106	Charging Pump to Reactor Coolant Isolation

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3CHS*MV8109A	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8109B	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8109C	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8109D	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8110	Charging Pump Miniflow Isolation
3CHS*MV8111A	Charging Pump Miniflow Isolation
3CHS*MV8111B	Charging Pump Isolation
3CHS*MV8111C	Charging Pump Isolation
3CHS*MV8112	Reactor Coolant Pump Seal Water Isolation
3CHS*MVP116	Charging Header Isolation
3CHS*MV8438A	Charging Header Isolation
3CHS*MV8438B	Charging Header Isolation
3CHS*MV8438C	Charging Header Isolation
3CHS*MV8468A	Charging Pump Suction Isolation
3CHS*MV8468B	Charging Pump Suction Isolation
3CHS*MV8507A	Boric Acid Gravity Feed
3CHS*MV8507B	Boric Acid Bravity Feed
3CHS*MV8511A	Charging Pump Miniflow Control
3CHS*MV8511B	Charging Pump Miniflow Control
3CHS*MV8512A	Charging Pump Miniflow Control
3CHS*MV8512B	Charging Pump Miniflow Control
3CMS*MOV24	Containment Atmosphere Monitoring Inside Containment Isolation

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3FWA*MOV35A	Steam Generator Auxiliary Feedwater Pump Isolation
3FWA*MOV35B	Steam Generator Auxiliary Feedwater Pump. Isolation
3FWA*MOV35C	Steam Generator Auxiliary Feedwater Pump. Isolation
3FWA*MOV35D	Steam Generator Auxiliary Feedwater Pump Isolation
3RCS*MV8000A	Pressurizer Relief Isolation
3RCS*MV8000B	
3RCS*MV8098	Reactor to Excess Letdown Isolation
3RHS*FCV610	Residual Heat Removal Pump Miniflow
3RHS*FCV611	Residual Heat Removal Pump Miniflow
3RHS*MV8701A	Residual Heat System Inlet Isolation
3RHS*MV8701B	Residual Heat System Inlet Isolation
3RHS*MV8701C	Residual Heat System Inlet Isolation
3RHS*MV8702A	Residual Heat System Inlet Isolation
3RHS*MV8702B	Residual Heat System Inlet Isolation
3RHS*MV8702C	Residual Heat System Inlet Isolation
3RHS*MV8716A	Residual Heat System Conn Isolation
3RHS*MV8716B	Residual Heat System Conn Isolation
3RSS*MOV20A	Containment Recirculation Spray Header Isolation
3RSS*MOV20B	Containment Recirculation Spray Header Isolation
3RSS*MOV20C	Containment Recirculation Spray Header Isolation
3RSS*MOV20D	Containment Recirculation Spray Header Isolation

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3RSS*MOV23A	Containment Recirculation Pump Suction
3RSS*MOV23B	Containment Recirculation Pump Suction
3RSS*MOV23C	Containment Recirculation Pump Suction
3RSS*MOV23D	Containment Recirculation Pump Suction
3RSS*MOV38A	Containment Recirculation Pump Miniflow
3RSS*MOV38B	Containment Recirculation Pump Miniflow
3RSS*MV8837A	Recirculation Spray System to Residual Heat Removal System Cross Connection
3RSS*MV8837B	Recirculation Spray System to Residual Heat Removal System Cross Connection
3RSS*MV8838A	Recirculation Spray System to Residual Heat Removal System Cross Connection
3RSS*MV8338B	Recirculation Spray System to Residual Heat Removal System Cross Connection
3SIH*MV8801A	Charging Pump to Cold Leg Isolation
3SIH*MV8801B	Charging Pump to Cold Leg Isolation
3SIH*MV8802A	Safety Injection Pump to Hot Leg Isolation
3SIH*MV8806	Refueling Water Stoarge Tank to Safety Injection Isolation
3SIH*MV8807A	Safety Injection High & Connection to Low Pressure Safety Injection.
3SIH*MV8813	High Pressure Safety Injection Pump Miniflow
3SIH*MV8814	High Pressure Safety Injection Pump Miniflow
3SIH*MV8821A	Safety Injection Pump Discharge Isolation
3SIH&MV8821B	Safety Injection Pump Discharge Isolation

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3SIH*MV8835	SIH Supply to LPSI
3SIH*MV8920	High Pressure Safety Injection Pump Miniflow
3SIH*MV8923A	Safety Injection Pump Suction
3SIH*MV8923B	Safety Injection Pump Suction
3SIH*MV8924	Safety Injection/Charging Suction Cross Connection
3SIL*MV8804A	Residual Heat Removal Pump to Charging Pump
3SIL*MV8804B	Residual Heat Removal Pump to Charging Pump
3SIL*MV8808A	Accumulation Isolation
3SIL*MV8808B	Accumulation Isolation
3SIL*MV8808C	Accumulation Isolation
3SIL*MV8808D	Accumulation Isolation
3SIL*MV8809A	Residual Heat Removal Cold Leg Isolation
3SIL*MV8809B	Residual Heat Removal Cold Leg Isolation
3SIL*MV8812A	Refueling Water Storage Tank to Residual Heat Removal
3SIL*MV8812B	Refueling Water Storage Tank to Residual Heat Pump
3SIL*MV8840	Residual Heat Removal Hot Leg Isolation
3SWP*MOV24A	Service Water Backwash
3SWP*MOV24B	Service Water Backwash
3SWP*MOV24C	Service Water Backwash
3SWP*MOV24D	Service Water Backwash
3SWP*MOV50A	Reactor Plant Component Cooling Water Heat Exchanger

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3SWP*MOV50B	Reactor Plant Component Cooling Water Heat Exchanger
3SWP*MOV54A	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV54B	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV54C	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV54D	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57A	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57B	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57C	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57D	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV71A	Turbine Plant Component Cooling Water Heat Exchanger
3SWP*MOV71B	Turbine Plant Component Cooling Water Heat Exchanger
3SWP*MOV102A	Service Water Pump Discharge
3SWP*MOV102B	Service Water Pump Discharge
3SWP*MOV102C	Service Water Pump Discharge
3SWP*MOV102D	Service Water Pump Discharge
3SWP*MOV115A	Circulating Water Pump Bearing Lube Watch Supply

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
WITH BYPASS DEVICES BYPASSED ONLY DURING ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3SWP*MOV115B	Circulation Water Pump Bearing Lube Water Supply
3SWP*MOV130A	Motor Control Center and Rod Control Service Water Isolation
3SWP*MOV130B	Motor Control Center and Rod Control Service Water Isolation

TABLE 3.8-2b

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED
3CVS*MOV25	Containment Atmosphere Monitoring Isolation
3FWS*MOV35A	Steam Generator Feedwater Pump Isolation
3FWS*MOV35B	Steam Generator Feedwater Pump Isolation
3FWS*MOV35C	Steam Generator Feedwater Pump Isolation
3FWS*MOV35D	Steam Generator Feedwater Pump Isolation
3IAS*MOV72	Containment Instrument Air Isolation
3LMS*MOV40A	Containment Open Pressure Tap Isolation
3LMS*MOV40B	Containment Open Pressure Tap Isolation
3LMS*MOV40C	Containment Open Pressure Tap Isolation
3LMS*MOV40D	Containment Open Pressure Tap Isolation
3MSS*MOV17A	Steam Generator Auxiliary Feedwater Pump Steam Supply Non-Return
3MSS*MOV17B	Steam Generator Auxiliary Feedwater Pump Steam Supply Non-Return
3MSS*MOV17D	Steam Generator Auxiliary Feedwater Pump Steam Supply Non-Return
3MSS*MOV18A	Main Steam Pressure Relief Isolation
3MSS*MOV18B	Main Steam Pressure Relief Isolation
3MSS*MOV18C	Main Steam Pressure Relief Isolation
3MSS*MOV18D	Main Steam Pressure Relief Isolation
3MSS*MOV74A	Main Steam Pressure Relief Bypass
3MSS*MOV74B	Main Steam Pressure Relief Bypass
3MSS*MOV74C	Main Steam Pressure Relief Bypass
3MSS*MOV74D	Main Steam Pressure Relief Bypass

TABLE 3.8-2b (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED
3QSS*MOV29A	Refueling Water Chemical Addition Tank Discharge
3QSS*MOV29B	Refueling Water Chemical Addition Tank Discharge
3QSS*MOV34A	Quench Spray Isolation
3QSS*MOV34B	Quent Spray Isolation
3QSS*MOV44A	Quench Spray Pump Discharge
3QSS*MOV44B	Quench Spray Pump Discharge
3RCS*MV8001A	Reactor Coolant Loop Stop (Hot)
3RCS*MV8001B	Reactor Coolant Loop Stop (Hot)
3RCS*MV8001C	Reactor Coolant Loop Stop (Hot)
3RCS*MV8001D	Reactor Coolant Loop Stop (Hot)
3RCS*MV8003A	Reactor Coolant Bypass Leg Stop
3RCS*MV8003B	Reactor Coolant Bypass Leg Stop
3RCS*MV8003C	Reactor Coolant Bypass Leg Stop
3RCS*MV8003D	Reactor Coolant Bypass Leg Stop
3RCS*MV8002A	Reactor Coolant Loop Stop (Cold)
3RCS*MV8002B	Reactor Coolant Loop Stop (Cold)
3RCS*MV8002C	Reactor Coolant Loop Stop (Cold)
3RCS*MV8002D	Reactor Coolant Loop Stop (Cold)

TECHNICAL SPECIFICATION ADDITIONAL REVIEW AND
INFORMATION ITEMS

<u>Item 67</u>: 4.8.4.2.1.a, Electrical Equipment Protection Devices Motor-Operated Valves Thermal Overload Protection.

