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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July 22, 1983

Docket No. 50-423
B10845

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

References: (1) B. J. Youngblood to W. G. Counsil, Request for Additional Information for Millstone Nuclear Power Station, Unit No. 3, dated, May 3, 1983.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3:
Response to Select Requests for Additional Information

In order to assist the NRC in the review of our responses to questions contained in Reference (1), the responses to those questions listed in Attachment 1 are being forwarded in advance of the August 1, 1983 due date. Responses to the remaining questions will be forwarded on August 1, 1983. Responses provided now and those provided on August 1, 1983 have been or will be provided as they will appear in Amendment 3 to our OL application. On or before September 1, 1983, the required 60 copies of Amendment 3 will be forwarded to you for insertion into your FSAR sets.

Because our response to Reference (1) is being forwarded via two (2) transmittals some of the revised FSAR pages associated with responses contained herein are common to those to be forwarded on August 1, 1983. Such revisions should not be construed as final until they are forwarded on August 1, 1983.

B001
1/60

If you have any concerns related to commitments contained herein or any questions related to our responses, please contact our licensing representative directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL
By NORTHEAST NUCLEAR ENERGY COMPANY, Their Agent

W. G. Counsil

W. G. Counsil
Senior Vice President

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

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensees herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico

Notary Public

My Commission Expires March 31, 1988

Attachment I

List of Provided Responses

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NRC Letter: May 3, 1983

Question No. Q220.10 (SRP Section 3.5.3)

Is there any concrete barrier whose thickness is less than that shown in Table 1 of SRP Section 3.5.3? If yes, please identify and justify them.

Response:

FSAR Section 3.5.1.4 (Amendment 2) gives the minimum concrete barrier thickness. Barriers are not less than those values given in SRP Section 3.5.3.

NRC Letter: May 3, 1983

Question No. Q220.16 (SRP Section 3.7.1)

FSAR Section 3.7B.1.1 states that Regulatory Guide 1.60 spectra are not used. Does this mean that "site specific spectra" have been developed for this plant? If so, have these site specific spectra been reviewed and approved by the Geosciences Branch of NRC?

Response:

Refer to FSAR Table 1.8-1 under Regulatory Guide 1.60 for the justification for taking exception to this Regulatory Guide.

NRC Letter: May 3, 1983

Question No. Q220.18 (SRP Section 3.7.1)

The format and some of the percent of critical damping values of Table 3.7B-1 are different from that of Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Either revise the contents in Table 3.7B-1 to comply with Regulatory Guide 1.61 or provide justifications for the deviations, as stated in SRP Section 3.7.1II.2.

Response:

Refer to FSAR Table 1.8-1 under Regulatory Guide 1.61 for the justification for taking exception to this Regulatory Guide.

NRC Letter: May 3, 1983

Question No. Q220.19 (SRP Section 3.7.2)

SRP Section 3.7.2.II.4 "Soil-Structure Interaction" requires that two different ways of modeling the supporting soil media of a soil-structure interaction system be considered: half-space and finite boundary, and then envelop response results in structures for the use of designing Seismic Category I structures, systems, and components. Please revise your FSAR to comply with these requirements.

Response:

Refer to FSAR Section 1.9, Table 1.9-2, SRP 3.7.2, A.1 and B.1.

NRC Letter: May 3, 1983

Question No. Q220.20 (SRP Section 3.7.2)

FSAR Section 3.7B.2.9 states that floor response spectra for the cracked and uncracked cases were enveloped. Provide the criteria that was used to determine the cracked or uncracked cases.

Response:

The member stiffness properties of the containment shell for the uncracked case were determined based on full concrete dimensions, homogeneous isotropic concrete and linear-elastic beam theory, including the effects of shear deformation.

For the cracked case, the containment structure shell was assigned one half the uncracked stiffness as a reasonable estimate for the member stiffness properties of cracked concrete.

NRC Letter: May 3, 1983

Question No. Q220.25 (SRP Section 3.7.2)

FSAR Section 3.7B.2.2.1 mentioned "the amplified response spectra (ARS)". Please define the new term and explain how do they differ from "design response spectra" and "floor response spectra"?

Response:

The term "amplified response spectra (ARS)" used in FSAR Section 3.7B.2.2.1 is the same as "floor response spectra" described in FSAR Section 3.7B.2.5. Design response spectra is defined in Section 3.7B.1.1.

NRC Letter: May 3, 1983

Question No. Q220.26 (SRP Section 3.8.1)

FSAR Section 3.8.1.3.1 states that the allowable compressive stress in concrete is 0.45fc'. However, the ASME Section III, Division II Code allowable for primary membrane is only 0.30fc'. Please justify the deviation.

Response:

Refer to FSAR Table 1.9-2, Amendment 1 under SPR 3.8.1, Item 3 for the response to this question.

NRC Letter: May 3, 1983

Question No. Q220.31 (SRP Section 3.8.1)

Provide temperature profiles that were used for containment (SRP 3.8.1) thermal analysis for both operating and accident conditions.

Response:

Refer to FSAR Appendix 3B Attachment 1, Appendix A, page 3 of 6 for operating condition temperature profiles.

For accident conditions a temperature of 280°F for 1 hour has been used.

NRC Letter: May 3, 1983

Question No. [illegible].32 (SRP Section 3.8.1)

FSAR Section 3.8.1.5.1 states that design of the containment equals or exceeds ACI 318-71 requirements for serviceability. Since the ACI 318 code is for conventional building structures, not for containment structures, we fail to see the connection between containment serviceability and conventional building serviceability. Please list the requirements for serviceability and explain the connection.

Response:

Serviceability of containment refers to the requirements for weathering, crack control, and displacements of the structure at service loads. For service loads, the behavior of the containment structure is similar to the behavior of any other concrete structure, so ACI 318-71 provides adequate guidance.

NRC Letter: May 3, 1983

Question No. Q220.33 (SRP Section 3.8.1)

SRP Section 3.8.1.II.4 requires that an analysis should be performed to determine the ultimate capacity of the containment. Please revise the FSAR to comply with this requirement.

Response:

Refer to FSAR Table 1.9-2, Amendment 1 under SRP 3.8.1, Item 2 for the response to this question.

NRC Letter: May 3, 1983

Question No. Q220.38 (SRP Section 3.8.4)

Are any safety related masonry walls in the plant? If yes, revise the FSAR to comply with the requirements in Appendix A to SRP Section 3.8.4.

Response:

Refer to revised FSAR Section 3.8.4.8 for the response to this question.

TABLE 1.9-1 (Cont)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section	
3.8.4 (Rev. 1)	II.2 - ACI 349-76.	ACI 349-76 was not used.	3.8.4.2, 3.8.1.2	
	II.4.d - Design report format.	FSAR does not use this format.	3.8.4	220.38
3.8.5 (Rev. 1)	II.4.b - ACI 349-76.	ACI 318-71 was used rather than ACI 349-76.	3.8.5.2, 3.8.1.2	
3.9.1 (Rev. 2) (BOP Scope)	III.1 - Plant conditions identified as design levels A,B,C,D.	FSAR identifies plant conditions as normal, upset, emergency, and faulted.	3.9B.1.1	
	III.4 - Methods used in stress analysis of components.	FSAR contains no justification for methods used.	3.9B.1.4	
3.9.1 (Rev. 2) (NSSS Scope)	II.2 - Computer codes used in design and analysis of siesmic Category I components.	Only a brief description of computer codes used by Westinghouse is given.	3.9N.1.2	
3.9.2 (Rev. 2) (BOP Scope)	II.1.d - List snubbers on systems which experience sufficient thermal expansion.	FSAR does not provide a list of snubbers.	3.9B.2	
	II.1.e and f - Tests to verify thermal expansion/vibration measurements.	FSAR does not provide a description of tests.	3.9B.2	
	II.2 - Seismic subsystem analysis.	Information is not contained in FSAR Section 3.9B.2.	3.9B.3	
3.9.2 (Rev. 2) (NSSS Scope)	II.2.e - Criteria for combining closely spaced modes.	Westinghouse method is provided in FSAR Section 3.7N.3.7.	3.7N.3.7	
3.9.3 (Rev. 1) (BOP Scope)	II.1 - Stress limit criteria.	FSAR does not reflect the stress limit criteria.	3.9B.3.1	
	II.2 - Information on Class 3 safety/relief devices.	FSAR does not address Class 3 safety/relief devices.	3.9B.3	
	II.3 - Information on snubbers.	Requirements not addressed in FSAR.	3.9B.3	

TABLE 1.9-2 (Cont)

3. Millstone 3 did not use Article 3000 of ASME III, Division 2 for loads, load combinations, and stress allowables as described in SRP 3.8.1, Paragraph II.5.

4. Millstone 3 did not use ASME III, Division 2, Article CC-3000 for the analysis and design of the containment structure tangential shear as described in SRP 3.8.1, Paragraph II.4.f. 220.30

B. Justification for differences from SRP

1. Regulatory Guide 1.136 does not apply to Millstone 3. See FSAR Section 1.8 for position on Reg. Guide 1.136.

2. The ultimate capacity of the reactor containment with respect to failure modes has been considered in the PRA study, which will be submitted as a separate report.

3. ASME III, Division 2, was not available at the time of the Millstone 3 Construction Permit (CP). ACI 318 and AISC-1969 Ed. were the codes used. ASME III, 1971 Ed., with Addenda thrugh Summer 1973, Subsections NC and NE were used as a guide. Guidance found in 10CFR50 regulation does not require continuous upgrading of the codes and standards used in the design. 220.29

4. ASME III, Division 2, was not available at the time of the Millstone 3 Construction Permit. The procedure used for analysis and design of the containment structure tangential shear, as described in FSAR Section 3.8.1.4.1, meets the intent of SRP Section 3.8.1, Paragraph II.4.f. 220.30

SRP 3.8.3

SRP TITLE: CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

A. Actual differences between FSAR and SRP

ACI 349-76 was not used as described in SRP 3.8.3, Paragraph II.2.

B. Justification for differences from SRP

This code was not in effect at the time of the Construction Permit. ACI 318, AISC-1969 Ed. and ASME III 1971 Ed. through Summer 1973 addenda were the codes used. Guidance found in 10CFR50 regulation does not require continuous upgrading of the codes and standards used in the design.

TABLE 1.9-2 (Cont)

SRP 3.8.4

SRP TITLE: OTHER SEISMIC CATEGORY I STRUCTURES

A. Actual differences between FSAR and SRP

1. ACI 349-76 was not used during the design stage of Millstone 3 as described in SRP 3.8.4, Paragraph II.2.

2. SRP 3.8.4, Paragraph II.4.d addresses the use of the design report format presented in Appendix C to this SRP. Our design
220.38 information is not in this format.

B. Justification for differences from SRP

1. The ASME III 1971 Ed. through Summer 1973 addenda and AISC-1969 Ed. codes were in effect during the design stage of Millstone 3. Guidance found in 10CFR50 regulation does not require continuous upgrading of the codes and standards used in the design.

2. The material described in Appendix C of this SRP can be found in the design criteria and design calculations which are contained in an auditable file located at the Millstone 3 site.

220.38

SRP 3.8.5

SRP TITLE: FOUNDATIONS

A. Actual differences between FSAR and SRP

ACI 318-71 was used rather than ACI 349-76 as specified in SRP 3.8.5, Paragraph II.4.b.

B. Justification for differences from SRP

ACI 349-76 was not in effect at the time the construction permit was issued. Guidance found in 10CFR50 regulation does not require continuous upgrading of the codes and standards used in the design.

SRP 3.9.1

SRP TITLE: SPECIAL TOPICS FOR MECHANICAL COMPONENTS

A. Actual differences between FSAR and SRP (BOP Scope)

1. FSAR Section 3.9B.[illegible].1 identifies plant conditions as normal, upset, emergency, and faulted, whereas SRP 3.9.1, Paragraph III.1, requires them to be identified as design level A, B, C, and D.

TABLE 1.9-2 (Cont)

Also, allowables used in stress analysis are not based on service limits.

2. SRP 3.9.1, Paragraph III.4 requires the FSAR to include justifications as well as the demonstration of acceptability of stress strain curves employed.

problem. The only building with significant height, the turbine building, has been checked for wind deflections. These deflections are limited such that overall frame stability is maintained and deflections under service loads are within common practice for the industry.

3.8.4.6 Materials, Operating Control, and Special Construction Techniques

Sections 3.8.1.2 and 3.8.1.6 describe material and quality control. There are no special techniques used in constructing the structures (Section 3.8.4.1).

The 60-day compressive strength of concrete for the spent fuel pool and the fuel building is specified as 5,000 psi.

Section 17.1B describes the quality assurance activities required by this section.

3.8.4.7 Testing and Inservice Surveillance Requirements

There are no special testing or inservice surveillance requirements for Category I structures outside the containment.

3.8.4.8 Masonry Walls

220.38 Masonry walls in safety related areas in the plant comply with the requirements in Appendix A to SRP Section 3.8.4. Locations of walls are given in Figure 3.8-64, Sheet 1 and 3.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

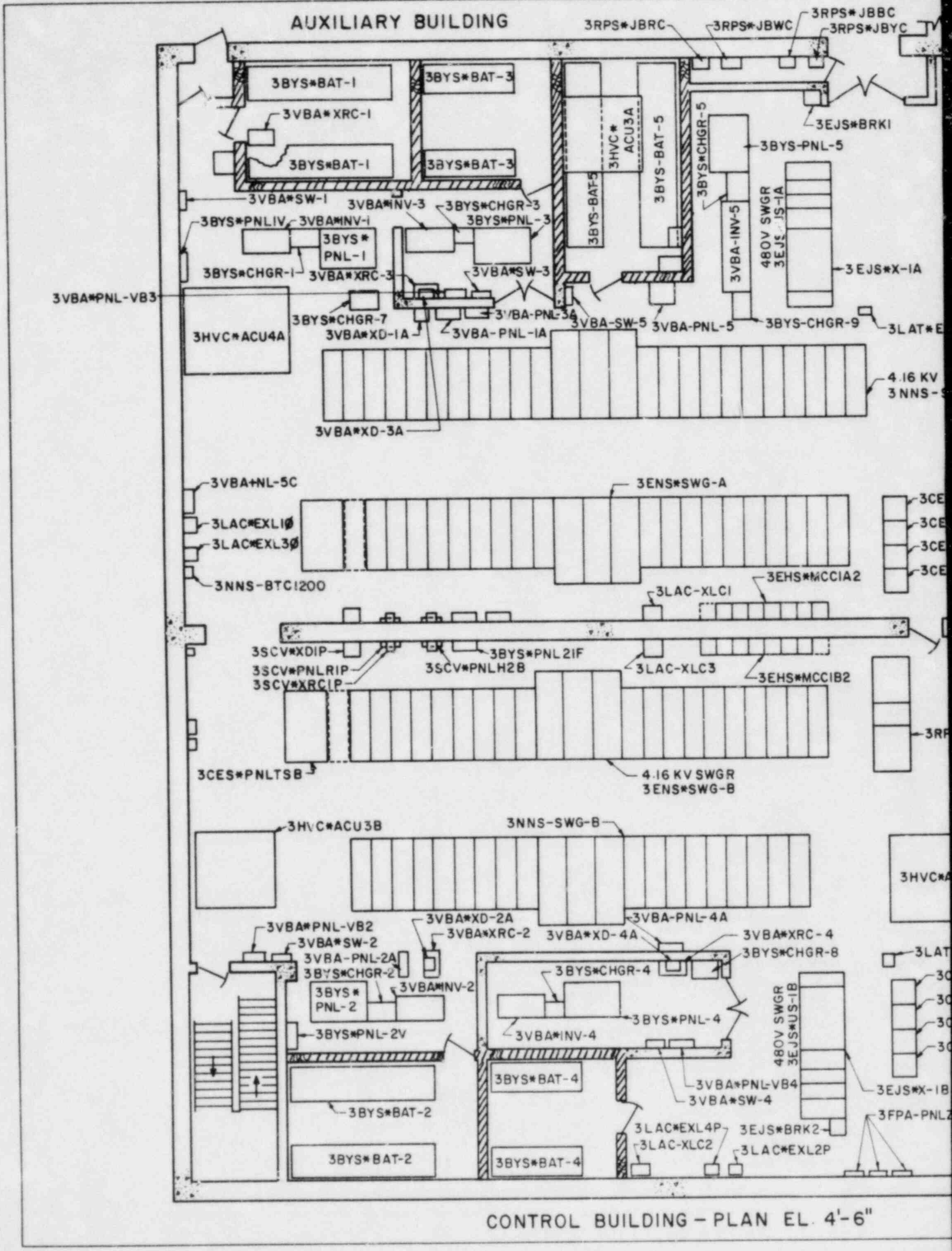
Foundations for all of the major structures consist of soil or rock supported reinforced concrete mats or spread footings as described in Table 2.5.4-14. Figures 2.5.4-1 through 2.5.4-17 show plan and section views of the major foundations.