Justify verifying the thermal overload protection for the required valves are bypassed by an operable integral bypass device by performance of a Trip Actuation Device Operational Test of the bypassed circuitry: every refueling instead quarterly.

<u>NNECO's Response</u>:

This surveillance requirement to test the Thermal Overload (TOL) bypass device every 92 days and following maintenance of the MOV starter is based on the device being integral to the Motor Operated Valve (MOV) starter. The Millstone 3 bypass design utilizes an ESF slave relay contact to operate the valve for accident conditions, as well as bypass the TOL. This relay is in the control room, not integral to the MOV starter. The ESF slave relays are tested every quarter by energization and verification of operability, including devices. Every eighteen months during refueling or cold shutdown, the MOV is demonstrated operable by verifying a test signal will actuate the valve to its accident position. These surveillance requirements verify the operability of the valve actuating circuitry, and since the same circuitry is used to bypass the TOL, it also verifies the operability of the TOL "bypass device". Therefore, surveillance 4.8.4.2.1 is not applicable to MP3 and NNECO proposes it to be deleted or reworded to delete reference to the motor starter and reworded as follows:

> 4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed by the appropriate accident signal(s) by performance of a Trip Actuation Device Operational Test of the bypass circuitry during the Cold Shutdown or Refueling Mode at least one per 18 months.

This change will bring the TOL bypass testing requirement frequency in line with that of the rest of the valve control circuitry. There is no justification for testing a highly reliable component of the circuitry. The above surveillance is already accomplished as part of the surveillance of the valves, therefore, it does not represent new or additional testing.

TECHNICAL SPECIFICATION REVIEW AND
INFORMATION ITEMS

Item 74: Figure 3.4-4a
Figure 3.4-4b

Figure B 3/4.2-1a
Figure B 3/4.2-1b

Figure B 3/4.4-1
Figure B 3/4.4-2

Figure 5.1-3

NNECO's Response:

The above Figures are attached.

Figure 3.4-4a
Nominal Maximum Allowable PORV
Setpoint For The Cold Overpressure System
(Four Loop Operation)

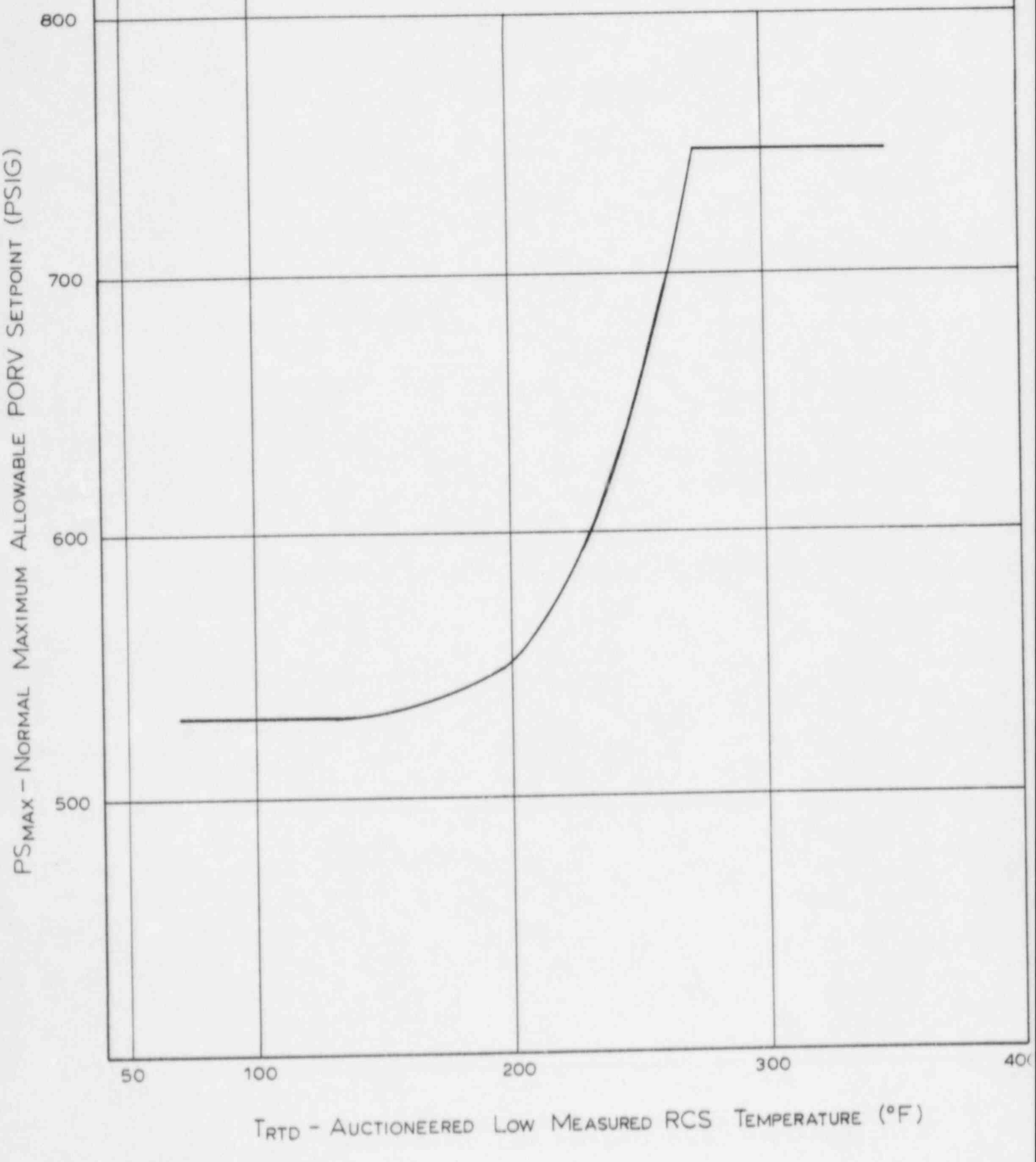
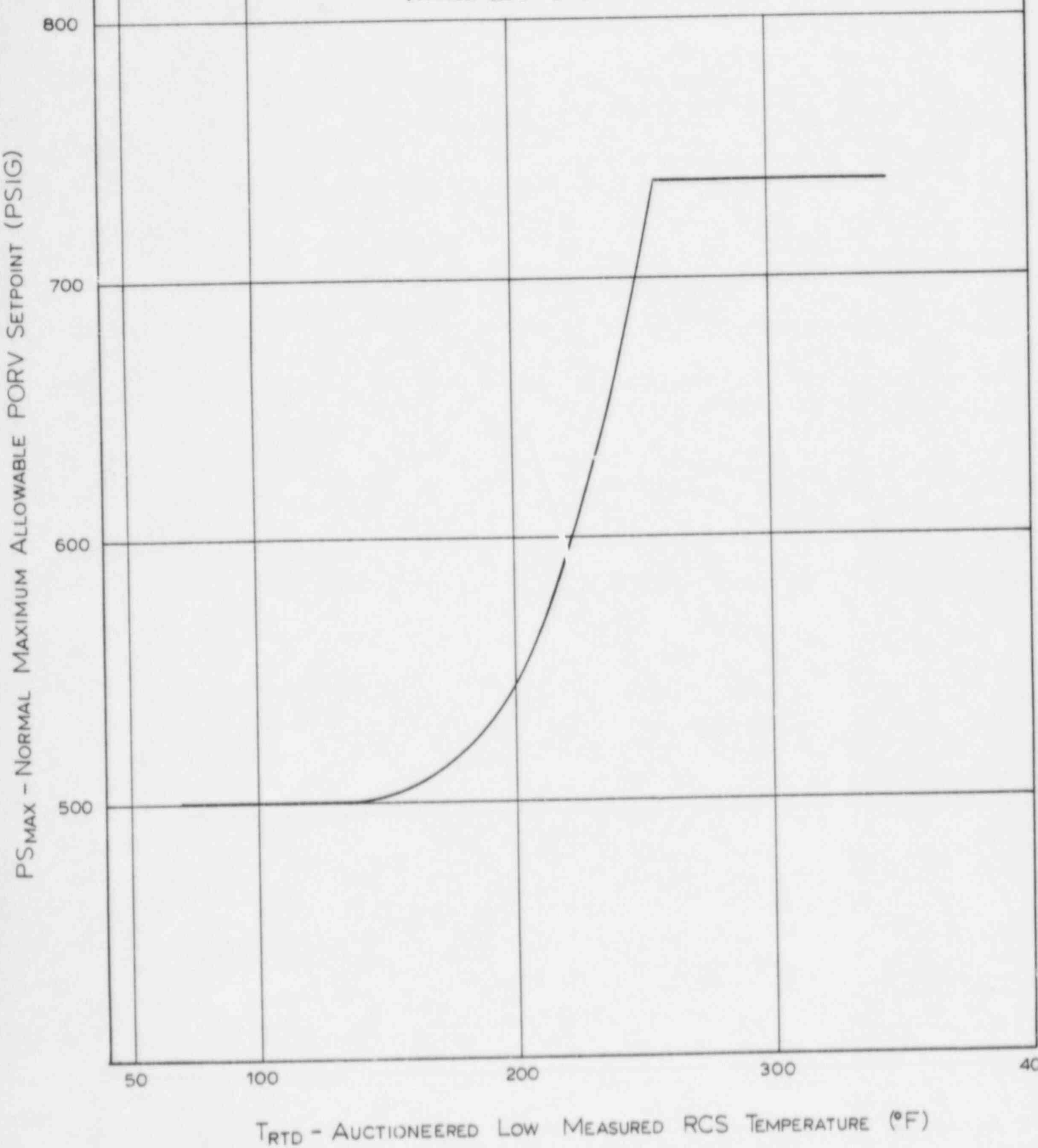