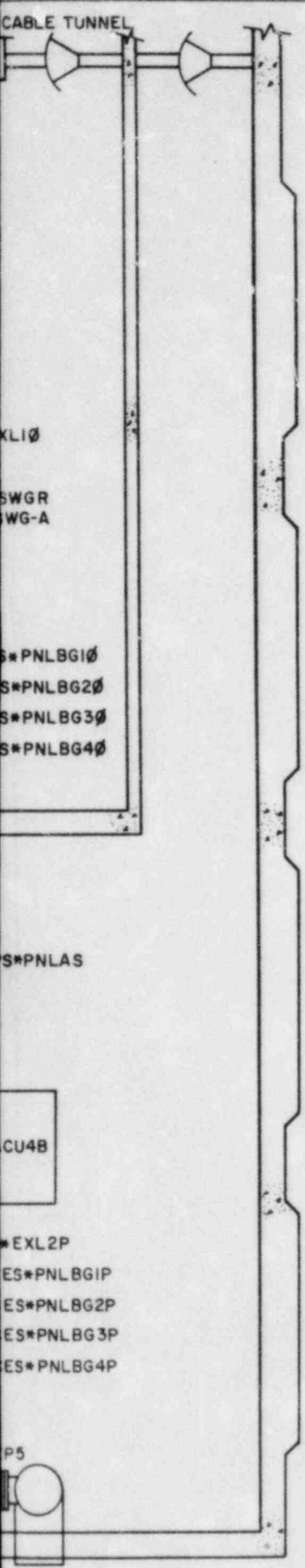
To provide for independent movement of structures during a seismic event, compressible material 1 inch thick is provided below grade and a separation of 2 inches is provided above grade, between all structures. The containment is separated from all adjacent structures by a 4-inch space filled with a compressible material.

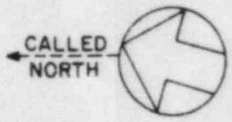
Horizontal shear keys are provided for the control, fuel, and auxiliary buildings and for the circulating and service water pumphouse foundations. Figure 3.8-77 shows the arrangement of these horizontal shear keys and Figure 3.8-78 shows a typical detail. Rock dowels are used in the auxiliary building foundation (Section 3.8.5.5).

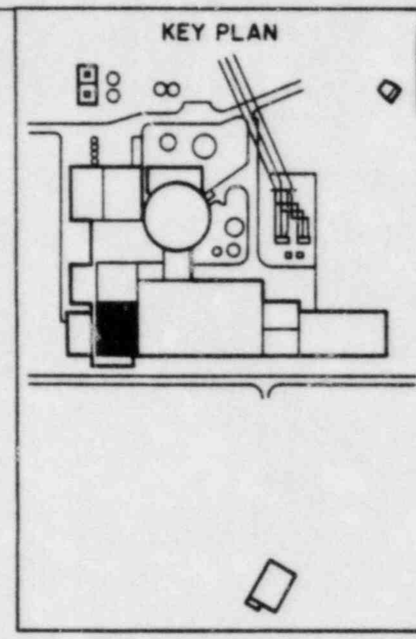
Horizontal shear resulting from seismic acceleration of the containment structure is transferred to the surrounding rock by bearing of the edge of the containment mat.

Rock dowels are used in the exterior walls of the auxiliary building to resist uplift during seismic loading. Section 2.5.4 describes the rock dowels and the installation program.


AUXILIARY BUILDING
3RPS*JBRC
3RPS*JBWC
3RPS*JBBC
3RPS*JBYC
3BYS*BAT-1
3BYS*BAT-3
3VBA*XRC-1
3HVC*ACU3A
3BYS-BAT-5
3BYS*CHGR-5
3BYS-PNL-5
3EJS*BRK1
3VBA*SW-1
3VBA*INV-3
3BYS*CHGR-3
3BYS*PNL-3
3BYS*PNLIV
3VBA*INV-1
3BYS* PNL-1
3BYS*CHGR-1
3VBA*XRC-3
3VBA*SW-3
3VBA-INV-5
480V SWGR 3EJS*US-1A
3EJS*X-1A
3VBA*PNL-VB3
3BYS*CHGR-7
3VBA-PNL-3A
3VBA-SW-5
3VBA-PNL-5
3BYS-CHGR-9
3VBA*XD-1A
3VBA-PNL-1A
3HVC*ACU4A
4.16 KV
3VBA*XD-3A
3VBA*NL-5C
3ENS*SWG-A
3LAC*EXL1Ø
3LAC*EXL3Ø
3NNS-BTC1200
3LAC-XLC1
3EHS*MCC1A2
3SCV*XD1P
3SCV*PNLR1P
3SCV*XRC1P
3SCV*PNLH2B
3BYS*PNL21F
3LAC-XLC3
3EHS*MCC1B2
3CES*PNLTSB
4.16 KV SWGR 3ENS*SWG-B
3HVC*ACU3B
3NNS-SWG-B
3VBA*XD-2A
3VBA-PNL-4A
3VBA*PNL-VB2
3VBA*XRC-2
3VBA*XD-4A
3VBA*XRC-4
3VBA*SW-2
3BYS*CHGR-8
3VBA-PNL-2A
3BYS*CHGR-2
3BYS*CHGR-4
3BYS* PNL-2
3VBA*INV-2
3BYS*PNL-4
480V SWGR 3EJS*US-1B
3BYS*PNL-2V
3VBA*INV-4
3BYS*BAT-4
3VBA*PNL-VB4
3VBA*SW-4
3EJS*X-1B
3BYS*BAT-2
3LAC*EXL4P
3EJS*BRK2
3LAC-XLC2
3LAC*EXL2P
CONTROL BUILDING - PLAN EL. 4'-6"







LEGEND

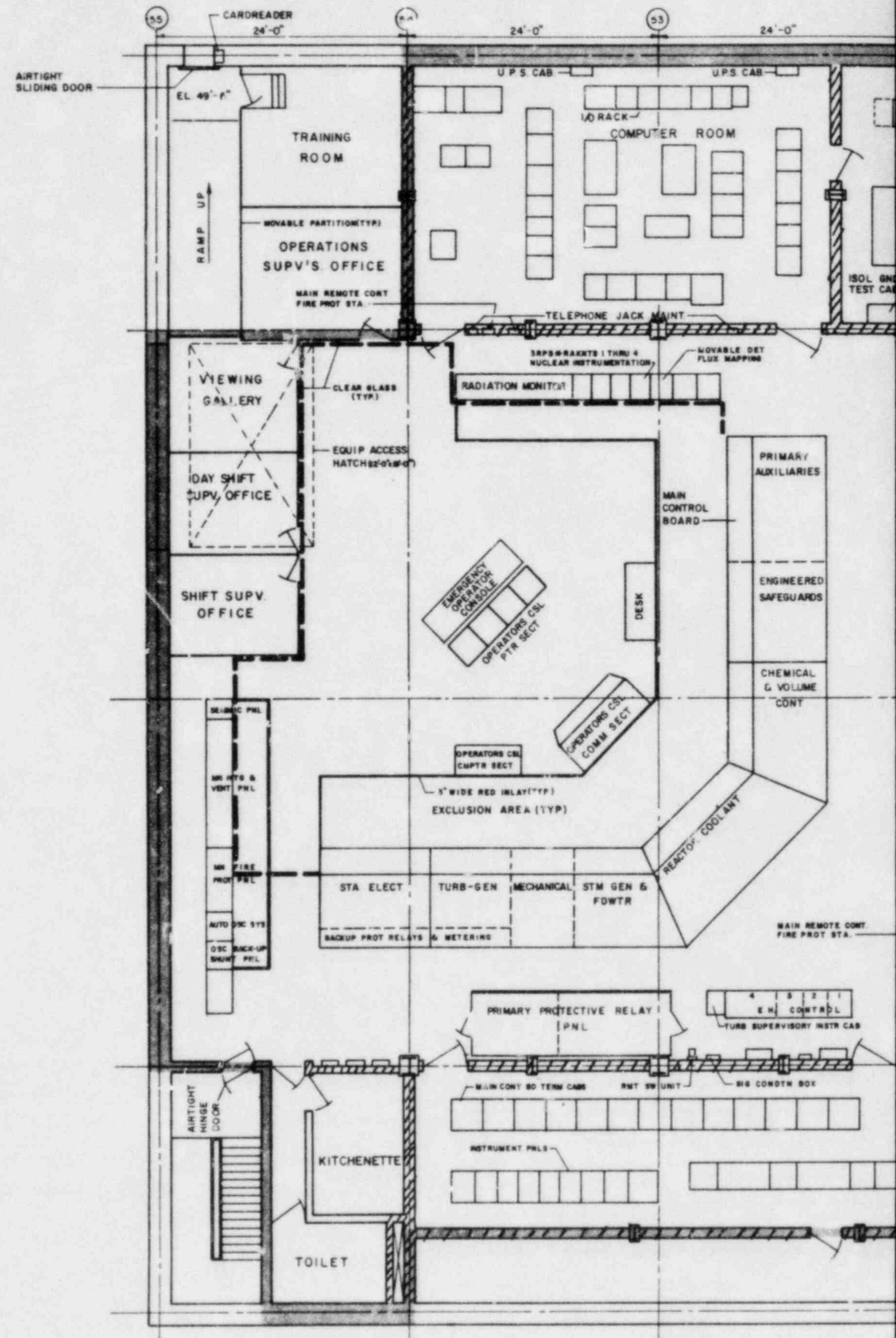
SAFETY RELATED MASONRY WALLS

PRC
APERTURE
CARD

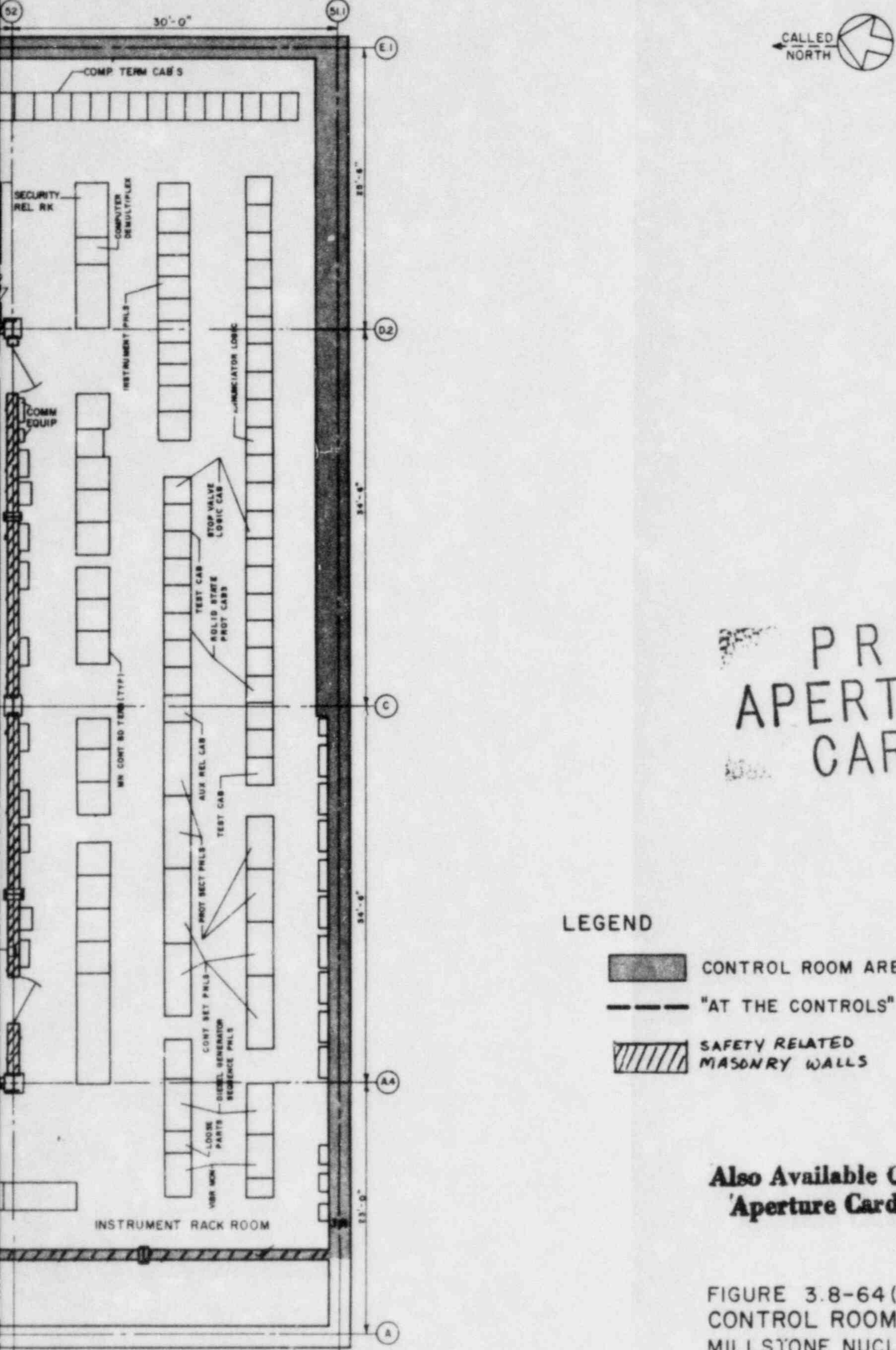
Also Available On Aperture Card

FIGURE 3.8-64 (1 OF 4)
CONTROL BUILDING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

8307260197-01



CONTROL BUILDING
PLAN EL. 47'-6"



PRC APERTURE CARD

LEGEND

CONTROL ROOM AREA

"AT THE CONTROLS"

SAFETY RELATED MASONRY WALLS

Also Available On Aperture Card

FIGURE 3.8-64 (3 OF 4)
CONTROL ROOM AREA
MILLSTONE NUCLEAR POWER PLANT
UNIT 3
FINAL SAFETY ANALYSIS REPORT

8307260197-02

NRC Letter: January 31, 1983

Question No. Q220.7 (Section 3.8.4.8)

Provide a Section 3.8.4.8 that discusses the effects of masonry walls on other structures in accordance with SRP 3.8.4 in NUREG-0800.

Response:

Refer to Question Q220.38 for the response to this question.

NRC Letter: May 3, 1983

Question No. Q240.2 (Sections 2.4.5.2, 2.4.5.3; SRP Section 2.4.5)

Discuss the reason for the difference in PMH maximum stillwater level values cited in the FSAR (19.7 feet msl) and the PSAR, Amendment 14 and the CP-SER (18.2 feet msl).