Figure 3.4-4b
Nominal Maximum Allowable PORV
Setpoint For The Cold Overpressure System
(Three Loop Operation)



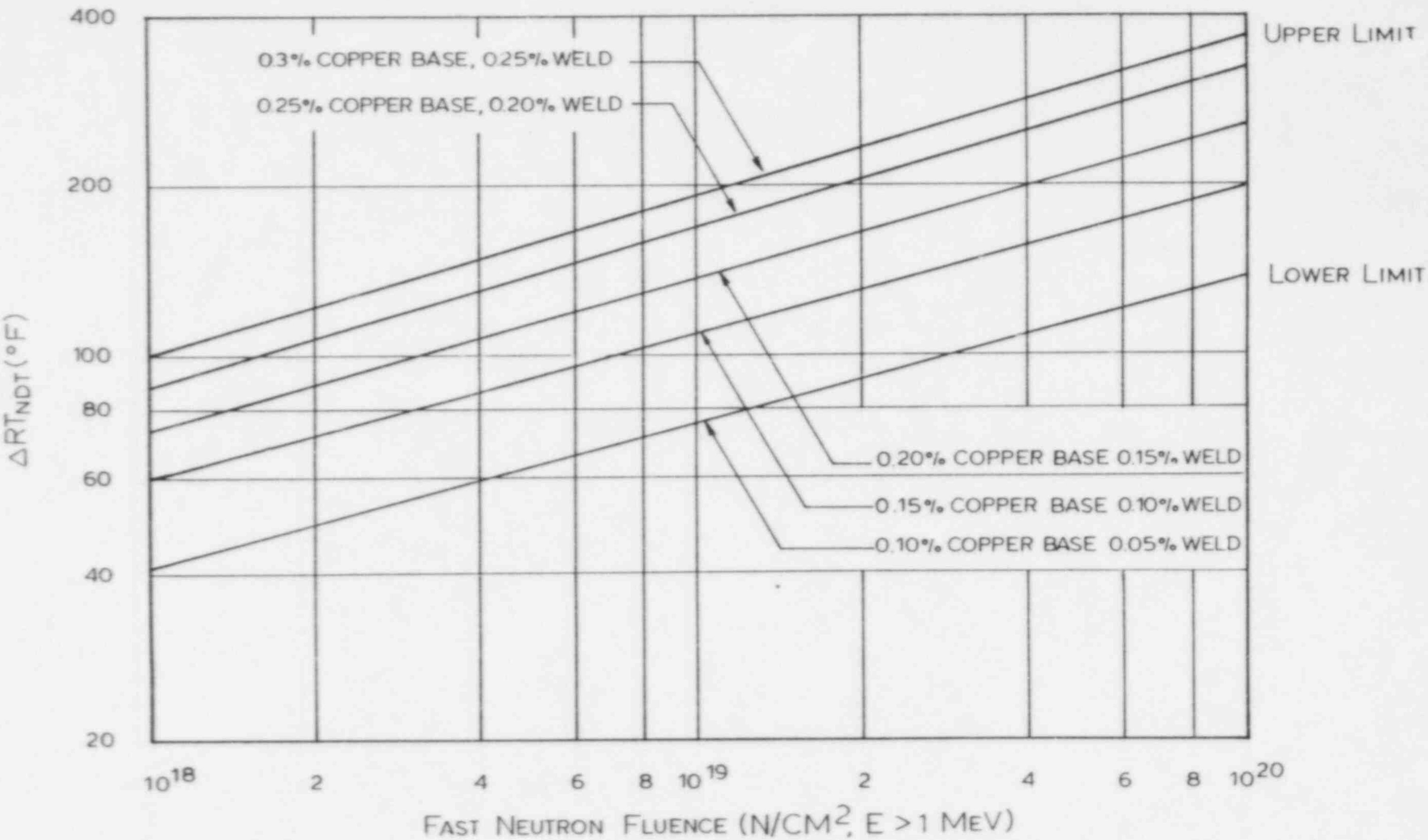


FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER ON SHIFT OF RT_{RDT} FOR REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F