Response:

The FSAR value of 19.7 feet msl has been adjusted for a 1.3 feet difference in tidal height used. In the PSAR Amendment 14 and the CP-SER, a 1.1 feet msl (2.5 feet mlw) astronomical tide was used. In the FSAR a 2.4 feet msl (3.8 feet mlw) 10 percent exceedance high spring tide was used in accordance with Regulatory Guide 1.59, Revision 2, August 1977. Also included was an adjustment of 0.2 feet for a typographical error.

NRC Letter: May 3, 1983

Question No. Q240.3 (Section 2.4.5.2, SRP Section 2.4.5)

The correct reference for Initial Rise in Regulatory Guide 1.59 is Table C.1 not Table 3.1.3-1.

Response:

The reference for initial rise will be changed to Regulatory Guide 1.59, Table C.1.

Radius to maximum wind	48 nmi
Speed of translation	15 knots
Astronomical tide (10 percent exceedance high tide)	2.4 feet above msl
Initial rise (Regulatory Guide 1.59, Table C.1)	1.0 feet
Bottom friction	0.0025
Wind stress coefficient factor	1.10
Bottom profile (Figure 2.4-8)	
Hurricane track (Figure 2.4-12)	

240.3

Surge analyses based on different types of hurricanes show that the large radius, slow forward speed hurricane produces the maximum stillwater level at the Millstone site.

The resulting maximum surge stillwater level is +19.7 feet msl. Additional surge data, including surge hydrographs for all three large radius storms, are shown on Figures 2.4-9 through 2.4-11.

2.4.5.3 Wave Action

Wave characteristics are dependent upon wind speed and duration, wind direction, fetch length, and water depth. Millstone Point is sheltered from the direct onslaught of open ocean waves by Long Island. Moreover, the unit itself is located on the western side of the Point and a considerable distance (about 2500 feet) inland from the southernmost tip. Thus, the topography of the Point itself protects the unit area from breaking waves during the period of peak tidal flooding when the winds are from the southeast quadrant.

For maximizing hurricane effects, the hurricane track was bent in order to have the maximum wind attack the site for the maximum possible time. The tracks are shown on Figures 2.4-12 through 2.4-14. Because of the location of the site, two possible methods of generating maximum waves, deep- and shallow-water waves, were considered.

2.4.5.3.1 Deep Water Waves

The first method was to generate deep-water waves offshore of the continental shelf and let them propagate over the shelf to Block Island Sound, finally reaching the Millstone location. Two independent analyses, one graphically by Wilson (1955, 1963) and the other computational by Bretschneider (1972) provide comparison for deep water waves.

NRC Letter: May 3, 1983

Question No. Q240.9 (Section 2.4.14, SRP 2.4.14)

In your discussion of the flood protection of the service water cubicles credit was taken for the watertight steel doors. Are these doors normally closed and secured? If so, what alerts operators if they are not secured? If they are not normally secured, discuss flood protection procedures to be taken to secure service water cubicles prior to the arrival of a surge level including concurrent wave action that will exceed el 14.0 feet MSL.

Response:

Water-tight doors will normally be left open. When notified by CONVEX of an impending storm, high winds and/or high water levels, doors will be closed by an operating procedure. (i.e., administratively controlled.)

NRC Letter: May 3, 1983

Question No. Q241.2 (Section 2.5.4.5.1 and SRP Section 2.5.4)

Rock Failures

Rock failures resulting from blasting during excavation have been reported in the FSAR. Please provide additional information to identify the locations and extent of those failures. Cross-sections showing the high angle jointing should be provided.

Response:

Refer to revised FSAR Section 2.5.4.5.1 for the response to this question.

Elevation (ft)	Material	"P" Wave (fps)	"S" Wave (fps)
+14 to +2	Fill	1,363-3,060	814-1,238
+2 to -13	Alluvium	4,820-5,818	383-684
-13 to -18	Ablation Till	6,053-6,597	398-654
-18 to -30	Basal Till	7,539-7,603	1,246-2,387

The following conservatively estimated elastic constants were used for investigating dynamic response of structures, based on "P" wave and "S" wave velocity measurements from "explosive" and "impact" sources:

Material	Young's Modulus, E (psi)	Shear Modulus, G (psi)	Poisson's Ratio
Rock	4×10^6	1.5×10^6	0.33
Basal Till	4×10^5	1.4×10^5	0.44
Ablation Till	2.7×10^4	9.0×10^3	0.49

2.5.4.5 Excavations and Backfill

The extent of excavations and backfill for major Seismic Category I structures is shown on Figure 2.5.4-40. Final grading, which includes dredging and backfilling in the vicinity of the circulating and service water pumphouse, is shown on Figure 2.5.4-41. Profiles delineating the extent of the excavation and backfill are shown on Figures 2.5.4-33 through 2.5.4-35. Geologic mapping of the excavated surfaces is described in Section 2.5.4.1.1.

2.5.4.5.1 Excavation

The founding materials for major plant structures are listed in Table 2.5.4-14. Most of the major safety related structures are founded on bedrock, with the exception of the control building, emergency diesel generator building, and the hydrogen recombiner building. The control building is founded on 1 to 4 feet of compacted structural backfill overlying basal till of thickness varying between 1 foot on the east side and 15 feet on the west. The emergency generator enclosure building wall footings are founded on basal till. The diesel generator pads are supported in approximately 8 feet of structural backfill
241.3 overlying basal till as shown on Figure 2.5.4-35 (Section J-J). The hydrogen recombiner is founded on concrete fill overlying bedrock.

Most of the circulating water discharge tunnel is founded on bedrock. Near the ventilation stack, for a distance of approximately 500 feet, the discharge tunnel is founded on crushed stone and concrete fill overlying basal till. Section 2.5.4.8.4 and Figure 2.5.4-51 (Geologic
241.1 Profile H-H") describe the founding conditions of the discharge tunnel in this area.

The service water intake lines are founded on bedrock in the main plant area; however, between the main plant area and the pumphouse they are founded on soil. When soil was encountered as a founding material all

unsuitable overburden was removed to sound basal till. Where the invert elevation was higher than the excavated grade, compacted structural backfill was placed in thin lifts to the subgrade elevation of the pipe encasement. All compacted structural backfill was placed in accordance with procedures described in Section 2.5.4.5.2. Figure 2.5.4-52 (Geologic Profile I-I"") shows the extent of structural backfill placed beneath the service water intake lines between the turbine building and the circulating and service water pumphouse. 241.1

The locations of field density tests of structural backfill placed beneath the service water intake lines near the pumphouse, where the deposit of beach and outwash sand was removed above the basal till, are presented in Figure 2.5.4-53. Table 2.5.4-19 summarizes the results of the density tests in this area.

Rock in the containment area was blasted and excavated in segmented areas, each approximately 10 feet deep. Rock bolts, discussed in Section 2.5.4.12, were installed in the southwest sector of the excavation to prevent potential sliding failures along the foliation. In addition, intercept drains were installed into the southwest excavation face to reduce the hydrostatic pressure on the foliation and joint planes. No rock slides were noted during the time the excavation was in service. However, some areas were overbroken due to blasting and to subsequent scaling operations to remove loosened rock wedges. The overbreak areas were localized and generally limited in size to approximately 2 cubic yards and less. The surfaces of the wedges generally conformed to the predominant joint sets mapped at the site and discussed in Section 2.5.4.1. The nature and extent of overbreak experienced during site excavation is considered normal for bedrock of this type and does not indicate instability in the rock mass. 241.2

Various techniques were utilized when blasting near the perimeter of structures to limit overbreak and minimize damage to adjacent rock. The methods used include line drilling, cushion blasting, presplitting, and smooth wall blasting. The purpose of each of these techniques was to develop a shear plane along the perimeter of the excavation so that the excavated rock breaks cleanly from the face. In line drilling, the perimeter holes were closely spaced and left unloaded during the blast. Cushion blasting was used to blast a narrow berm left from a previous blast. A single row of closely spaced holes was drilled along the berm, lightly loaded, and fired simultaneously. Presplitting consisted of the firing of a single row of lightly loaded, closely spaced holes, prior to the primary blast. The purpose was to produce a crack along the line of presplit holes which the subsequent primary blast could break. Smooth wall blasting is similar to cushion blasting except that the lightly loaded perimeter holes were the last delay in the blast.

Controlled blasting techniques were used to limit the vibrations felt at Millstone 1 and 2 and to preclude any structural damage to concrete or bedrock near the blast. Peak particle velocity was measured for each blast, using Sprengnether 3 - component seismographs. No damage to any structure or component in the two operating units or the Millstone 3 construction site was observed as a result of the blasting.

The inflow of water into the excavation was controlled by means of pumping from local sumps. This was possible due to the low permeability of the soils and the tightness of the joints in the bedrock. Concrete working mats were poured on all foundation surfaces upon excavating each area in order to minimize the impact of construction activities on the undisturbed founding surfaces.

Some softening of the basal till in sections of the excavation was observed. The softening is attributable to the exposure of the till to the affects of weathering and construction traffic. When this condition was encountered, the softened material was hand-excavated to firm, dry till and replaced with either fill concrete or compacted structural backfill. In the control building excavation, softened till approximately 1 foot in thickness was hand-excavated to firm till and replaced with structural backfill. The extent of the softening was verified by excavating two test trenches into the till to a depth of 4 feet. No additional softened till was encountered below the softened surface layer. The groundwater level was maintained below the subgrade by pumping from sumps outside the structure, and no seepage infiltrated the excavation after removal of the softened till and placement of the structural backfill.

2.5.4.5.2 Backfill

Category I structures founded totally or partially on structural backfill include the control building and emergency generator enclosure
241.3 building. In addition, sections of the service water line and some of the buried electrical ducts are founded on Category I structural backfill.

Material used for Category I structural backfill is predominantly obtained from glacial outwash deposits located at the Romanella Pit in North Stonington, Connecticut. Test data on borrow material from the Romanella Pit have been previously reported in July and November 1974 and are included in Appendix 2.5M. A small percentage is obtained from other borrow sources having similar geologic characteristics. A description of the borrow material from three alternate sources located in the towns of North Stonington, Preston, and Canterbury is included in a report submitted in June 1976 and is included herein as Appendix 2.5M.

All structural backfill is processed at the borrow pit by means of passing the soil through a screen, ensuring that the maximum particle size and gradation meet the backfill specification requirements. For Category I structural fill, the gradation limits are:

U.S. Standard Sieve Size	Cumulative Percent Passing
3 inches	100
3/4 inch	75 to 100
3/8 inch	65 to 90
No. 10	40 to 60
No. 40	15 to 35
No. 100	0 to 20
No. 200	0 to 15

Coefficient of Uniformity, $Cu = D_{60}/D_{10} \geq 10$

All structural backfill was compacted to 95 percent of the maximum dry density determined from the Modified Proctor Test, ASTM D1557, Method D. Moisture content was maintained within 4 percent of optimum. Structural backfill for Category I structures was placed in loose lifts not exceeding 8 inches and uniformly compacted by heavy vibratory rollers.

A continuing program of testing, inspection, and documentation was in effect during construction to ensure satisfactory placement of backfill. Category I structural backfill was tested every 500 cubic yards for conformance to the specified gradation limits prior to being allowed into the construction area. In addition, the maximum density was determined by ASTM D1557, Method D, for every 500 cubic yards of fill placed. Field density tests, using ASTM D1556, were performed for each lift of fill, but not less than one test for every 500 cubic yards of fill placed.

Locations of field density tests under the emergency generator enclosure and control building are shown in Figure 2.5.4-54, and the test results are summarized in Table 2.5.4-20. Cross-sections showing generalized subsurface profiles beneath these two structures are presented in Figures 2.5.4-55 (Section J-J) and 2.5.4-56 (Section K-K). 241.3

Shear strength of compacted backfill materials was determined from drained direct shear tests on samples compacted to 95 percent of maximum dry (ATMS D1557) density. Samples tested in the direct shear box contained only the minus No. 4 portion of the sample. For consistency, the maximum density was also determined on the minus No. 4 portion of the sample. However, the maximum density of the minus 3/4-inch fraction was tested in the field, and it can be assumed that the maximum density of the minus No. 4 fraction would be less than the maximum density attainable at the site on the whole sample. Consequently, testing the minus No. 4 fraction results in values of shear strength more conservative than would be expected for the whole soil sample. A comparison of maximum densities for the minus No. 4 and minus 3/4-inch fractions for representative samples from the major borrow areas used for Category I structural backfill is presented below:

Backfill Source	γd max (-3/4") (pcf)	γd max (-#4) (pcf)	ϕ @ 95% γd max (-#4) (deg)	ϕ @ 90% γd max (-#4) (deg)
Romanella Pit (Sample "R")	136.4	129.5	41.5	-
Preston Pit	138.8	131.0	35.0	34.6
No. Stonington Pit	148.0	136.1	37.9	34.0
Canterbury Pit	140.0	131.6	39.4	34.0
Hathaway Pit (Waterford)	129.7	121.1	39.1	-
241.3 Ledyard Pit (Soneco)	132.6	122.9	41.4	-

The maximum shear modulus and Young's modulus at static strain levels were calculated for the structural fill based on the Hardin and Richart (1963) equation for round-grained sands at very low strains:

$$G_{max} = \frac{2630\ (2.17 - e)^2}{1 + e} \left(\bar{\sigma}_o\right)^{1/2} \qquad (2.5.4\text{-}1)$$

where:

G_{max} = maximum shear modulus in psi

e = void ratio

$\bar{\sigma}_o$ = effective octahedral stress in psi

Void ratio was calculated assuming full saturation and a water content equal to 12 percent, which represents the water content at full saturation for a density of 95 percent of maximum, based on the moisture-density curve for Sample "R" in Appendix 2.5M. The octahedral stress was assumed to be equal to two-thirds of the effective overburden stress for a particular depth. The maximum shear modulus at a depth of 10 feet, which corresponds to the midpoint of the backfill layer beneath the emergency generator enclosure building, is 13,400 psi. A profile of G_{max} vs effective confining pressure is plotted on Figure 2.5.4-42.

A resonant column test was performed on a sample of the structural backfill compacted to 95 percent of a maximum dry density. The values of G_{max} plotted on Figure 2.5.4-42 obtained from this test are in agreement with the Hardin and Black (1968) equation. The low strain damping ratio was calculated to be 1.4 percent.

Young's modulus for static strain levels was obtained through an iterative process where a value of vertical strain was used to obtain a reduction factor for the G value. The value of E was calculated using the equation:

$$E = 2G\,(1 + u) \qquad (2.5.4\text{-}2)$$

where:

u = Poisson's ratio

The strain level assumed was checked with the expected strain level caused by the structural loading, using the equation:

(2.5.4-3)

where: $\sum = \frac{\Delta \sigma_z}{E}$

E = Vertical strain

$\Delta \sigma_z$ = Increase in vertical stress from structure load

E = Calculated value of Young's Modulus

For the emergency diesel generator enclosure building, the calculated vertical strain was approximately 10^{-3}, and Young's modulus at a depth of 10 feet was approximately 10,000 psi. The profile of E static vs effective confining pressure is also plotted on Figure 2.5.4-42.

Backfill placed behind concrete walls is described in Section 2.5.4.10.3.

NRC Letter: May 3, 1983

Question No. Q241.6 (Section 2.5.4.7 and SRP Section 2.5.4)

Sliding Stability

You state that the service water encasement has been analyzed for sliding stability due to seismic loading. Provide the details of the analysis and identify the cross-section used in your analysis.

Response:

The analysis performed to determine the sliding stability of the service water encasement was reviewed and it was determined that sliding of a concrete encased buried pipeline is not a viable mode of failure. The encasement is buried under level ground, so that there is uniform soil depth on both sides of the structure. The dynamic driving forces proposed by Mononobe (1929), and Okabe (1926), and included in Figure 2.5.4-43 were developed for retaining walls, where the height of soil on one side of the wall is greater than on the other side. Such a condition does not occur with structures buried beneath level ground.