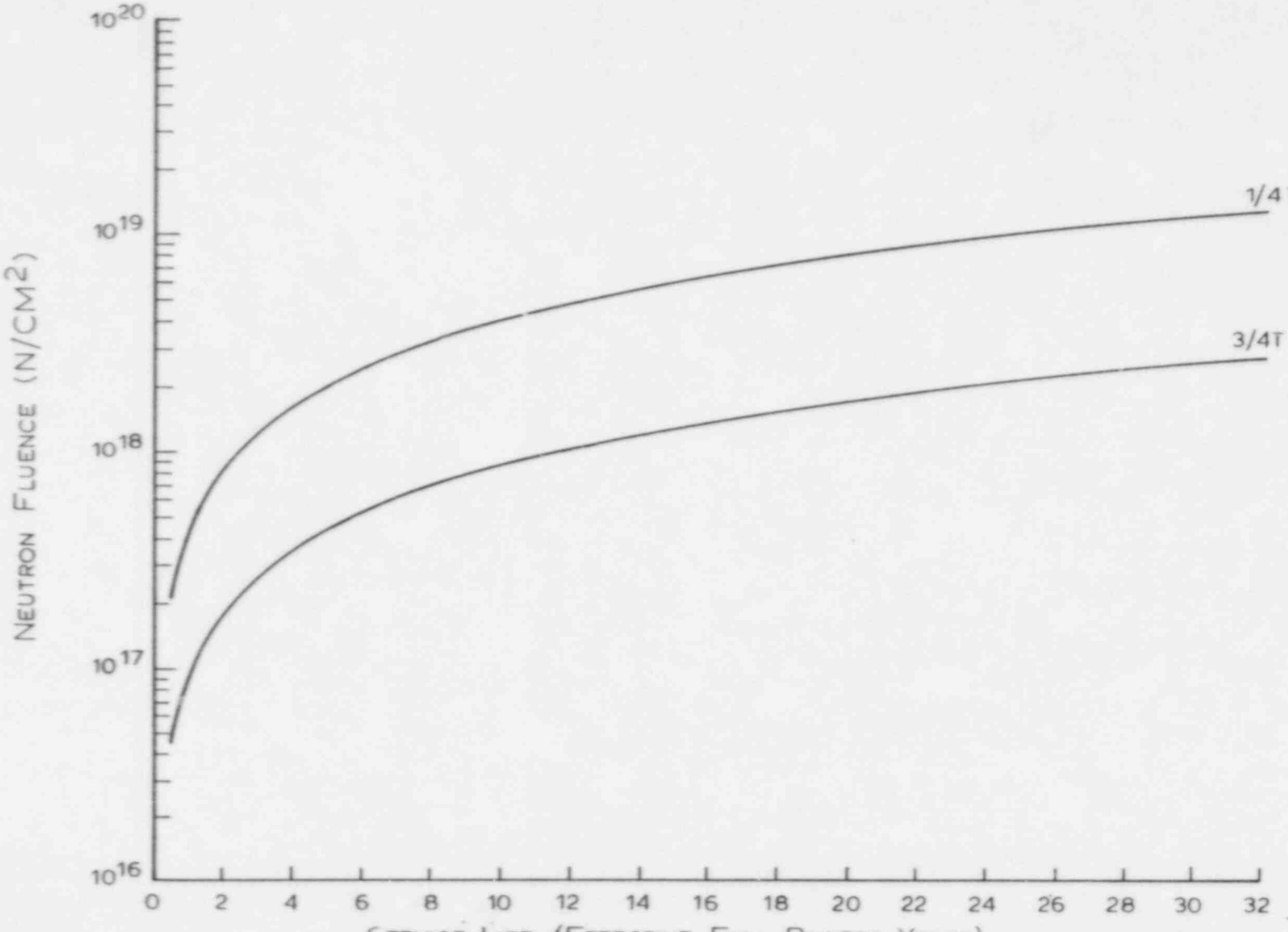


FIGURE B3/4.4-1

FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

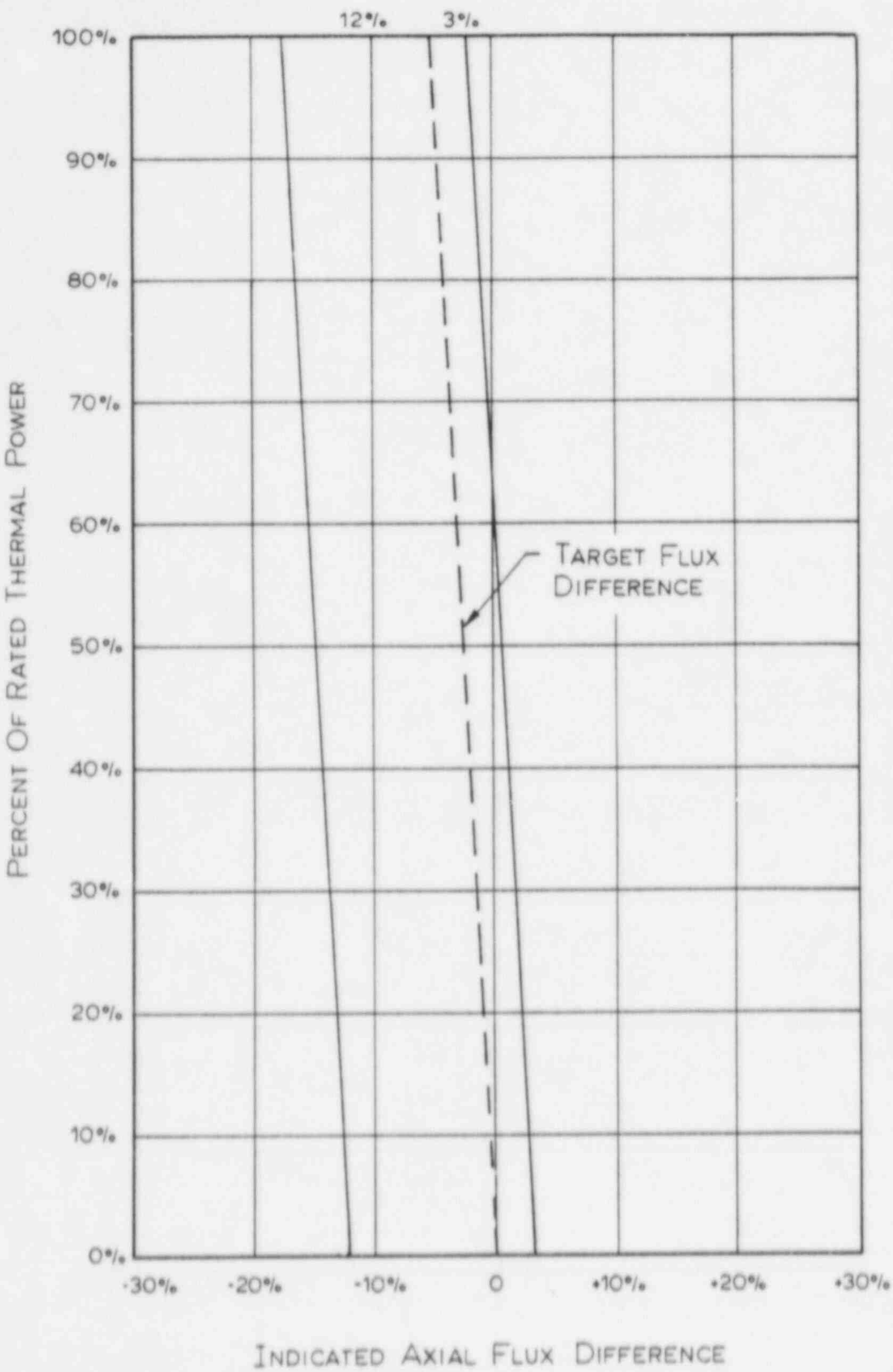


Figure B 3/4.2-1a

Typical Indicated Axial Flux Difference Versus Thermal Power

For Four Loop Operation

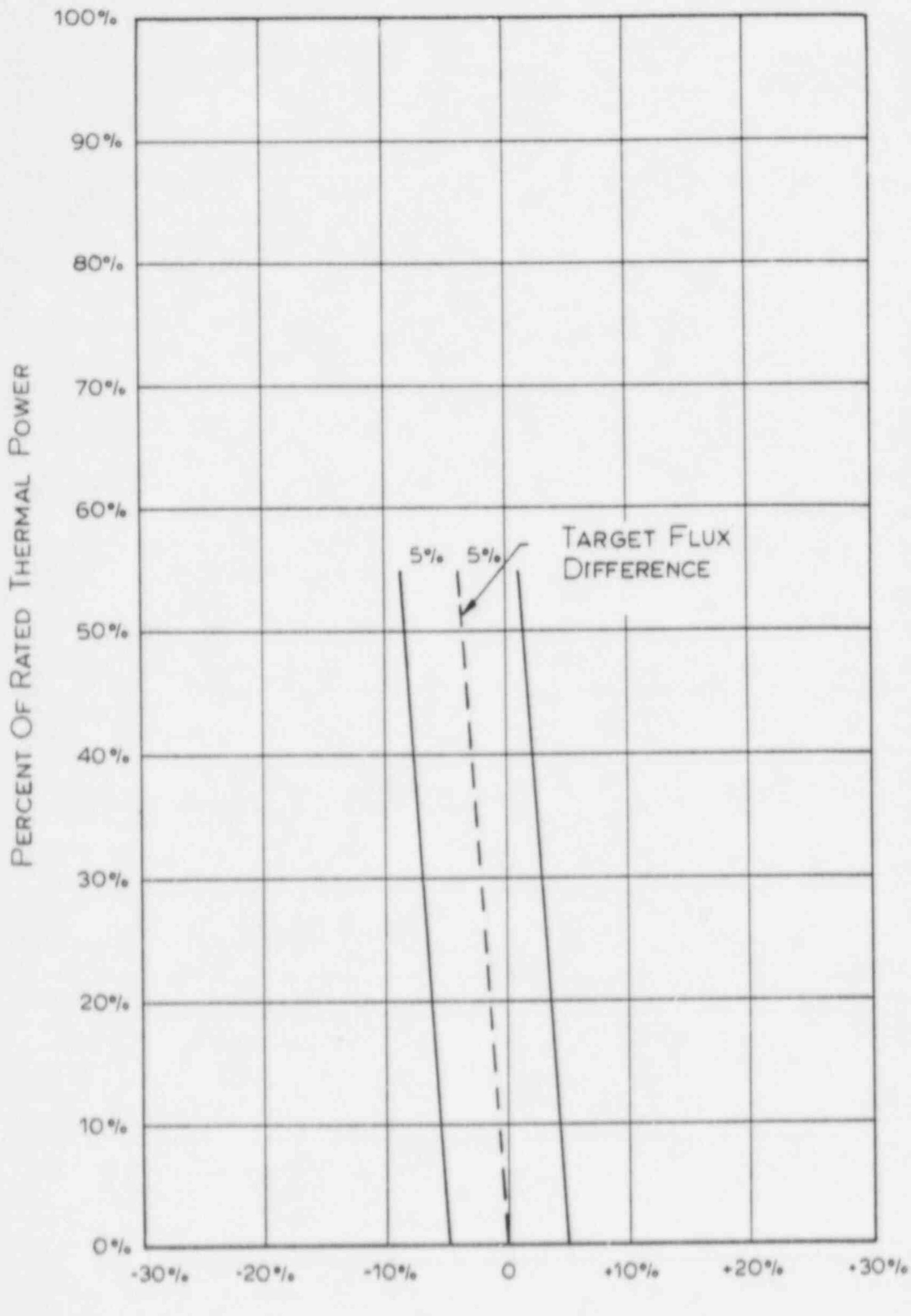


Figure B3/4.2-1b

Typical Indicated Axial Flux Difference Versus Thermal Power For Three Loop Operation

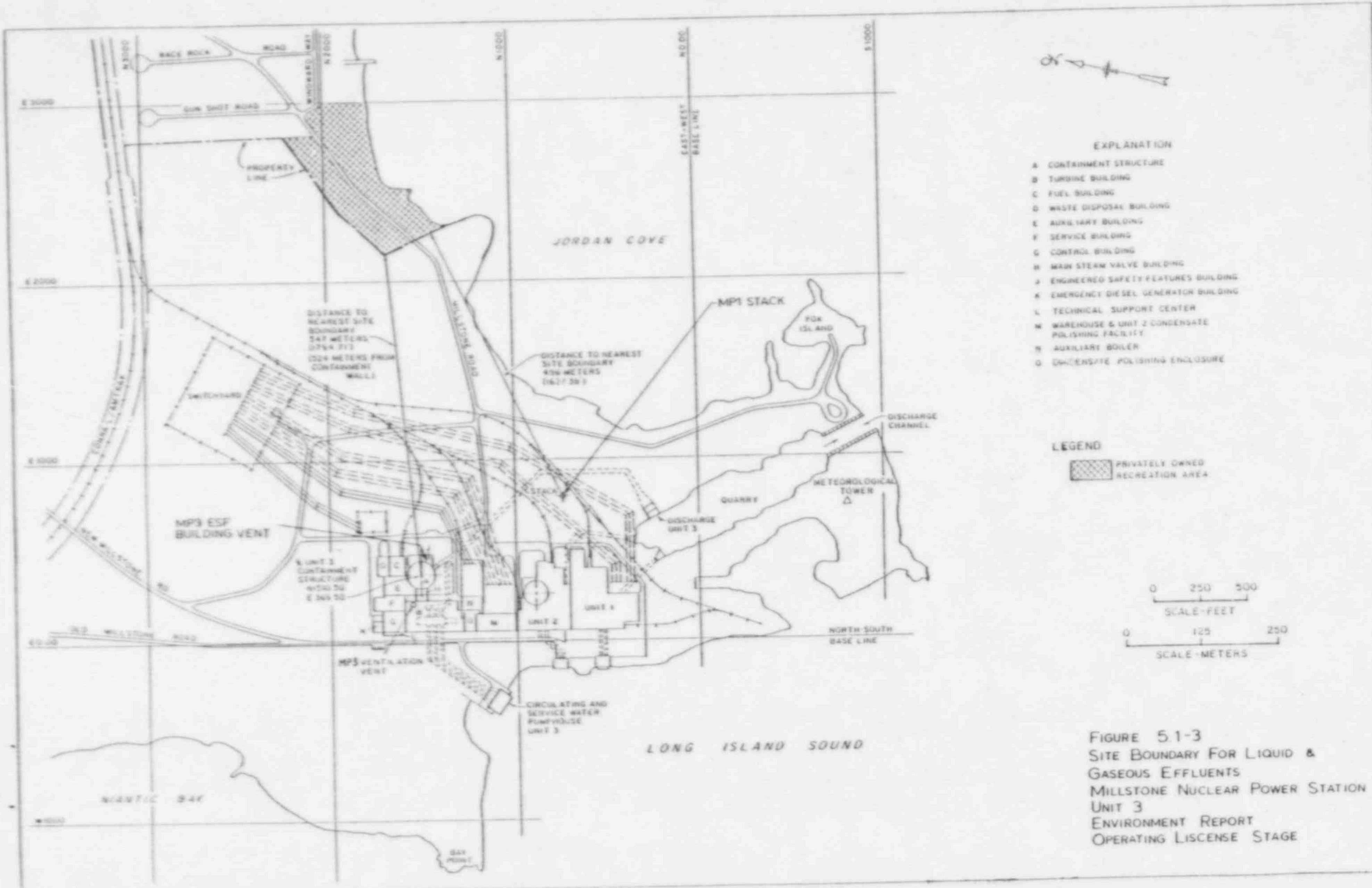


Figure 5.1-3
Site Boundary For Liquid & Gaseous Effluents
Millstone Nuclear Power Station
Unit 3
Environment Report
Operating Liscense Stage

TECHNICAL SPECIFICATION ADDITIONAL REVIEW AND INFORMATION ITEMS

Item 75: 3.1.1.2, Shutdown Margin

Provide the Shutdown Margin

NNECO's Response:

The correct value of SHUTDOWN MARGIN in Mode 5 should be 1.6% Δ k/k.

The BASES for Specification 3/4.1.1.2 should be modified to read as follows:

> "With Tavg less than 200°F, the reactivity transients resulting from a postulated steamline cooldown are minimal. A 1.6% Δ k/k value of SHUTDOWN MARGIN is required to provide protection against a boron dilution accident."

As a result of the change to Specification 3.1.1.2 the value of SHUTDOWN MARGIN in the following Technical Specifications should be changed from 2% Δ k/k to 1.6% Δ k/k.

3.1.2.2, 3.1.2.4, 3.1.2.6, 3.4.1.5, and 3.4.1.6.

SER Item

Item 2: 3/4.2.3, RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor

SER Section 4.4.4.2, page 4-31-RCS flow measurement needs to be in technical specifications.

NNECO's Response:

The revised response to FSAR Question 492.7 will provide the correct value for RCS Flow uncertainty.

SER Item

Item 6: 3/4.8.1 A. C. Sources

SER Section 8.2.2.5, page 8-4 would require the Severe Line Outage Detector (SLOD) Scheme surveillance and operability requirements for the protective scheme be in technical specifications.

NNECO's Response:

Section 8.1.3 of the Millstone Unit No. 3 (MP-3) FSAR describes a Severe Line Outage Detector (SLOD) Scheme. The SLOD scheme provides protection against potential system instability in the event of a loss of the two 345KV circuits on either of the two double circuit towers at a time when one of the remaining two circuits is out of service and Millstone site generation is heavy.

Question 430.7, part b, requesting additional details relative to SLOD resulted in NNECO subsequently providing this information. It was our understanding that the NRC reviewer, upon detailed review of that supplement, found the scheme to be acceptable.

The Millstone 3 Safety Evaluation Report (SER) was issued in July of 1984. Section 8.2.2.5 of the SER, entitled Generation Rejection Scheme documents your determination that the design meets GDC 17 and 18 criterion and is acceptable. Further stated however, is that "Surveillance and operability requirements for the protection scheme will be included in the Technical Specifications". NNECO strongly disagrees with this position, for the following reasons.

The transmission system (grid) within the State of Connecticut is operated by the Connecticut valley Electric Exchange (CONVEX). The Millstone switchyard is an important element of the Connecticut and New England grids, and is operated by CONVEX in close coordination with NNECO. Each of the Millstone generating units has two dedicated connections to the Millstone switchyard over which the respective unit has administrative control, and addressed by the unit Technical Specifications.

All the protective features contained within the switchyard (Millstone and all other switchyards) fall under the jurisdiction of the Northeast Utilities Transmission Organization with CONVEX serving as the operator. The SLOD scheme at Millstone is not Millstone Unit No. 3 specific, but the scheme required by the transmission system, wherein protection against the occurrence of system instability is deemed prudent. In this respect, it is only one of many system features which in total give rise to a highly reliable transmission network.

We have complete confidence in the ability of CONVEX to satisfactorily control the transmission system. They have demonstrated that ability as evidenced by the high reliability of the Connecticut Power System and by their sensitivity to the special needs of the nuclear units. CONVEX has in place operating procedures, and their equivalent of limiting conditions for operation (complete with compensatory interim measures), to provide for any transmission system off normal condition.

Item No. 6 continued

General Design Criterion (GDC) 17 states that "Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure...".

The statement "minimize to the extent practical" fully recognizes the non Class 1E and non-seismically qualified nature of the transmission system. The installation of the SLOD scheme is one element in the design of the transmission system to minimize the probability of loss of offsite power. The Technical Specifications the NRC desires to impose upon a small portion of that system (SLOD) are inconsistent with normal CONVEX controls. Although system reliability is largely affected by the quality and reliability of the hardware; it is also true that software elements are important to reliability. Such decisions as to how VARS will be dispatched (a continuously changing target/plan) and how much spinning reserve and of what type in what location, are all part of CONVEX's day to day activity.

Regarding SLOD specifically, the scheme was engineered for transmission system stability purposes, and it will be operated and maintained to support a high degree of reliability. No offsite Millstone switchyard transmission lines will be removed from service when all three Millstone Units are operating except under forced outage circumstances.

If, however unlikely, SLOD is not in service at that time, CONVEX will act to reduce Millstone site generation, within a matter of minutes. Similarly, if a line were forced out of service whenever SLOD is in service, (the expected case) CONVEX will act to optimize load flows and spinning reserve to prepare for the unlikely probability that SLOD could operate, and disconnect significant amount of nuclear generation.

We believe that this response appraises you of the complexities associated with controlling the transmission system, and clarifies the purpose of the SLOD scheme in relation to the transmission/generation system. Imposing plant specific technical specifications to address SLOD unavailability is inappropriate to effectively ensure the availability of offsite power.

SER Item

Item 7: 4.8.2.1, D. C. Sources

SER Section 8.3.2.2 would require battery float charge to be periodically monitored as a technical specification surveillance.

NNECO's Response:

There are no bases for requiring the battery float current to be periodically monitored, for the following reasons:

1) All conditions which would result in loss of battery float charging current are either automatically annunciated by the battery trouble and battery charger trouble alarms, or would be detected by routine surveillance of the battery pilot cells.

2) At MP-3, monitoring battery float charge requires a portable meter be connected to a test jack installed on the front of the main DC distribution panels. This is not a normal operator evolution. This capability was mentioned in the response to Q430.43 as being available should the need to measure float current arise. This need would be determine from the alarms and indications mentioned above. Otherwise, there is no reason for taking the measurement.

SER Items

Item 11: 3/4.8.1, Electrical Power Systems, A.C. Sources

Section 9.5.8, page 9-82 of the SER requests the access hatch of the diesel exhaust be included in Millstone Unit 3 technical specification.

NNECO's Response:

In a January 24, 1985 response to the staff (B11422), it was indicated that the measures taken to ensure access hatch operability would be addressed in the plant preventative maintenance procedures and thus do not need to appear in the Technical Specifications. This proposal was previously verbally accepted by the staff.

ADDITIONAL REVIEW REQUIRED

Item 5: 3/4.1.3.3, Position Indication System

Surveillance 4.1.3.3 does not specify an interval.

NNECO's Response:

Revised specification 3/4.1.3.3 is attached. The interval chosen is the same as that in Revision 4 of the Standard Technical Specifications, at least once per 18 months.

Items - Additional Review Required

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REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **.

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel, at least once per 12 months.

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

ADDITIONAL REVIEW REQUIRED

Item 8: Table 3.3-3, Engineered Safety Features Actuation System Instrumentation Trip Setpoints.

Request addition of "if required" to Action 19 statement.

NNECO's Response:

Action 19 statement should have the words "if required" added to the end of the paragraph so that the plant is placed in a mode for which the specification does not apply rather than COLD SHUTDOWN, should this not be an APPLICABLE MODE.

ADDITIONAL REVIEW REQUIRED

Item 10: Table 4.3-6, Remote Shutdown Monitoring Instrumentation Surveillance Requirements.

Discuss Instrument No. 15, Source Range Count Rate Channel Check.