As discussed in Section 2.5.2.2, Flood Design Considerations, the controlling event for flooding at the Millstone 3 site is a storm surge resulting from the occurrence of the probable maximum hurricane (PMH). The flooding that would occur as a result of the PMH would be of short duration. Therefore, because of the low vertical permeability of the overburden materials at the site, the groundwater level would not be significantly changed due to infiltration from flooding. Design criteria for flood conditions are discussed in Section 3.4.

2.5.4.7 Response of Soil and Rock to Dynamic Loading

All Seismic Category I structures and associated piping are founded either on bedrock, basal till, or structural backfill. Portions of the circulating water discharge tunnel are founded on ablation till in the vicinity of the ventilation stack north of Millstone Unit 1. A listing of the founding strata for all Category I structures is included in Table 2.5.4-14. 241.4

Hard crystalline bedrock forms the basement complex of the area. The overlying dense basal till consists of a hard, compact soil which has been heavily preloaded by continental ice. Static and dynamic properties of the basal till and bedrock are discussed in Sections 2.5.4.2.5 and 2.5.4.2.6, respectively. Static and dynamic properties for the compacted structural backfill are discussed in Section 2.5.4.5.2.

The bedrock, basal till, ablation till, and structural backfill are stable materials under vibratory motion caused by the SSE. The basal till, ablation till, and structural backfill are not susceptible to liquefaction, as discussed in Section 2.5.4.8. 241.4 241.4

The soil-structure interaction analyses for Seismic Category I structures founded on soil were performed using the computer program PLAXLY-3. The nonlinear behavior of the subgrade was accounted for by use of the computer program SHAKE (LaPlante and Christian 1974) which was used to determine the strain-corrected soil properties. The subsurface material properties used in the SSI analysis are discussed in Section 2.5.4.7.1. The method of SSI analysis and the results are discussed in Section 3.7.2.4. 241.4

The response of buried piping to seismic loadings is discussed in Section 3.7.3.12.

The shorefront west of the circulating and service water pumphouse consists of a structural fill and beach and outwash and slope varying from 5H:1V to 10H:1V, protected by graded layers of armor stone. A plan showing the extent of the shoreline protection system is presented on Figure 2.5.4-41. A typical section is shown on Figure 2.5.5-1. Static and dynamic properties of the beach sands are discussed in Section 2.5.4.2.2 and documented in the reports in Appendix 2.5G. The liquefaction potential of the beach and outwash sand is discussed in Section 2.5.4.8. The stability of the shoreline slopes under static and dynamic loading is discussed in Section 2.5.5.2. 241.4 241.4

The service water intake pipes, between the circulating and service water pumphouse and the main plant area, are embedded in a rectangular concrete encasement. Soils encountered in the pipeline excavation include beach sands, unclassified stream deposits, and ablation till. These soils were removed under the pipeline to dense basal till and replaced with Category I structural backfill. The fill was placed at a 1:1 slope from the till surface to the base of the encasement and compacted to the requirements outlined in Section 2.5.4.5.2. The sides of the encasement were backfilled with nonstructural fill similar to the material used to backfill behind retaining walls and described in Section 2.5.4.10.3. The backfill was compacted to 90 percent of maximum dry density as determined by ASTM 1557, Method D.

2.5.4.7.1 Subsurface Material Properties Used in SSI Analysis

The subsurface profiles used in the soil-structure interaction analyses for the control building and the emergency generator enclosure (EGE) are idealized, horizontal profiles based on subsurface explorations conducted at the site and described in Section 2.5.4.3. The computer program SHAKE was used to determine strain corrected values of shear modulus obtained from low strain values previously determined from field testing, laboratory testing, or empirical formulae based on laboratory test data. The program iterates to obtain values of modulus that are compatible with strain levels induced in a particular soil layer by the earthquake. The strain levels normally induced by earthquakes of
241.6 magnitudes similar to the Millstone SSE are several orders of magnitude
higher than the low strain levels achieved during laboratory or field
241.4 testing, resulting in a reduction in shear modulus when these properties
are corrected for strain and input into PLAXLY-3.

For the emergency generator enclosure, the values input into SHAKE for each layer for the free field case and the strain-corrected shear modulus and damping values are listed in Table 2.5.4-21.

For the control building, the said profile input into SHAKE and used in the soil-structure interaction analysis was the section where rack was the deepest; i.e., elevation -15 feet. Shear wave velocities were used to define said stiffness. The low strain and strain-corrected soil properties for the free field case are listed in Table 2.5.4-22.

2.5.4.8 Liquefaction Potential

The foundation materials beneath some of the Seismic Category I structures consist of limited depths of dense to very dense basal tills and/or compacted select granular backfill. These materials are not susceptible to liquefaction under earthquake motions as described in the following sections.

2.5.4.8.1 Structural Backfill

Based on studies of soils where liquefaction has been observed (Seed 1968, Lee and Fitton 1969, Kishida 1969), it is concluded that the structural backfill described in Section 2.5A.4.2 in areas below the

groundwater table is not susceptible to liquefaction, as discussed below.

1. A liquefiable soil is generally a uniform sand with a uniformity coefficient of not more than 10 (Kishida 1969). The structural backfill has a uniformity coefficient ranging from 25 to 50 (Figure 2.5.4-44).

2. A soil having a relative density of more than 75 percent is not likely to liquefy (Kishida 1966, 1969; Koizumi 1966; Lee and Seed 1967; Seed and Lee 1966). Accordingly, compaction criteria of the structural backfill given in Section 2.5.4.5.2 have been designed to yield a relative density higher than 75 percent.

3. According to the envelope of "most liquefiable soils" given by Lee and Fitton (1969), which also contains the envelope given by Kishida (1969), the average particle size, D_{50}, of the "most liquefiable soils" envelope is between 0.02 and 0.7 mm, whereas the corresponding particle size of the structural backfill used is larger than 1.0 mm (Figure 2.5.4-44).

NRC Letter: May 3, 1983

Question No. Q241.11 (Section 2.5.4.8.4 and SRP Section 2.5.4)

Bedrock Profile

In the FSAR Section 2.5.4.8, you state that in the vicinity of the ventilation stack north of Millstone 1, bedrock drops sharply to a trough. Identify the location of this bedrock trough on a plot plan and provide the subsurface profiles of the trough and the overlying soils. Information pertinent to the disclosing of the bedrock trough, such as exploratory boring and/or trenching, should be identified and discussed.

Response:

Refer to revised FSAR Section 2.5.4.8.4 for the response to this question.

NRC Letter: May 3, 1983

Question No. Q241.12 (Section 2.5.4.8.4 and SRP Section 2.5.4)

Dynamic Response Analysis of Ablation Till

Identify the location where the idealized profile was obtained for the dynamic response analyses of Ablation Till and justify the groundwater level assumption.

Response:

Refer to revised FSAR Section 2.5.4.8.4.1 for the response to this question.

NRC Letter: May 3, 1983 1.8

Question No. Q241.14 (Sections 2.5.4.10 and SRP Section 2.5.4) 1.11

Rock Bearing Capacity 1.12

Provide the bases for allowing the bedrock bearing load as high as 1.13
200 KSF.

Response: 1.14

Refer to revised FSAR Section 2.5.4.10.1 for response to this 1.15
question.

NRC Letter: May 3, 1983

Question No. Q241.16 (Section 2.5.4.10.3 and SRP Section 2.5.4)

Lateral Earth Pressure

Provide the design values of the lateral earth pressures used in the design of rigid, unyielding, foundation walls.

Response:

FSAR Figure 2.5.4-43 shows the lateral pressure distribution used in designing rigid unyielding foundation walls. The figure has been revised to correct errors and to add the unit weight of backfill. Also, FSAR Section 2.5.4.10.3 has been changed to correct typographical errors.

stress history and multidirectional shaking. Based on these data, Figure 6-1 of Seed et al (1975) (included herein as Figure 2.5.4-48) presents lower bounds of the cyclic stress ratios causing liquefaction versus the standard penetration resistances of sands for magnitudes 5 to 6 and 7 to 7 1/2 earthquakes, corrected to an effective overburden pressure of 1 ton per square foot (N_1) based on the Gibbs and Holtz (1957) correlation of relative density of sands to blow count and effective stress. A plot of N_1 values vs effective stress used in this method is the SPT blow count for borings P1 through P8 and I2, I3, I8, I9, and I10 is included as Figures 2.5.4-28 and 2.5.4-29. The mean value of corrected blow count for these borings was calculated as 20.0, which corresponds to a cyclic stress ratio of 0.278 for a magnitude 5 to 6 earthquake, using Figure 2.5.4-48. When compared with the earthquake induced shear stresses obtained from the SHAKE analysis described in Section 2.5.4.7, the minimum factor of safety against liquefaction calculated by this method was 1.68 at a depth of 15 feet.

A very conservative factor of safety against liquefaction was also calculated using a cyclic stress ratio based on the mean corrected blow count less one standard deviation. An N_1 value of 13.1 was used to obtain a cyclic stress ratio of 0.185 from Figure 2.5.4-48. The minimum factor of safety calculated for the lower value of N_1 was 1.13 at a depth of 15 feet. This is considered acceptable, considering the fact that the mean value of N_1, less one standard deviation, is well below the mean value originally used by Seed et al in determining the curves in Figure 2.5.4-48. An additional conservatism in the analysis is the use of the magnitude 6.0 relationship for determining the cyclic stress ratio. The SSE at the site is based on an Intensity VI-VII earthquake, which corresponds to a magnitude of approximately 5.3, using relationships developed by Gutenberg and Richter (1942).

The factor of safety against liquefaction at various depths for each analysis is presented on Figure 2.5.4-49. It can be concluded that liquefaction will not occur in the beach and glacial outwash sands adjacent to the circulating and service water pumphouse, and that the shorefront is stable against sliding failures due to liquefaction of the sand. The stability against sliding of the shorefront during the SSE is discussed in Section 2.5.5.2. 241.9

2.5.4.8.4 Ablation Till

The circulating water discharge tunnel extends 1,700 feet from the main plant area to the Millstone quarry east of Millstone 1. For approximately 1,200 feet, the tunnel is founded on bedrock. However, in the vicinity of the ventilation stack north of Millstone 1, bedrock drops sharply to a trough. The maximum thickness of the overburden in this trough is approximately 60 feet. Borings 402 through 412 were drilled in this area to determine the subsurface conditions. A cross-section of the trough along the discharge tunnel is presented in Figure 2.5.4-51. The location of the section is shown on Figure 2.5.4-31. In this area, which extends for approximately 500 feet, the fill and alluvium overlying the ablation and basal tills were excavated and replaced with crushed stone and concrete fill to the base elevation of the discharge tunnel. Because the ablation till is a sandy material 241.11

below the groundwater table, the liquefaction potential was analyzed. The analysis described in Section 2.5.4.8.4.1 shows that liquefaction of the ablation till is not possible under the site SSE. The structural fill and basal till have been shown to be nonliquefiable in Sections 2.5.4.8.1 and 2.5.4.8.2, respectively.

2.5.4.8.4.1 Dynamic Response Analysis of Ablation Till

The dynamic response of the ablation till has been evaluated to determine earthquake induced shear stresses caused by ground motions applied at the bedrock surface and amplified through the soil profile. This evaluation was made using the computer program SHAKE, similar to the analysis in Section 2.5.4.8.3.1.

241.12 A horizontally stratified idealized soil profile was selected to model the subsurface conditions input into the SHAKE analysis for the discharge tunnel. This profile was based on soil strata encountered in boring 411, which encountered the deepest rock, and represents the most conservative profile in the study area. The generalized soil profile (Figure 2.5.4-50) used in the analysis of the tunnel consisted of 5 feet of structural fill, 13 feet of ablation till, and 22 feet of basal till. Groundwater level was established at 10 feet below the ground surface, elevation +4 feet, based upon the average groundwater levels measured in borings 407 and 411. (See Figure 2.5.4-31 for locations). The shear moduli values of the soils were obtained from cross-hole tests described in Section 2.5.4.4.3. The values of shear modulus (G) and damping (D) for low strain levels used in the SHAKE analysis for each layer are:

Layer	Elevation (ft)	Depth (ft)	Soil Type	G_{max} (ksf)	D_{max} (%)
1	+14 to -8	0-22	Discharge Tunnel	-	0.5
2	-8 to -13	22-27	Structural Fill	1.93×10^3	0.5
3	-13 to -26	27-40	Ablation Till	1.30×10^3	0.5
4	-26 to -48	40-62	Basal Till	2.0×10^4	0.5

The reduction of G_{max} with strain was performed through a series of iterations similar to the method described in Section 2.5.4.8.3.1 using the same earthquake records normalized to 0.17g.

This analysis indicated that the average maximum shear stress in the ablation till induced by the SSE, varied from 515 psf to 533 psf. The average shear stress is assumed to be 0.65 of the peak value.

2.5.4.8.4.2 Liquefaction Analysis of Ablation Till

Procedures used for liquefaction analysis of the ablation till were similar to the empirical approach described in Section 2.5.4.8.3.2.

Standard penetration resistance data (N_1 values) were related to liquefaction potential in accordance with methods developed by Seed, Arango, and Chan (1975) and DeAlba, Chan, and Seed (1975). N_1 values for the ablation till were obtained from borings taken at the discharge

tunnel location (400 series) and samples of ablation till from the main plant borings (300 series).

The following table summarizes the results of the liquefaction analysis. It compares earthquake induced shear stresses calculated from SHAKE with shear strength values determined from average corrected blow count values and average N_1 values less one standard deviation ($N_1 - \sigma$):

Midpoint of Layer Elevation (ft)	Induced Shear Stress (psf)	Mean N_1	Shear Strength (psf)	F.S.	$N_1-\sigma$	Shear Strength (psf)	F.S.
-15.2	515	28.7	1,079	2.10	15.5	578	1.12
-19.5	531	28.7	1,218	2.29	15.5	652	1.23
-23.9	533	28.7	1,357	2.60	15.5	726	1.36

It can be concluded, therefore, that the ablation till under the discharge tunnel is not susceptible to liquefaction, even considering the ultraconservative case of the shear strength calculated from the mean corrected blow count less one standard deviation.

2.5.4.9 Earthquake Design Basis

A safe shutdown earthquake of 0.17g and a 1/2 SSE value of 0.09g in the horizontal direction and two-thirds of these values in the vertical direction, input at the bedrock surface, have been used as the design bases for seismic loading at the site. The derivation of these values is described in Sections 2.5.2.6 and 2.5.2.7.

For structures founded on soils, amplification effects have been considered by means of a soil-structure interaction analysis using the computer program PLAXLY-3 described in detail in Section 3.7.2.4.

For the liquefaction analysis of the beach sands adjacent to the circulating and service water pumphouse, the SSE value of 0.17g was input at the bedrock surface, and the average amplified ground motion at the surface determined from the SHAKE program using three earthquake records and described in Section 2.5.4.8.3.1 was calculated to be 0.27g. Consequently, a value of 0.25g was conservatively used for the entire

soil column as the average seismic loading of shoreline slopes in the stability analysis described in Section 2.5.5.2.