NNECO's Response:

The Channel Check surveillance periodicity should be M*, where * will denote "NA with power above P-6". Add to the bottom of the page "*NA with power above P-6". The Source Range instrumentation are not energized when above the P-6 setpoint where the Intermediate Range Instrumentation are active.

ADDITIONAL REVIEW REQUIRED

Item 11: Reactor Coolant System, 3.4.1.4.1 and 3.4.1.4.2, APPLICABILITY.

NNECO's Response:

For the APPLICABILITY statement of 3.4.1.4.1 and 3.4.1.4.2 add the words "less than two" so that it reads "Mode 5 with less than two Reactor Coolant loops filled".

This provides the necessary flexibility for Millstone to exercise its designed ability to isolate loops using its Reactor Coolant loop isolation valves. The RHR suction and discharge penetrations are between the reactor vessel and the reactor coolant isolation valves.

ADDITIONAL REVIEW REQUIRED

Item 15: 4.4.3.3 Pressurizer

Surveillance 4.4.3.3 would require that the pressurizer heaters be switched "from the normal to the emergency power supply" at least once per 18 months.

NNECO's Response:

This surveillance is not applicable to the Millstone Unit 3 design. There are five separate groups of pressurizer heaters and each group is capable of being powered from one and only one source. Heater groups 3RCS*H1A and 3RCS*H1B (each rated at 346 KW) are powered from 480 volt load centers, orange and purple respectively. These load centers are Class 1E sources. The remaining groups (3RCS-H1C (150 KW), 3RCS-H1D (346 KW), and 3RCS-H1E (346 KW)) are each powered independently from a non-vital 480 V load center.

ADDITIONAL REVIEW REQUIRED

Item 20: Table 3.3-1, Reactor Trip System Instrumentation.

Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements.

Discuss the four source range monitors at Millstone 3 (Gammametric Fission Chambers).

NNECO's Response:

Millstone Unit 3 has two Gamametric Fission Chambers installed on opposite sides of the core to provide Neutron Accident Monitoring Instrumentation (AMI). These detectors are sensitive down into the source range and provide adequate indication for operations personnel. Therefore, the use of the AMI Neutron Monitors in Modes 3 through 6 should be allowed. A revised copy of Table 3.3-1 and Table 4.3-1 are attached to reflect the use of AMI Neutron Monitors in Modes 3 through 5.

The wording of Specification 3.9.2 should be modified to say " two Source Range Indicating Neutron Flux Monitors".

When used to meet the requirements of Specification 3.3-1 the AMI Neutron Monitors will be connected to the High Flux at Shutdown Alarm and the Reactor Trip Breakers shall be open.

When used to meet the requirements of Specification 3.9.2 the AMI Neutron Monitors shall provide audible indication.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	11
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	~~2~~ 4	0	1	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	11
7. Overtemperature ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	4	1**	3	1, 2	9
8. Overpower ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	4	1**	3	1, 2	9
9. Pressurizer Pressure--Low	4	2	3	1	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1	7#

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TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the Interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Monthly surveillance shall include verification of the ~~Boron Dilution Alarm Setpoint of less than or equal to~~ (~~an increase of twice the count rate within a 10-minute period~~).

high flux at shutdown alarm of less than or equal to 5x background. When the neutron accident monitors are being used to meet the provisions of this specification, Monthly surveillance shall include include verification of the high flux at shutdown alarm of ~~less~~ than or equal to 5x background.

ADDITIONAL REVIEW REQUIRED

Item 21: Table 3.3.1 and Table 4.3-1.

Discuss Functional Unit #15a and 15b, Turbine Trip.

NNECO's Response:

The APPLICABLE MODES and TRIP MODES FOR WHICH SURVEILLANCE IS REQUIRED for Turbine Trip should have a superscript symbol to denote that 15.a and 15.b are only applicable with power above P-9 setpoint. This symbol should be included under Table Notations with the statement "With power above P-9".

ADDITIONAL REVIEW REQUIRED

Item 22: Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements. Review which operational test would more adequately surveil the RCP low shaft speed trip.

NNECO's Response:

It has been determined that the ANALOG CHANNEL OPERATIONAL TEST provides a better demonstration of system operability.

ADDITIONAL REVIEW REQUIRED

Item 24: Table 3.3-3 Engineered Safety Features Actuation System Instrumentation;

Table 3.3-4 Engineered Safety Features Activation System Instrumentation Trip Setpoints;

Table 3.3-5 Engineered Safety Features Response Times;

Table 4.3-2 Engineered Safety Features Actuation System Instrumentation Surveillance Requirements.

These tables would require surveilling the containment purge and exhaust isolation valves as part of the Engineered Safety Feature system.

NNECO's Response:

Considering the purge and exhaust isolation, an ESF function is not consistent with the Millstone Unit 3 design. The only automatic isolation signal to these valves is a high radiation closure. This function is surveilled in accordance with 4.6.3.2.c. The signal is generated at the radiation monitor and is transmitted directly to respective valve control circuit without being processed by Solid State Protection. In addition, specification 3/4.6.1.7 requires that these valves be locked shut for operation in Modes 1 through 4. Therefore, references to the containment purge and exhaust isolation should be removed from the ESF tables.

ADDITIONAL REVIEW REQUIRED

Item 29: 3.4.9.3, Overpressure Protection System

Justify NNECO's submittal using COPS and RHR relief valves.

NNECO's Response:

RCS Overpressure Protection at Low Temperature (vs. tech. spec. 3.4.9.3)

As described in FSAR sections 5.2.2 and 5.2.2.11, and SER (NUREG 1031) section 5.2.2.2, the reactor coolant system (RCS) is protected by pressurizer power operated relief valves (PORV's) and residual heat removal (RHR) reliefs at low temperature operation. The PORV's regulate RCS pressure by responding to open and close signals provided by a manually initiated automatic control system. This control system regulates RCS pressure within limits shown on technical specification Figure 3.4.-4a & b. Because of this range of control, specific to the Millstone 3 design, it is inappropriate to reference a maximum PORV setpoint pressure in technical specifications. NNECO requests the NRC reconsider its original technical specification proposal with respect to the PORV's.

When the RHR system is in operation during low temperature water solid RCS conditions, the RHR liquid relief valves also provide overpressure protection by design. Credit for this means of RCS pressure control should be acknowledged in MP-3 technical specification as originally presented in our December 7, 1984 submittal.

3.4.9.3.a

The RHR relief valves, which help to protect the RCS against overpressurization at low temperature operation, have a setpoint of 450 psig (per SER section 5.2.2.2 and FSAR section 5.4.7.2.4.

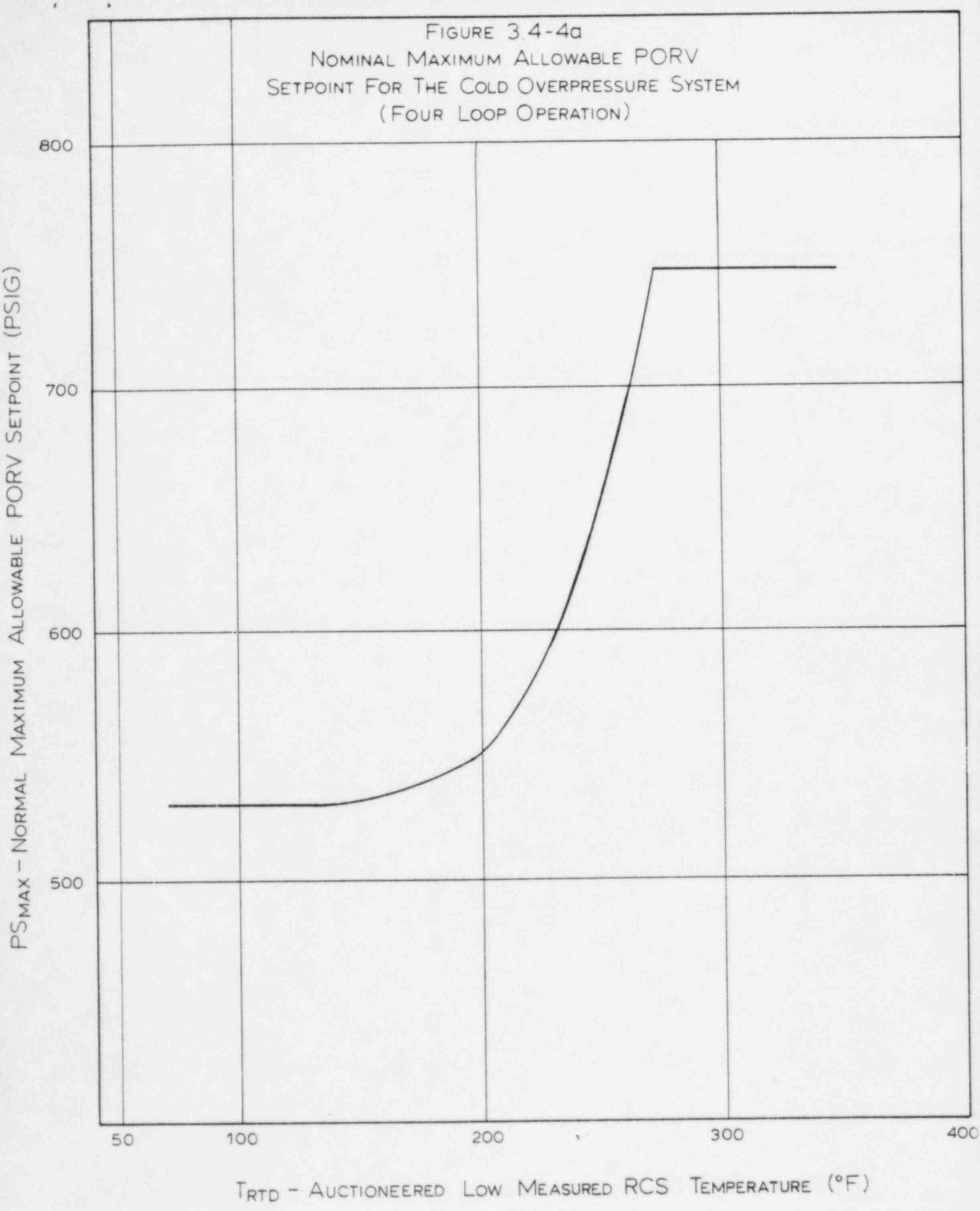
FIGURE 3.4-4a
NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM
(FOUR LOOP OPERATION)
800
700
600
500
PSMAX - NORMAL MAXIMUM ALLOWABLE PORV SETPOINT (PSIG)
50
100
200
300
400
TRTD - AUCTIONEERED LOW MEASURED RCS TEMPERATURE (°F)