2.5.4.10 Static Stability

2.5.4.10.1 Bearing Capacity

Table 2.5.4-14 summarizes the bearing pressures for mats or individual spread footings founded on various foundation materials. 241.13

The selection of the bearing capacity values used in footing design were based on the bearing capacity formulae (Terzaghi and Peck 1967, Vesic 1975) for an estimated angle of internal friction for basal till equal to 40 degrees and for structural backfill equal to 34 degrees. The total unit weight for the till was assumed to be equal to 145 pcf and for the structural backfill, a total unit weight equal to 140 pcf was used. Test values reported in Section 2.5.4.5.2 showed a range of ϕ angles varying from 35 to 41.5 degrees for the structural fill compacted to 95 percent of the maximum modified Proctor density. Inputting the relevant soil parameters described above, and taking into account the effect of the groundwater table, the bearing capacity formula for square footings or mats on basal till reduces to:

$$q_{all} = 1.9\ D + 1.1\ B$$

$$q_{all}\ (max) = 12\ ksf$$

For structural backfill:

$$q_{all} = 0.9\ D + 0.4\ B$$

$$q_{all}\ (max) = 8\ ksf$$

where:

q_{all} = Allowable bearing capacity in ksf with a minimum safety factor = 3

D = Depth of embedment (feet)

B = Width of footing (feet)

Table 2.5.4-23, Bearing Capacity of Major Structures, presents a summary of the allowable bearing capacity for the material beneath each structure. In all cases, the factor of safety is greater than 3, which is the minimum required value. 241.13

Based on Teng (1962), the design bearing capacity of foundations on rock is commonly taken as 1/5 to 1/8 of the crushing strength (factor of safety of 5 to 8). A value of 200 ksf was selected for the maximum allowable bearing capacity of bedrock at the site. This corresponds to approximately 1/7 of the average unconfined compressive strength of approximately 1,440 ksf (10,000 psi) reported in Table 2.5.4-10. The 200 ksf value also corresponds to the presumptive surface bearing value 241.11

given by the Connecticut Basic Building Code (1978) for massive crystalline rock, including granite and gneiss.

241.14 From Table 2.5.4-14, the maximum average foundation pressure for a structure on rock is 8 ksf. Thus, the factor of safety against a bearing capacity failure is much greater than 3 for all structures founded on rock.

2.5.4.10.2 Settlement of Structures

Rock and soil supported Seismic Category I structures will experience only elastic displacements under the design loads. Analyses using linear elasticity principals, assuming rigid foundations, indicate that the vertical settlements of structures founded on rock are very small under the design loads, as shown by the summary included in Table 2.5.4-14.

The settlement of structures embedded in rock, such as the containment, was calculated using elastic solutions for circular rigid mats on a semi-infinite mass by Butterfield and Banjerjee (1971). The main steam valve, auxiliary, and engineered safety features buildings, founded on rock, were analyzed using equations for rigid rectangular mats on a semi-infinite mass developed by Whitman and Richart (1967). Structures founded totally or partly on soil, such as the control, emergency diesel generator enclosure, fuel, and waste disposal buildings, were analyzed using solutions obtained by Sovinc (1969) for rigid rectangles on a finite layer. The settlement of the underlying rock layer was also
241.15 estimated using the Whitman and Richart equations.

Elastic properties of the rock and basal till are discussed in Section 2.5.4.4.3. The elastic modulus (E) for static strain levels was estimated equal to 10,000 psi, as discussed in Section 2.5.4.5.2.

241.15 Table 2.5.4-14 indicates that the maximum estimated settlement within any one structure occurs at the emergency generator enclosure building and is equal to 0.22 inch. Even though the fuel building is founded on three different materials (rock, basal till, and compacted fill), it undergoes a lesser differential settlement of 0.01 inch as indicated in
241.15 Table 2.5.4-14. Maximum estimated differential settlement between adjacent structures occurs between the control building and the emergency generator enclosure building, and is equal to about 0.20 inch. The rate of these settlements would essentially be the same as the rate of loading because of elastic nature of the bearing material.

2.5.4.10.3 Lateral Earth Pressures

The magnitude and distribution of lateral earth pressures is a function of the allowable yielding of the wall, the backfill material characteristics, water pressure, surcharge loads from adjacent structures, and, for seismically designed structures, the earthquake loading. The concrete foundation walls were conservatively assumed to be rigid, unyielding walls. Therefore, the coefficient of earth pressure at rest, K_o, has been used in evaluating lateral loads on these
241.16 walls. For the backfill at the site, a value of K_o = . _ was used.

Backfill placed behind walls consisted of well graded sands and gravels compacted to 90 percent of maximum density (ASTM D1557) to minimize the horizontal loads induced by high compactive stresses. Tests on similar soils, compacted to 90 percent of maximum dry density and reported in Section 2.5.2.5.2, resulted in friction angles in excess of 34 degrees.

Dynamic loadings include pressures due to the soil mass, water, and surcharge, accelerated in the vertical and horizontal directions. Methods of analysis are based on procedures proposed by Mononobe (1929), Okabe (1926), and Seed and Whitman (1970) and are graphically depicted on Figure 2.5.4-43.

a corrugated PVC casing and fully grouted for double corrosion protection. Each anchor was proof loaded to 150 kips and then the load was reduced to 125 kips for 24 hours. The anchor was subsequently locked off at a permanent load of 25 kips and encased in the concrete foundation mat.

Rock anchors were installed in the service building to provide resistance to uplift loads due to buoyant forces and seismic forces. These anchors consisted of 1 3/8-in diameter, high strength steel bar sheathed in a corrugated PVC casing and fully grouted for double corrosion protection. The ultimate strength of each anchor is 237k, with a working load equal to 60 percent of the ultimate strength, or 142k. Each anchor was tensioned to a test load of 168k and held at 150k for a 24 hour period. Two anchors were proof tested to 190k. The anchors were locked off at a load of 40k, which corresponds to the hydrostatic uplift component of the anchor design load. The remaining capacity of the anchor is mobilized during seismic loading.

Temporary rock bolts were installed in the southwest sector of the containment excavation face to prevent potential sliding failures along the foliation planes. These bolts consisted of Grade 60 steel, No. 11 reinforcing bars with a working load of 45 kips. Anchorage of the rock bolts was provided by Celtite polyester resin encapsulation.

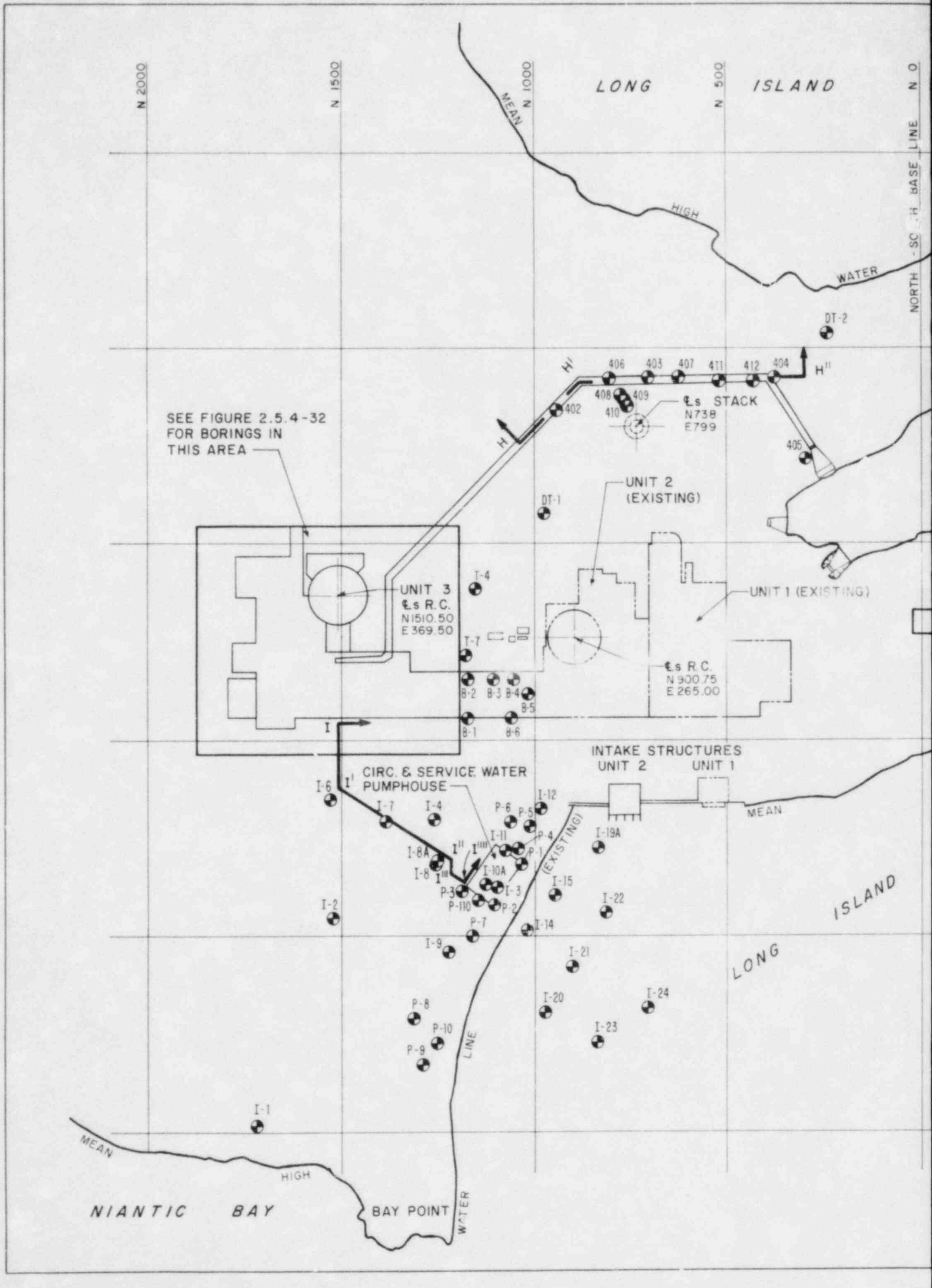
Detailed geologic mapping of bedrock surfaces at the site, described in detail in Section 2.5.4.1.1, identified certain preferred joint surfaces that may cause potential sliding planes with the containment excavation face. As a result of these findings, a reinforced concrete ring beam was placed in the annular space between the excavation face and the containment exterior wall to stabilize the wedges. The slope stability analysis for the containment excavation is discussed in detail in Section 2.5.5.1. The structural analysis is discussed in Chapter 3.

2.5.4.13 Subsurface Instrumentation

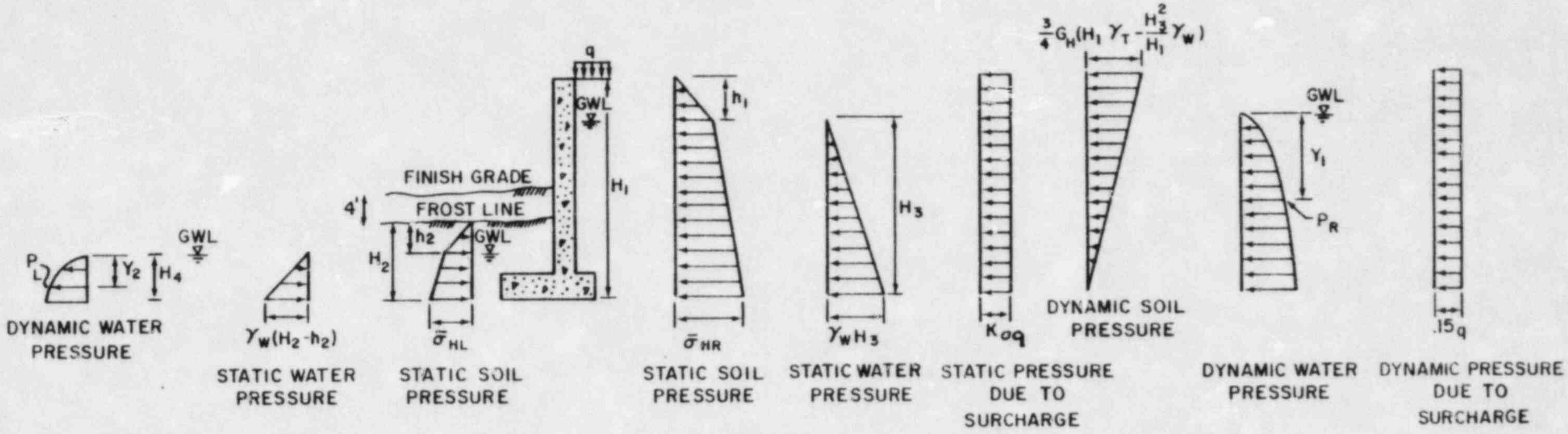
Most of the Category I structures at the site are founded on sound bedrock. Predicted settlements listed in Table 2.5.4-14 for these structures are very small. A plan of the location of the settlement monitoring benchmark locations is shown on Figure 2.5.4-59. Settlement
241.15 predictions for structures founded on basal till or structural backfill
indicate that the maximum expected settlement will be less than 1/4 inch and that this settlement will occur over a relatively short period of time due to the elastic nature of the subsurface materials. Settlement has been monitored for the control, fuel, waste disposal, and emergency generator enclosure buildings during construction. Plots of observed
241.15 settlement versus time for these structures are presented in
Figures 2.5.4-60 through 2.5.4-64. The records show no significant movement of any structure, although some heave has occurred due to rebound from excavation. Settlement of these structures will be periodically measured for a short period of time after construction until the rate of change of structure movement decreases.

2.5.4.14 Construction Notes

No significant problems were encountered during construction that required extensive redesign of structures. A small amount of basal till



N 2000
N 1500
N 1000
N 500
N 0
LONG ISLAND
MEAN HIGH WATER
NORTH - SO H BASE LINE
DT-2
406
403
407
411
412
404
H'
H''
H
408
409
410
402
℄s STACK
N738
E799
405
SEE FIGURE 2.5.4-32
FOR BORINGS IN
THIS AREA
UNIT 2
(EXISTING)
DT-1
I-4
UNIT 3
℄s R.C.
N1510.50
E369.50
UNIT 1 (EXISTING)
T-7
℄s R.C.
N 900.75
E 265.00
B-2
B-3
B-4
B-5
B-1
B-6
I
INTAKE STRUCTURES
UNIT 2
UNIT 1
CIRC. & SERVICE WATER
PUMPHOUSE
I-6
I'
I-7
I-4
P-6
P-5
I-12
MEAN
(EXISTING)
I-19A
I-11
P-4
P-1
I-8A
I''
I''''
I-8
I'''
I-10A
P-3
I-3
I-15
P-110
P-2
I-22
I-2
ISLAND
P-7
I-14
I-9
I-21
LONG
I-24
I-20
P-8
P-10
I-23
P-9
LINE
I-1
MEAN
HIGH
WATER
NIANTIC BAY
BAY POINT