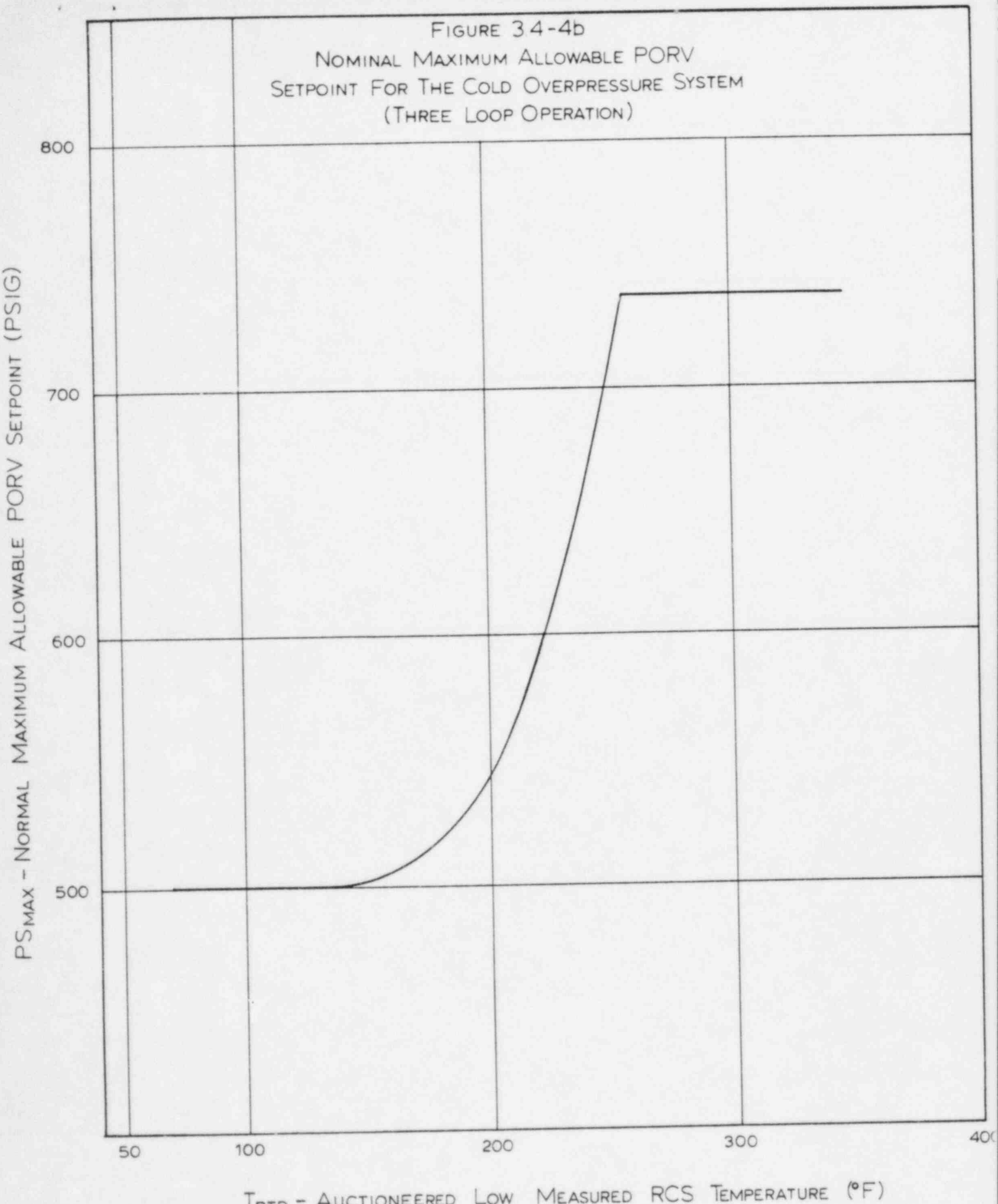
FIGURE 3.4-4b
NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM
(THREE LOOP OPERATION)
800
700
600
500
50
100
200
300
400
PSMAX - NORMAL MAXIMUM ALLOWABLE PORV SETPOINT (PSIG)
TRTD - AUCTIONEERED LOW MEASURED RCS TEMPERATURE (°F)

ADDITIONAL REVIEW REQUESTED

Item 31: Justify deletion of 4.4.4.3

4.4.4.3 - The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

NNECO's Proposed Change:

The surveillance requirements (4.4.4.3) for the PORVs and block valves are not applicable to Millstone Unit 3 because the PORVs and block valves are powered only from the Class 1E AC power system (Ref. FSAR Section 5.4.13.2) The PORVs and block valves are included in the Inservice Testing Program and are tested in accordance with those requirements.

ADDITIONAL REVIEW REQUIRED

Item 33: 3.4.11 Reactor Coolant System Vents Explain Millstone 3 specific design.

NNECO's Response:

As discussed in tech. spec. 3.4.11, the MP-3 reactor coolant system (RCS) has high point vents on the reactor vessel head and the pressurizer (venting the steam space). All vent paths incorporate power operated vent valves in series with solenoid or motor operated block valves. Redundant vent paths are provided. Valve actuator electrical power is supplied from diverse Class 1E vital sources.

The Westinghouse NSSS does not incorporate RCS hot leg high point venting capability. This is because the Westinghouse design used at MP-3 does not use the once through steam generators typical of the Babcock and Wilcox (B&W) NSSS. In the B&W design, the hot leg rise up over the top of the steam generators forming potential gas pockets. The MP-3 hot legs enter at the bases of the steam generators, making the high point at the generators the top of the U-tubes. Venting of all of these tubes is not practical.

Therefore, the vessel head and pressurizer steam space venting provided at MP-3 represents the maximum high point RCS venting possible with this NSSS design.

ADDITIONAL REVIEW REQUIRED

<u>Item 41:</u> 4.6.2.3 The Spray Additive System shall be demonstrated OPERABLE:

d. At least once per 5 years by verifying each solution flow rate (to be determined during preoperational tests) from the following drain connections in the Spray Additive System:

1) Drain line location $\pm$ gpm, and
2) Drain line location $\pm$ gpm.

<u>NNECO's Proposed Change:</u>

NNECO requests the deletion of surveillance 4.6.2.3.d.

The chemical addition tank (CAT)/refueling water storage tank (RWST) system is aligned such that valves V-29 and V-32 downstream of the CAT are locked open. Surveillance 4.6.2.3.a requires once per month that these valves actuate to their correct position once every 18 months. The above two surveillances adequately verify the system flow path with respect to any mechanical interferences.

Additionally, the gravity feed line from the CAT through the block valves, into the quench spray supply is a 304 Stainless Steel Line of standard thickness. According to the Corrosion Resistance Tables pg. 1036 by Philip A. Schweitzer, 304 Stainless Steel pipe exhibits excellent corrosion resistant capabilities in a 10% solution of NaOH in temperatures up to 220^{o}F. The CAT has a maximum of 2% NaoH solution at an operating temperature of 40^{o}F. Therefore, system flow will not be inhibited by chemical/corrosion debris. The periodic examination of both valve positions and actuations combined with no corrosion problems will adequately ensure the CAT flowpath remains operable.

ADDITIONAL REVIEW REQUIRED

Item 43: 4.6.4.2.b.2 Electric Hydrogen Recombiner Surveillance

Justify modification to 18 months visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.).

NNECO's proposed revision and justification:

Surveillance 4.6.4.2.b.2 could be mis-interpreted as a requirement to dismantle portions of the hydrogen recombiners every eighteen months. Therefore, NNECO has proposed a clarification of this surveillance. The intent remains the same, to visually examine the recombiners for signs of abnormal conditions (i.e. loose wiring, structural connections, deposits of foreign material, etc.).

To further demonstrate that the environmental integrity of the recombiners will remain, the following information is supplied.

1) The hydrogen recombiners are located in a separate dedicated building and are connected to containment only by 2" penetrations.

2) The recombiner cubicles are supplied with centifugal fans, air and fire dampers. Since the recombiner cubicles are not normally open to containment, the recombiners are not subject to either the harsh environment of containment (i.e. caustic spray, 100% humidity etc.) or normal operating conditions.

3) The cubicles are provided with controls and instrumentation to monitor temperature, radiation, ventilation along with numerous recombiner alarms. Thus abnormal conditions in the recombiner enclosure will not occur without activation of alarms and subsequent correction by plant operators.

4) The recombiner system is functionally tested every six months and instruments and control circuits are calibrated every 18 months. Thus, degradation, if it should occur, would be determined during this surveillance testing.

In conclusion, a visual examination of the recombiners is sufficient due to the provisions provided by the dedicated recombiner cubicle and other recombiner surveillances.

We propose that a more meaningful inspection be conducted which would complement the other tests/inspections contained in 4.6.4.2.b. The proposed specification should read:

Item 43 continued:

4.6.4

b. At least once per 18 months by:

2) Verifying through a visual examination of all external recombiner electrical connections that there is no evidence of abnormal conditions (i.e. loose wiring structural connections, deposits of foreign material, etc). and

ADDITIONAL REVIEW REQUIRED

Item 70: 4.7.11.1.1.c - Review exception on valves inside containment (Also 4.7.11.2.a)

NNECO's Response:

Add the following Tech. Spec. 4.7.11.2.c and 4.7.11.1.1.d. "At least once per 18 months by verifying that each valve inside of containment in the flow path is in its correct position."

ADDITIONAL REVIEW REQUIRED

Item 71: 4.7.11.1.2.b Fire Pump Diesel Engine

Justify standard years for consistency with Units 1 and 2.

NNECO's Response:

The sampling requirements for the fire pump diesel fuel oil, present in the existing technical specification for Millstone Unit 1 and 2 require performance in accordance with ASTM-D270-65 and values within the limits of Table 1, ASTM-D975-74. The NRC has proposed in the Proof and Review copy of the Technical Specifications that the technical specifications be changed and ASTM-D270-75 and ASTM-D975-77 be used respectively. The change is unnecessary in that the limits utilized are identical to those proposed.

	Grade of Fuel Oil	Saybolt Viscosity (SUS at 100ºF)		Water & Sediment (Vol %)
		Min	Max	Max
	10	---	34.4	0.05
D975-74	20	32.6	40.1	0.05
	40	45.0	125.0	0.50
	10	---	34.4	0.05
D975-77	20	32.6	40.1	0.05
	40	45.0	125.0	0.50

In addition, the sampling procedures (ASTM D270-65 and -75) are very similar with the only difference being the minimal additional guidance provided in the later revision.

ADDITIONAL REVIEW REQUIRED

Item 74: 3.7.13.c.2, Perform Engineering Evaluation of the affected equipment to demonstrate continued operability of the equipment until the temperature can be restored to within its limits, or

Explain the words "Engineering Evaluation"

NNECO's Response:

The Engineering Evaluation will consist of reviewing the total time and temperature excursion above the designated temperature parameter specified on the SCEW sheet. The computer program will calculate the amount of life lost at the elevated temperature and time excursion. This life (Q lost) will be subtracted from the original Qualified Life (QLo) the difference will be the New Qualified Life (QLn).

Therefore, in equation form it can be represented as follows;

$$QLn = QLo - Qlost$$

The SCEW sheets for the affected equipment will be revised to indicate the new Qualified Life (QLn). The new QLn will indicate on early change out schedule.

ADDITIONAL REVIEW REQUIRED

Item 75: 4.8.1.1.2, A. C. Sources

Justify 4160 $\pm$ 420 volts

NNECO's Response:

The 4160 volt switchgear complies with the nominal $\pm$ 10% voltage range as called out in the Switchgear Standards, ANSI C37.04. Therefore, $\pm$ 420 volts is correct to operate the equipment.

ADDITIONAL REVIEW REQUIRED

Item 77: 4.8.1.1.2.f.13, Diesel Generator Surveillance, lock-out features.

NNECO's Response:

The list of lockout features provided in the Proof and Review copy meet the intent of this specification. If the diesel generator were to trip while running the lockout features listed would prevent engine restart.

ADDITIONAL REVIEW REQUIRED

Item 85: 3/4.9.8.1 and 3/4.9.8.2, RHR and Coolant Circulation

Provide a comparison between the Standard Technical Specification (STS) versions and that proposed by NNECO.

NNECO's Response:

The intent of the STS version is to a) require that there be RHR loop flow at any reactor water level, and b) require that a second RHR loop be OPERABLE as a backup when the reactor water level is less than 23 feet above the vessel flange. The proposed specification is consistent with this intent. (A copy of the proposed specification is attached for convenience.) Proposed LCO 3.9.8.1 would require that one RHR loop be OPERABLE and in operation; and the action statement is consistent with the corresponding STS action statements. Proposed LCO 3.9.8.2 would require that two independent RHR loops be OPERABLE as required by the applicability statement. The applicability statement is designed to require two RHR loops whenever the immediate capability to raise the reactor vessel/refuel pool water level to 23 feet above the flange does not exist. The proposed action statement is consistent with action a of 3.4.1.4.2 (RHR loop requirements when in Mode 5 and RCS loops not filled). The proposed surveillance more adequately demonstrates OPERABILITY.

The proposed specification is desired since it incorporates a great deal of flexibility with respect to RHR loop maintenance and ISI while providing an equivalent level of safety. It allows removing a loop from service without requiring an immediate flood up of the refuel pool as long as the requisites for delivering water to the pool are met. This can represent valuable schedular flexibility during a refueling outage and provides for enhanced safety by allowing optimum scheduling of safety system outages.

Item 85:

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, ~~when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet~~. at all reactor water levels.

ACTION:

With no RHR loop OPERABLE or in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 5 2800 5 gpm at least once per 12 hours.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

Item 85:

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REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, ~~and at least one RHR loop shall be in operation.~~

APPLICABILITY: ~~MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.~~ INSERT A

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, ~~or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.~~

~~b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.~~

SURVEILLANCE REQUIREMENTS

4.9.8.2 ~~At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to [2800] gpm at least once per 12 hours.~~

INSERT B

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

~~Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.~~

Item 85

Insert A

MODE 6, whenever all the following conditions are not satisfied:

a. reactor vessel water level at or above the vessel change;

b. the reactor vessel pit seal installed;

c. the combined available volume of water in the refuel pool and refueling water storage tank exceed 350,000 gallons; and

d. (1) one RHR pump not in residual heat removal service and aligned to take suction from the RWST and deliver flow to the RCS is OPERABLE: or

(2) one safety injection pump aligned to take suction from the RWST and deliver flow to the RCS is operable; or

(3) one charging pump aligned to take suction from the RWST and deliver flow to the RCS in OPERABLE.

Item 85

Insert B

4.9.8.2 Once per 7 days, the required RHR loops, if not in operation, shall be determined OPERABLE by verifying correct breaker alignment and indicated power availability for pump and RHR cooling valves, or:

Verifying that the reactor vessel water level is at or above the vessel flange, the reactor vessel pit seal is installed, and greater than 350,000 gallons of water is available as a heat sink, as indicated by either:

a. refuel pool level grater than 23 feet above the reactor vessel flange, or

b. the combined available volume of the refuel pool and refueling water storage tanks exceeds 350,000 gallons and a flow path available from the refueling water storage tank to the refuel pool.

Question No. Q430.51 (SRP Sections 8.3.1 and 8.3.2)

Identify all electrical equipment, both safety and nonsafety, that may become submerged as a result of a LOCA. For all such equipment that is not designed and qualified for service in such an environment, provide analysis to determine the following:

1. The safety significance of the failure of this electrical equipment (e.g. spurious actuation or loss of actuation function) as a result of flooding.

2. The effects on Class 1E power sources serving this equipment as a result of such submergence.

3. Any proposed design changes resulting from this analysis.

Response:

There is no safety- or nonsafety-related electrical equipment, which is required post-LOCA or whose failure position will affect station shutdown capability, located inside the containment that may be submerged.

The following safety-related equipment is connected to a Class 1E power supply and is located inside the containment. It may become submerged as a result of a LOCA but is not designed and qualified for submergence:

<u>Group One</u>

3SIL*MV8808A
3SIL*MV8808B
3SIL*MV8808C
3SIL*MV8808D

Each of the circuits for Group One equipment is deenergized during normal plant operation. Each of the circuits for Group Two equipment is provided with two series connected interrupting devices which meet the requirements of Regulatory Guide 1.75 for an isolation device and regulatory Guide 1.63 for penetration protection.

There is nonsafety-related electrical equipment connected to Class 1E power supplies, located inside the containment, which may become submerged as a result of a LOCA and which are not designed and qualified for submergence. Each of the circuits for this equipment is provided with two series connected interrupting devices which meet the requirements of Regulatory Guide 1.75 for an isolation device and Regulatory Guide 1.63 for penetration protection.

There is nonsafety-related electrical equipment connected to the non-Class 1E power supplies, located inside the containment, which may become submerged as a result of a LOCA and which is not designed and qualified for submergence. These circuits are not powered from Class 1E power supplies. Moreover, where the available fault current excees the current carrying capability of the

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Question No. Q430.51

penetration conductors, secondary (i.e. backup) penetration protection is provided in addition to the normal circuit protection (i.e., primary penetration protection) in accordance with Regulatory Guide 1.63.