LEGEND

$\bar{\sigma}_{HL} = [\gamma_T h_2 + \gamma_{SUB}(H_2 - h_2)] K_o$

$\bar{\sigma}_{HR} = [\gamma_T h_1 + \gamma_{SUB}(H_1 - h_1)] K_o$

$P_R = 7/10[7/8 G_H \gamma_W \sqrt{H_3 Y_1}]$

$P_L = 7/10[7/8 G_H \gamma_W \sqrt{H_4 Y_2}]$

γ_T = TOTAL UNIT WEIGHT OF SOIL

γ_{SUB} = SUBMERGED UNIT WEIGHT OF SOIL

γ_W = UNIT WEIGHT OF WATER

K_o = COEFFICIENT OF LATERAL EARTH PRESSURE

q = UNIFORM SURCHARGE LOAD

G_H = HORIZONTAL EARTHQUAKE COEFFICIENT

G_V = VERTICAL EARTHQUAKE COEFFICIENT

GWL = GROUND WATER LEVEL

P_R = DYNAMIC WATER PRESSURE AT ANY DEPTH, H_1

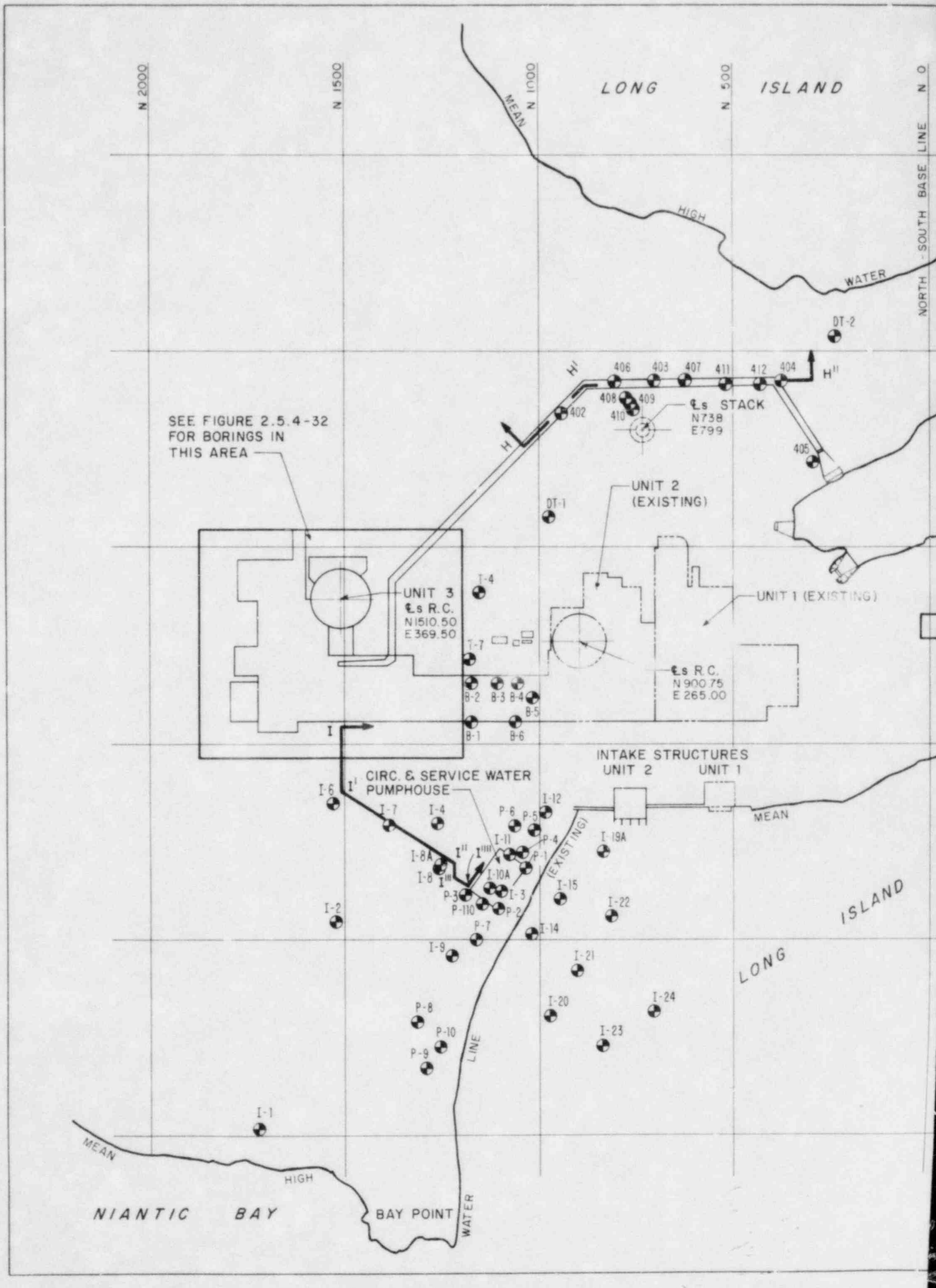
NOTES

1. DYNAMIC WATER PRESSURE ON LEFT HAND SIDE IS NEGATIVE.
2. CURVE FOR DYNAMIC WATER PRESSURE IS PARABOLA.
3. CRANE IS ASSUMED TO BE A POINT LOAD.
4. $K_o = 0.5$ $\gamma_T = 143$ pcf
   $G_H = 0.17$ $\gamma_{SUB} = 80.6$ pcf
   $G_V = 0.114$ $\gamma_{dry} = 132$ pcf
   $\gamma_W = 62.4$ pcf

FIGURE 2.5.4-43

LATERAL PRESSURE DISTRIBUTION

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT


N 2000
N 1500
N 1000
N 500
N 0
NORTH - SOUTH BASE LINE
LONG ISLAND
MEAN HIGH WATER
SEE FIGURE 2.5.4-32 FOR BORINGS IN THIS AREA
DT-2
406
403
407
411
412
404
408
409
410
402
℄s STACK N738 E799
405
H
H'
H''
DT-1
UNIT 2 (EXISTING)
UNIT 3 ℄s R.C. N1510.50 E369.50
UNIT 1 (EXISTING)
℄s R.C. N900.75 E265.00
I-4
T-7
B-1
B-2
B-3
B-4
B-5
B-6
I
I'
I''
I'''
I''''
INTAKE STRUCTURES UNIT 2 UNIT 1
CIRC. & SERVICE WATER PUMPHOUSE
(EXISTING)
I-6
I-7
I-12
P-6
P-5
P-4
P-1
I-11
I-19A
I-8A
I-8
I-10A
P-3
I-3
P-110
P-2
I-15
I-22
I-2
P-7
I-14
I-9
I-21
I-20
I-24
I-23
P-8
P-10
P-9
I-1
NIANTIC BAY
BAY POINT

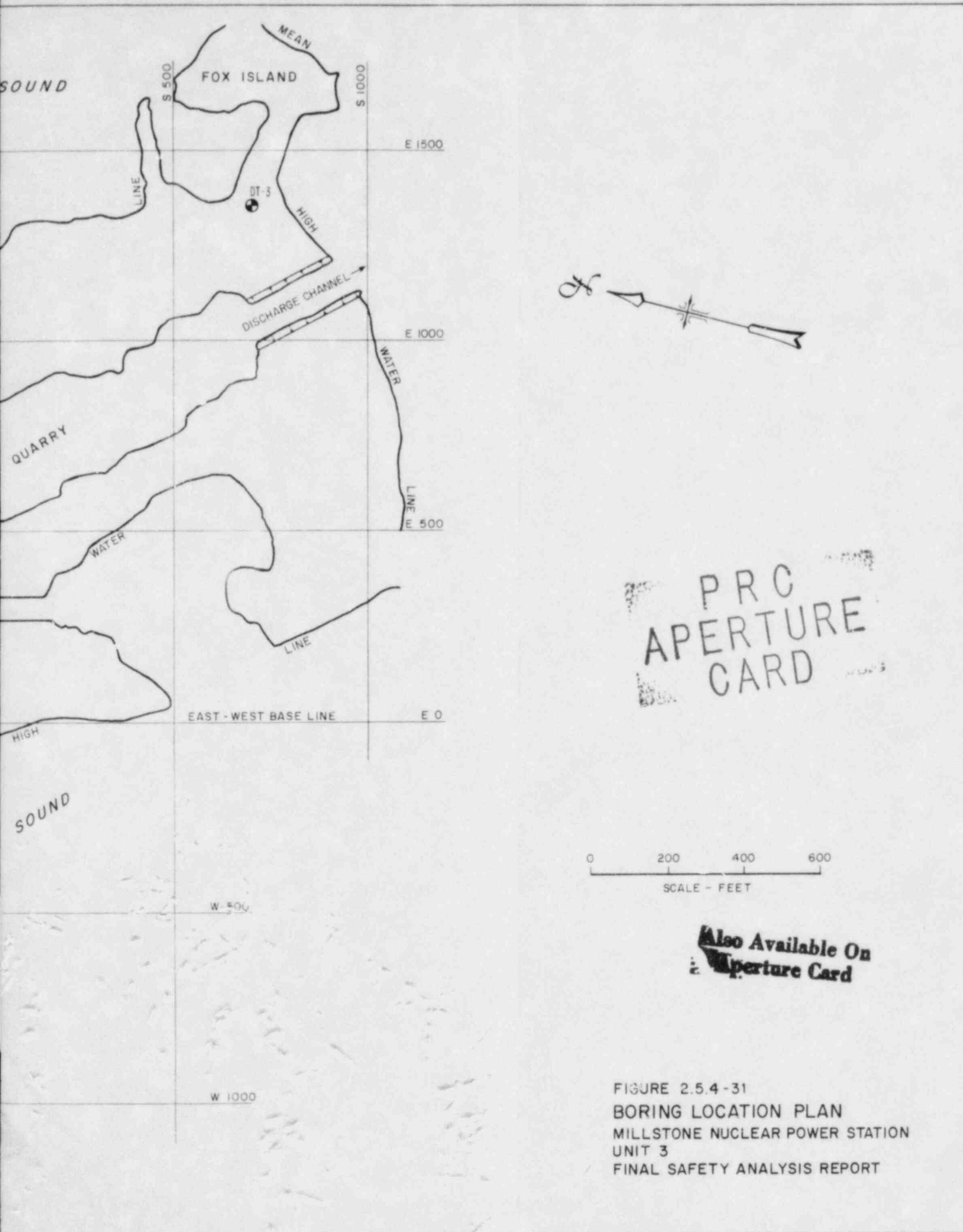
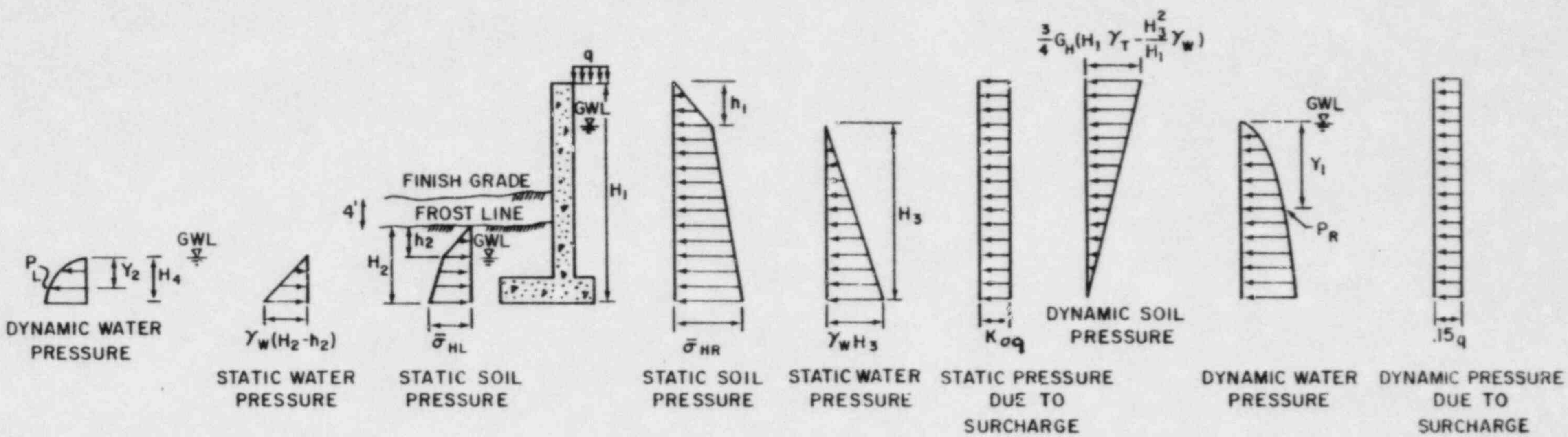


FIGURE 2.5.4-31
BORING LOCATION PLAN
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT



LEGEND

$\bar{\sigma}_{HL} = [\gamma_T h_2 + \gamma_{SUB}(H_2 - h_2)] K_o$

$\bar{\sigma}_{HR} = [\gamma_T h_1 + \gamma_{SUB}(H_1 - h_1)] K_o$

$P_R = 7/10 [7/8 G_H \gamma_W \sqrt{H_3 Y_1}]$

$P_L = 7/10 [7/8 G_H \gamma_W \sqrt{H_4 Y_2}]$

γ_T = TOTAL UNIT WEIGHT OF SOIL

γ_{SUB} = SUBMERGED UNIT WEIGHT OF SOIL

γ_W = UNIT WEIGHT OF WATER

K_o = COEFFICIENT OF LATERAL EARTH PRESSURE

q = UNIFORM SURCHARGE LOAD

G_H = HORIZONTAL EARTHQUAKE COEFFICIENT

G_V = VERTICAL EARTHQUAKE COEFFICIENT

GWL = GROUND WATER LEVEL

P_R = DYNAMIC WATER PRESSURE AT ANY DEPTH, H_1

NOTES

1. DYNAMIC WATER PRESSURE ON LEFT HAND SIDE IS NEGATIVE.
2. CURVE FOR DYNAMIC WATER PRESSURE IS PARABOLA.
3. CRANE IS ASSUMED TO BE A POINT LOAD.
4. $K_o = 0.5$ $\gamma_T = 143$ pcf
   $G_H = 0.17$ $\gamma_{SUB} = 80.6$ pcf
   $G_V = 0.114$ $\gamma_{dry} = 132$ pcf
   $\gamma_W = 62.4$ pcf

FIGURE 2.5.4-43

LATERAL PRESSURE DISTRIBUTION

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

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NRC Letter: May 3, 1983

Question No. Q280.2

The fire protection program will be reviewed to the guidelines of BTP CMEB 9.5-1 (NUREG-0800), July 1981. Provide a comparison that describes conformance of the plant fire protection program to these guidelines. Deviations from the guidelines should be specifically identified. A technical basis should be provided for each deviation.

Response:

Fire Protection Evaluation Report, Appendix B, identifies areas in which the Millstone 3 design does not fully conform to BTP CMEB 9.5-1 (NUREG-0800) dated July 1981. Also refer to FSAR Table 1.9-2 under SRP BTP CMEB 9.5-1 for deviations to BTP CMEB 9.5-1 and the justification.

(approaching 1 mile or more), separate yard fire main loops should be used."

RESPONSE

The fire-water supply basically consists of two 245,000-gallon fire-water tanks. Three fire-water pumps are interconnected (piped and valved) such that each pump can take a suction from either or both tanks.

Fire-water supply tanks are valved to ensure that a leak in one tank or the associated piping would not cause both tanks to drain. Water supply to the fire-water tanks is through a 12-inch city water main that can refill a tank in 8 hours. As a backup source to the city water main, well water is available through a reversible elbow connection (normally left disconnected). Fire-water tanks are strictly piped for fire-protection water storage and are independent of sanitary-or service-water storage.

The largest existing single demand for fire water for the entire system is expected to be from the Millstone 3 main transformer deluge system, which requires approximately 1,500 gallons per minute. It is calculated that, with this expected flow and the 500 gallons per minute required for two hose streams, sufficient water storage and flow capacity is available with one fire-water pump running. 280.2

POSITION

(c) "If pumps are required to meet system pressure or flow requirements, a sufficient number of pumps should be provided so that 100 percent capacity will be available with one pump inactive (e.g., three 50 percent pumps or two 100 percent pumps). The connection to the yard fire main loop from each fire pump should be widely separated, preferably located on opposite sides of the plant. Each pump should have its own driver with independent power supplies and control. At least one pump (if not powered from the emergency diesels) should be driven by nonelectrical means, preferably diesel engine. Pumps and drivers should be located in rooms separated from the remaining pumps and equipment by a minimum three-hour fire wall. Alarms indicating pump running, driver availability, or failure to start should be provided in the control room.

"Details of the fire pump installation should as a minimum conform to NFPA 20, 'Standard for the Installation of Centrifugal Fire Pumps.'"

RESPONSE

The three centrifugal fire-water pumps are 100-percent capacity units rated for 2,000 gallons per minute. Two of the pumps are electric-motor-driven horizontal units, and the third is a diesel-engine-driven horizontal unit. All three pumps feed the underground main

separately but feed lines leaving the pumphouses are adjacent to each other.

The Millstone 1 fire-water pumphouse contains the electric fire pump (M7-8), the Millstone-1 diesel fire pump (M7-7), and the 50 gallons per minute electric jockey pump (M7-11). The Millstone-2 fire-water pumphouse is a separate structure located adjacent to the Millstone 1 pumphouse and contains the Millstone 2 electric pump (MP-82). These pumphouses are adjacent but are completely independent of each other and do not share a common barrier.

System operation is such that the electric jockey pump maintains system pressure by automatically starting and stopping as required. The other pumps also start automatically. When system pressure drops to 95 psig, the Millstone 2 electric pump will start. If pressure reaches 85 psig, the Millstone 1 electric pump will start, and if pressure continues to drop, the Millstone 1 diesel-driven pump will start at 75 psig.

Millstone 1 Fire Pumps

The electric-motor-driven fire pump M7-8 is supplied from the fire-water pumphouse feeder motor control center MCC-22A-2. This feeder is supplied by the normal 480-volt supply bus 1 and the standby 480-volt supply bus 1A through an automatic transfer switch. The power cables from buses 1 and 1A are installed in separate trays. The control cables for the bus 1 breaker are installed in the S1 trays to control room panel 908, and the bus 1A breaker control cables are installed in the S2 trays to control room panel 908.

An automatic transfer switch transfers the power supply of motor control center MCC-22A-2 from normal to standby on loss of normal power. Loss of normal power also will cause the electric fire pump M7-8 or the jockey fire pump M7-11 to trip while the bus is transferring. Should the line water pressure drop to 75 psig or less, the diesel-driven fire pump M7-7 will autostart. On restoration of power to MCC-22A-2, the electric fire pump M7-8 will restart if the system pressure is below 85 psig. Both pumps will then run until manually stopped.

Each of the electric and diesel fire pumps is controlled by a separate control panel on opposite sides of the room and supplied from separate control power sources. Both pumps have local-auto and remote-manual start features.

Within the Millstone-1 fire pumphouse, there is no separation of control or annunciator cables. The pumphouse and all enclosed equipment are grounded to the plant perimeter grid.

Power availability and operation of the two Millstone 1 fire-water pumps are monitored in the control room by alarms that annunciate when fire-water pumps start or when the power supply to the starters is interrupted.

NRC Letter: May 3, 1983

Question No. Q280.3

Provide the qualifications of the fire protection engineer responsible for the formulation and implementation of the fire protection program. (BTP CMEB 9.5-1 Section C.1.b.)

Response:

Refer to revised Fire Protection Evaluation Report, Section 3.2, for the response to this question.

SECTION 3

ADMINISTRATION

3.1 FIRE PROTECTION PROGRAM

A fire protection program has been established at the Millstone 3 Nuclear Power Plant. This program establishes the fire protection policy for the protection of structures, systems, and components important to the safety of the plant and the procedures, equipment, and personnel required to implement the program.

The fire protection program is under the direction of an individual who has been delegated authority commensurate with the responsibilities of the position, and who has available staff personnel with knowledge in both fire protection and nuclear safety

The fire protection program extends the concepts of defense-in-depth to fire protection in areas important to safety, with the following objectives:

to prevent fires from starting;

to detect rapidly and control and extinguish promptly those fires that do occur; and

to provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant

3.2 FIRE PROTECTION ORGANIZATION

The overall responsibility for the Fire Protection Evaluation at Millstone 3 rests with the Northeast Utilities Service Company's (NUSCo.) Executive Vice President of Engineering and Operations. Responsibility for developing, maintaining, and implementing the Fire Protection Program is delegated to the respective Vice Presidents of Generation Engineering and Construction and Nuclear Operations. The organizational responsibilities are shown in FSAR Figures 13.1-1 through 13.1-5.

The responsibility for developing the Fire Protection Program is assigned to the Generation Engineering and Construction Division under the specific cognizance of the Fire Protection Group. This Fire Protection Group is also responsible for conducting fire hazard analyses and provide engineering/design for fire protection systems.

NUSCo. provides a fire protection engineering staff which is responsible for supporting Millstone Unit No. 3. The supervisor of Generation Fire Protection Engineering has qualifications which exceed the requirements of BTP-CMEB 9.5-1, Section C.1.5.a. 280.3

The responsibility for the performance of audits to ensure compliance with the Fire Protection Evaluation rests with the NUSCo. Quality Assurance Department in accordance with existing divisional jurisdictions. Existing procedures will be used and, if necessary, augmented to ensure the performance of audits and the documentation and reporting of deficiencies and results of corrective action.

The responsibility for plant-related fire protection activities rests with the Nuclear Operations Department. The Station Superintendent of Millstone will provide the overall guidance and coordination for Nuclear Operations activities.

The responsibility for the development of specific fire-fighting procedures, inspection programs, maintenance, training, and drills rests with the Superintendent, Millstone 3. He has the overall responsibility for the Fire Protection Evaluation of the operating units at all times, including those periods when construction forces are working on operating or new structures.

Review of loss and property-damage analyses and dealings with insurance companies will continue to be the responsibility of the Assistant Secretary and Claims and Insurance Manager. The existing lines of communication will continue.

3.3 FIRE BRIGADE AND TRAINING

The Millstone Power Station Fire Brigade consists of the shift personnel within the Operations Department. At any one time, each shift contains a minimum of five fire crew members who are available to fight a fire. If additional plant personnel are required, this fire crew can be supplemented by fire crew members from the adjacent unit. The organization and number of people available during off-normal hours meet the guidelines of the draft ANSI Standard N18.10.

280.5 As a minimum, each fire brigade per shift will consists of five members, three of whom will be Operations personnel. They will be both knowledgeable in plant safety related systems operations and fire fighting techniques. The fire brigade will be fully established and functional prior to fuel load date. Although the five trained members of the shift crew will be able to effectively fight and control all postulated plant fires, supplemental assistance is available from two sources. The station fire crew is made up of all the shift personnel from all operating units. A list of the station fire crew members, with each person's name, title, unit, and telephone number, is maintained in the control room. In the event of fire, when backup fire crew response is required, the personnel on this list will be requested to respond. Although the backup fire crew is not kept ready to respond on an on-call status, the ability to recruit an adequate number of personnel to respond to a fire is good since this fire crew consists of about 50 persons.

Plant procedures regarding fires also state that, if conditions warrant, the local public fire departments shall be called. Within a 5 mile radius of the plant there are numerous local volunteer fire

companies. Letters of commitment to supply public fire department assistance have been obtained from these fire companies.

An inclusive fire-fighting training program is under development. Millstone Station Fire Fighting Training Program establishes the requirements of, and responsibilities for, the training of fire-fighting personnel. This program will be responsive to the requirements of BTP APCSB 9.5-1, Position B, and its development will be guided by the appropriate codes and standards referenced therein.

Since the local fire departments are included in the overall Fire Protection Evaluation, they will be included in the training program. Preliminary discussions held with them covered areas of access, equipment compatibility, and on-site direction. There will be an ongoing training program to instruct the local fire departments in subjects pertinent to the plant (e.g., radiation protection, plant layout, etc) to enhance their effectiveness. Periodic fire drills will be conducted with the participation of the local fire departments in order to evaluate the effectiveness of the training program.

3.4 QUALITY ASSURANCE

It is the intent of the NNECo to include in the Quality Assurance Program those areas of the Fire Protection Evaluation that are identified in BTP APCSB 9.5-1, Position C.

The Quality Assurance Program has been applied to the fire protection systems, components, and programs providing fire detection and suppression capabilities to those areas of the plant that are important to safety.

Although not included in the Quality Assurance Program, portions of the Fire Protection Evaluation have had quality assurance requirements applied.

NRC Letter: May 3, 1983

Question No. Q280.4

Describe administrative controls that will be developed and implemented to comply with BTP CMEB 9.5-1 Section C.2.

Response:

Administrative Control Procedures exist for Millstone Nuclear Power Station Units 1 and 2. These Administrative Control Procedures will be extended to provide coverage to Millstone Unit No. 3 Operations, prior to fuel load date.

The existing Administrative Control Procedures satisfy the intent of the requirements as stated in BTP CMEB 9.5-1, Section C.2 with only minor deviations, as indicated in revised Fire Protection Evaluation Report, Appendix B.

APPENDIX B

DEVIATIONS FROM BRANCH TECHNICAL POSITION CMEB 9.5-1 (NUREG 0800, JULY 1981)

2.d and e

Instead of a permit system, NNECo intends to use maintenance requests/work orders. All maintenance requests/work orders are reviewed and approved by responsible foremen, supervisors, or designees, which have received indoctrination on fire protection/prevention during plant staff training. The responsible foremen, supervisors, or trained designees are qualified to determine if a trained fire watch is or is not required as well as to determine the fire protection measures which should be observed during the operation. 280.4

2.o

In regards to fire fighting procedures as outlined in Section C.2.o, NNECo believes that the development of specific fire fighting procedures is not realistic because various combinations of fire situations could develop and specific procedures would actually restrict fire fighting by reducing flexibility. 280.4

Fire fighting strategies for safety related areas will be presented to the fire brigade members during the classroom portion of the Fire Brigade Training Program. This will include an active discussion between fire brigade leaders, fire brigade members, and classroom instructor on the best possible approaches and methods for fighting various types of fires in specific safety related areas.

5.a (1) (a) (b)

Safety related systems are isolated from fire hazards.

Redundant safety systems are separated from each other by fire barriers and fire shields except where the fire analysis shows a minimum fire loading or an alternate shutdown path is available to bring the plant to cold shutdown.

5.e (2)

Since there exists area fire detection (both smoke and heat) and the safe shutdown evaluation has determined that there exists an alternate means of bringing the plant to cold shutdown on loss of any fire area, line type heat detectors are not used in each cable tray. 280.12

6.b. (6)

Millstone Unit 3 is shared with Units 1 and 2. Pump running, driver availability, and failure to start are functions of the Unit 1 fire pumps and alarm in the Unit 1 control room.

6.b (9)

The fire-water supply basically consists of two 245,000-gallon fire-water tanks. Three fire-water pumps are interconnected (piped and valved) such that each pump can take a suction from either or both tanks.

Fire-water supply tanks are valved to ensure that a leak in one tank or the associated piping would not cause both tanks to drain. Water supply to the fire-water tanks is through a 12-inch city water main that can refill a tank in 8 hours. As a backup source to the city water main, well water is available through a reversible elbow connection (normally left disconnected). Fire-water tanks are strictly piped for fire-protection water storage and are independent of sanitary- or service-water storage.

The largest existing single demand for fire water for the entire system is expected to be from the Millstone 3 main transformer deluge system, which requires 1,500 gallons per minute. It is calculated that, with this expected flow and the 1,000 gallons per minute required for hose streams, sufficient water storage and flow capacity is available with two fire-water pumps running.

6.b (11) See comment for 6.b (9)

6.c (4)

There are no Category I water supplies to the Fire Protection System to protect systems required for safe shutdown. This is due to the fact that the Appendix R evaluation has determined that there exists an alternate shutdown path for any fire damaged equipment.

7.a (1)

Reactor coolant pumps are provided with a seismic oil collection system to preclude the possibility of spraying hot oil and starting a fire.

7.b

Since the control room is continuously manned, there are smoke detectors throughout the control room complex, and there is an alternate means of bringing the plant to cold shutdown from outside the control room, no detectors are located within the cabinets and consoles in the control room.

7.c

The primary means of fire suppression in the cable spreading room is total flooding CO_2. Adequate water coverage by water spray could not be assured due to cable tray sizing and arrangement. Back-up manual hose stations are provided outside this fire area.

Since there exists area fire detection (both smoke and heat) and the safe shutdown evaluation has determined that there exists an

alternate means of bringing the plant to cold shutdown on loss of the cable spreading room, line type heat detectors are not used in each cable tray.

7.e

Switchgear rooms are protected by a total flooding CO_2 system. No floor drains are provided, but all switchgear is placed on pads to preclude any water damage in the event back-up hose stations have to be used.

NRC Letter: May 3, 1983

Question No. Q280.5

Describe the plant fire brigade that will be provided to comply with BTP CMEB 9.5-1 Section C.3.

Response:

Refer to the revised Millstone Unit 3 Fire Protection Evaluation Report, Section 3.3, for the response to this question.

NRC Letter: May 3, 1983

Question No. Q280.6

Describe the plant fire brigade equipment that will be provided to comply with BTP CMEB 9.5-1 Section C.3.c.

Response:

All of the necessary fire fighting equipment as outlined in BTP CMEB 9.5-1, Section C.3.c will be provided for fire brigade use prior to fuel load date.

NRC Letter: May 3, 1983

Question No. Q280.7

Describe the fire brigade training program that will be provided to comply with BTP CMEB 9.5-1 Section C.3.d.

Response:

NNECo is currently developing a Millstone Unit No. 3 fire brigade training program. This program will satisfy the requirements of BTP CMEB 9.5-1, Section C.3.d with the exception of minor word changes to Sections C.3.d. 4 and 7.b. NNECo intends to change the wording, "every 3 months" to "quarterly", in order to be consistent with its fire brigade training program for its other nuclear facilities. The fire brigade training program will be completed and fully implemented prior to fuel load.

NRC Letter: May 3, 1983

Question No. Q280.9

Describe how the fire doors will be kept closed and supervised by one of the measures stated in BTP CMEB Section C.5.a.

Response:

In order to maintain the integrity of the fire barrier walls, properly rated fire doors equipped with automatic self-closing mechanisms will be provided for all door openings within fire walls. Automatic self-closing mechanisms will ensure that the fire door returns to its closed position.

Millstone Unit 3 Fire Door Inspection Program will be incorporated into Millstone Units 1 and 2 Existing Fire Door Inspection Station Procedures.

NRC Letter: May 3, 1983

Question No. Q280.10

Describe how fire protection has been provided for safe shutdown so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage and that systems necessary to achieve and maintain cold shutdown from either the control room or the emergency control station(s) can be repaired within 72 hours.

Provide an analysis which shows that one redundant train of equipment structures, systems, and cables necessary for safe shutdown can be maintained free of fire damage (BTP CMEB 9.5-1, C.5.b) by either:

a. Separation of cables and equipment and associated circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers should be protected to provide fire resistance equivalent to that required of the barrier

b. Separation of cables and equipment and associated circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system should be installed in the fire area

c. Enclosure of cable and equipment and associated circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system should be installed in the fire area

Response:

An evaluation has been performed for the plant and the results are reported in the Fire Protection Evaluation Report (FPER). The specific items requested are addressed in Sections 6, 7, 8, and 9 of the FPER.

NRC Letter: May 3, 1983

Question No. Q280.11

Identify those areas of the plant that will not meet the guidelines of Section C.5.b of BTP CMEB 9.5-1 and, thus alternative shutdown will be provided. Verify that all other areas of the plant will be in compliance with Section C.5.b of BTP CMEB 9.5-1.

Response:

An evaluation has been performed for the plant and the results are reported in the Fire Protection Evaluation Report (FPER).

The specific subject of alternative shutdown is addressed in FPER, Section 8.1. Description of shutdown methods and identification of area-by-area success paths are provided in FPER, Section 6. Section 8.2 through 8.6 describe items relocated to safe shutdown evaluation problem areas.

NRC Letter: May 3, 1983

Question No. Q280.12

Describe how redundant safety related cable systems outside the cable spreading room are protected to comply with BTP CMEB 9.5-1 Section C.5.e(2).

Response:

Refer to revised Appendix A, Section D.3(c) and revised Appendix B, Section 5.e(2) of the Fire Protection Evaluation Report for the response to this question.

TABLE 1.9-2 (Cont)

Plant operational procedures will address long-term ventilation outages.

SRP BTP CMEB 9.5-1 (SECTION 9.5.1)

SRP TITLE: GUIDELINES FOR FIRE PROTECTION FOR NUCLEAR POWER PLANTS

A. Actual differences between FSAR and SRP

1. All safety-related systems are not separated from potential fires in nonsafety-related areas by fire barriers as required by BTP Section C.5.a(1)(a).

2. Redundant safety systems are not in all cases separated from each other by fire barriers and fire shields as required by BTP Section C.5.a(1)(b).

3. No continuous line-type heat detectors for cable trays are used in any plant area, as required by BTP Section C.5.e(2).

4. Alarms indicating pump running, driver availability, and failure to start required by BTP Section C.6.b(6) to be provided in the control room are provided in the Unit 1 control room, as these are functions of the Unit 1 fire pumps. Millstone 3 shares the fire protection system with Millstone 1 and 2.

5. The fire-water supply consists of two 245,000 gallon fire-water tanks. A 300,000 gallon minimum is specified in BTP Sections C.6.b(9) and (11). 280.12

6. There are no Category-I water supplies to the standpipe system to protect systems required for safe shutdown as required in BTP Section C.6.c(4).

7. Automatic fixed suppression system is not provided for the coolant pump lube oil system within the containment, as required by BTP Section C.7.a(1).

8. Safety related cables within the reactor containment are not separated by 3 hour fire rated barrier walls nor by a radiant energy shield with one-half hour fire rating as required by BTP Section C.7.a(1)(b).

9. No smoke detectors are located within the cabinets or consoles in the control room as required in BTP Section C.7.b.

10. BTP Section C.7.c requires that the primary fire suppression in the cable spreading room should be an automatic water system. Millstone 3 utilizes total flooding CO_2.

TABLE 1.9-2 (Cont)

280.12 11. No continuous line-type heat detectors for cable trays are used inside the cable spreading room, as required by BTP Section C.7.c.

12. No floor drains are provided in the switchgear rooms, as required by BTP Section C.7.e.

B. Justification for differences from SRP

1. Safety related systems are separated from potential fires in nonsafety related areas by fire barriers. Where the fire analysis shows a minimum fire loading or an alternate shutdown path is available to bring the plant to cold shutdown, such barriers are not provided.

2. Redundant safety systems are separated from each other by fire barriers or fire shields. Where the fire analysis shows a minimum fire loading or an alternate shutdown path is available to bring the plant to cold shutdown, such barriers are not provided.

3. There exists area fire detection (both smoke and heat). Although we believe this to be adequate it has been determined that there is an alternate means of bringing the plant to cold shutdown on loss of any fire area.

4. The Unit 3 control room has dedicated communication capability with the Unit 1 control room.

5. The largest existing single demand for fire water for the
280.12 entire system is expected to be from the Millstone 3 turbine area sprinkler system, which requires 1500 gpm. It is calculated that, with this expected flow and the 500 gpm required for hose streams, sufficient water storage is available for the 2 hour fire fighting capability required by BTP Section C.6.b(11).

6. There are no seismic Category I water supplies to the standpipe system to protect systems required for safe shutdown due to the fact that the evaluation has determined that there exists an alternate shutdown path for any fire damaged equipment.

7. Reactor coolant pumps are provided with a seismic oil collection system to preclude the possibility of spraying hot oil and starting a fire. Also, a preaction suppression system is provided at electrical penetrations, and manual suppression on all elevations. The containment is equipped with automatic detection.

8. Sufficient design features exist to provide protection features equivalent to the requirements as described in Section 8 of the Millstone 3 Fire Protection Evaluation Report.

TABLE 1.9-2 (Cont)

9. The control room is continuously manned, there are smoke detectors throughout the control room complex, and there is an alternate means of bringing the plant to cold shutdown from outside the control room.

10. Adequate water coverage by water spray could not be assured due to cable tray sizing and arrangement. Back-up manual hose stations are provided outside this fire area. 280.12

11. There exists area fire detection (both smoke and heat). Although we believe this to be adequate, the evaluation has, in addition, determined that there is an alternate means of bringing the plant to cold shutdown on loss of the cable spreading room.

12. Switchgear rooms are protected by a total flooding CO_2 system. No floor drainage has been provided to insure that adequate CO_2 concentration can be maintained within the affected area. Ramps to elevated doorways are provided to prevent water from damaging equipment in other areas.

SRP 9.5.4

SRP TITLE: EMERGENCY DIESEL ENGINE FUEL OIL STORAGE AND TRANSFER SYSTEM

A. Actual differences between FSAR and SRP

1. SRP 9.5.4, Paragraph II.4.b, requires that each diesel generator be capable of operating continuously for 7 days. Each diesel fuel oil tank at Millstone 3 has a 3.5 day capacity of fuel oil.

2. SRP 9.5.4, Paragraph II.4.d addresses the use of NUREG/CR-0660 in the review of the fuel oil system meeting GDC 17. FSAR Section 9.5.4 does not address NUREG/CR-0660.

3. There is no tank design features which minimize turbulence of sediments as specified in SRP 9.5.4, Paragraph III.5.

4. The fill lines for the diesel generator fuel oil vaults are not missile protected as required by SRP 9.5.4, Paragraph III.6.a.

B. Justification for differences from SRP

1. Cross connect valves between the discharge of the fuel oil transfer pumps, which can be powered from either emergency bus, enables either diesel to run for 7 days. This was a Construction Permit (CP) commitment that was accepted during the PSAR stage. In addition, fuel oil may be delivered to the site within 24 hours from terminals in New Haven, Connecticut, or obtained from offsite storage facilities of the Applicant.

TABLE 1.9-2 (Cont)

Fuel oil is also available from onsite sources (i.e., existing fuel oil storage tanks of Millstone 1 and 2).

2. NUREG/CR-0660 considerations for diesel reliability will be addressed in a future amendment.

3. Formulation of corrosive product sediment is minimized by means of a sump and a sump pump with suitable controls for removal of condensation. Additionally, the tank interiors are coated with epoxy resin to preclude corrosion.

4. Alternate ways to fill the tank are provided through the sump piping or through the flame arrestor/vent line.

SRP 9.5.5

SRP TITLE: EMERGENCY DIESEL ENGINE COOLING WATER SYSTEM

A. Actual differences between FSAR and SRP

SRP 9.5.5, Paragraph II.4 addresses the use of NUREG/CR-0660 in the review of the cooling water system meeting GDC 17 and GDC 44. FSAR Section 9.5.5 does not address NUREG/CR-0660.

B. Justification fo differences from SRP

NUREG/CR-0660 considerations for diesel reliability will be addressed in a future amendment.

SRP 9.5.6

SRP TITLE: EMERGENCY DIESEL ENGINE STARTING SYSTEM

A. Actual differences between FSAR and SRP

1. SRP 9.5.6, Paragraph II.4.c addresses the use of NUREG/CR-0660 in the review of the air starting system meeting GDC 17. FSAR Section 9.5.6 does not address NUREG/CR-0660.

2. The Millstone 3 emergency generator air starting system does not utilize air dryers to remove entrained moisture as specified by SRP 9.5.6, Paragraph II.4.f

B. Justification for differences from SRP

1. NUREG/CR-0660 considerations from diesel reliability will be addressed in a future amendment.

2. The integrity of the air starting system will be maintained by periodic blowdown of the air storage tank. Other plant operating procedures consistent with the recommendations of the

TABLE 1.9-2 (Cont)

diesel manufacturer have been developed to assure proper functioning of the air starting system.

RESPONSE

Storage of all flammable liquids is in accordance with NFPA-30 and 31.

POSITION

3. Electric Cable Construction, Cable Trays, and Cable Penetrations

   (a) "Only noncombustible materials should be used for cable tray construction."

RESPONSE

Noncombustible aluminum and steel cable trays and conduit are used.

POSITION

   (b) "See BTP APCSB 9.5-1, for fire protection guidelines for cable spreading rooms."

RESPONSE

An automatic low-pressure CO_2 fire suppression system is provided for primary suppression and a fixed water system is used as a backup (manual hose stations).

Divisional cable separation is in accordance with Regulatory Guide 1.75, and separation is accomplished from the rest of the unit by 3-hour-rated fire walls. Two remote and separate entrances to cable spreading rooms are provided. Aisle separation is at least 3 feet wide and 8 feet high.

POSITION

   (c) "Automatic water sprinkler systems should be provided for cable trays outside the cable spreading room. Cables should be designed to allow wetting down with deluge water without electrical faulting. Manual hose stations and portable hand extinguishers should be provided as backup. Safety related equipment in the vicinity of such cable trays, that does not itself require water fire protection, but is subject to unacceptable damage from sprinkler water discharge, should be protected from sprinkler system operation or malfunction."

RESPONSE

Automatic CO_2 systems are provided, and manual hose stations are available as backup in the cable tunnels, motor control center area, and rod control areas of the plant. Automatic water systems are provided in other plant areas. 280.12

280.12 The Millstone 3 design includes distance separation, with either barrier, or non-combustible barrier and suppression system between redundant Class 1E circuits and between non-Class 1E and Class 1E circuits which is in agreement with Regulatory Guide 1.75 (refer to Section 8.3.1.4). Continuous line-type heat detectors for cable trays are not provided.

POSITION

(d) "Cable and cable tray penetration of fire barriers (vertical and horizontal) should be sealed to give protection at least equivalent to that fire barrier. The design of fire barriers for horizontal and vertical cable trays should, as a minimum, meet the requirements of ASTM E-119, 'Fire Test of Building Construction and Materials,' including the hose stream test."

RESPONSE

Fire stops are provided when cables pass through fire-rated floors and walls. They will provide protection consistent with the degree of hazard and will utilize the silicone foam manufactured by Dow Corning (Type Q36548 Silicone RTV Foam). The silicone foam has been tested in accordance with ASTM E 119 floor fire test and is suitable as a 3 hour fire stop.

Several fire barriers are provided in the electrical cable tunnel; a noncombustible material compose the major portion of an area, with silicone foam used to fill cable blockouts. The barriers are located where the tunnel enters different buildings (i.e., control building, service building, and auxiliary building). Penetrations from the auxiliary building through the containment are of metal construction and provide a 3 hour fire barrier.

POSITION

(e) "Fire breaks should be provided as deemed necessary by the fire hazards analysis. Flame or flame retardant coatings may be used as a fire break for grouped electrical cables to limit spread of fire in cable ventings. (Possible cable derating owing to use of such coating materials must be considered during design.)"

RESPONSE

No additional fire breaks are required as a result of the fire hazard analyses.

POSITION

(f) "Electric cable constructions should as a minimum pass the current IEEE No. 383 flame test. (This does not imply that cables passing this test will not require additional fire protection.)"

RESPONSE

All cables purchased for use at Millstone 3 are specified to pass, as a minimum, the IEEE 383 flame test (refer to FSAR Section 9.5.1.1.8). 280.13

Cables furnished as part of an equipment package which are not flame retardant are routed in dedicated raceways for their entire length.

POSITION

(g) "To the extent practical, cable construction that does not give off corrosive gases while burning should be used."

RESPONSE

To the extent possible, cable construction that does not give off corrosive gases while burning will be used. The fire-retardant characteristics of the cables are accomplished by the addition of halogens.

POSITION

(h) "Cable trays, raceways, conduit, trenches, or culverts should be used only for cables. Miscellaneous storage should not be permitted, nor should piping for flammable or combustible liquids or gases be installed in these areas."

RESPONSE

Cable raceways are not shared with other facilities.

POSITION

(i) "The design of cable tunnels, culverts and spreading rooms should provide for automatic or manual smoke venting as required to facilitate manual fire fighting capability."

RESPONSE

Smoke venting is provided in cable tunnels and spreading rooms.

POSITION

(j) "Cables in the control room should be kept to the minimum necessary for operation of the control room. All cables entering the control room should terminate there. Cables should not be installed in floor trenches or culverts in the control room."

RESPONSE

Cables in control room are kept to the minimum necessary. All cables terminate there. Cables are not installed in floor trenches or culverts in the control room.

POSITION

4. Ventilation

(a) "The products of combustion that need to be removed from a specific fire area should be evaluated to determine how they will be controlled. Smoke and corrosive gases should generally be automatically discharged directly outside to a safe location. Smoke and gases containing radioactive materials should be monitored in the fire area to determine if release to the environment is within the permissible limits of the plant Technical Specifications."

RESPONSE

All exhaust from the auxiliary, fuel and waste disposal buildings is directed by ductwork with the option of being filtered for radioactivity to the ventilation vent for discharge to the atmosphere. The ventilation vent is monitored for radioactivity.

POSITION

(b) "Any ventilation system designed to exhaust smoke or corrosive gases should be evaluated to ensure that inadvertent operation or single failures will not violate the controlled areas of the plant design. This requirement includes containment functions for protection of the public and maintaining habitability for operations personnel."

RESPONSE

Redundancy of ventilation equipment precludes single failure. Inadvertent operations will not violate controlled areas of plant design due to the radiation monitoring system in the vent stack as discussed in Item 4.(a) above.

Containment isolation prevents release of radioactive smoke to the atmosphere. The containment purge air system is connected to the

NRC Letter: May 3, 1983

Question No. Q280.13

Verify that electric cable construction will pass the flame test in the current IEEE Std. 383 to comply with BTP CMEB 9.5-1 Section C.5.e.

Response:

Refer to Appendix A, revised Section D.3(f) of the Fire Protection Evaluation Report for the response to this question.

NRC Letter: May 3, 1983

Question No. Q280.14

Describe how hydrogen and other flammable gas lines which are routed through safety related areas comply with BTP CMEB 9.5-1, Section 5.d(5).

Response:

Hydrogen gas lines are the only combustible gas lines permanently installed. The description complying with Section 5.d(5) of BTP CMEB 9.5-1 can be found in FSAR Section 9.5.9.1.3.

NRC Letter: May 3, 1983

Question No. Q280.16

Describe how fixed repeaters installed to permit use of portable radio communication units will be protected from exposure fire damage to comply with BTP CMEB 9.5-1 Section C.5.g.

Response:

Millstone Unit 3 intends to provide a fixed repeater station and separate base station to support its portable radio communication needs. Each unit will be remotely located from the other. The base station will provide back-up capability to the fixed repeater station.

In the event that a fire damages the Millstone Unit 3 fixed repeater station, the plant's portable radios have been equipped with multi-band frequency capability. This multi-band frequency capability will allow plant personnel the option to continue communications utilizing the base station as back-up communication center or the capability to change frequency bands and operate through either adjacent plant's fixed repeater system.

NRC Letter: May 3, 1983

Question No. Q280.17

Describe the Class A fire detection system that has been provided to comply with BTP CMEB-9.5-1 Section C.6a to protect all areas of the plant which contain or present an exposure fire hazard to safety related equipment and cables.

Response:

The fire detection system is described in Section 4.4 and Appendix A (Section E: Position 1 - Page A-24) of the Fire Protection Evaluation Report. In addition to the above, the Class A fire detection system is described as follows:

The fire detection system is designed to meet the Class A system defined in NFPA 72D as described below:

Signaling line circuits (CPU in the main control room to remote transponders) are supervised for open, shorts, and grounds).

Upon a line fault condition the system automatically transfers to a backup signaling line circuit and continues normal operation. A print-out and CRT display occurs to identify the fault and affected channel.

NRC Letter: May 3, 1983

Question No. Q280.18

Describe the primary and secondary power supplies for the fire detection systems provided to comply with Section 2220 of NFPA 72D, BTP CMEB 9.5-1 Section C6.a(6).

Response:

The primary power supply consists of 120 V ac supply feed from a 480 V normal motor control center described in FSAR Section 8.3.1.1.1.

The secondary power supply consists of:

a. For Local Zone Panels - dedicated 24 hour internal battery packs

b. For Main Console, CRT, and Printer - dedicated uninterruptable power supply system with 24 hour battery backup.

NRC Letter: May 3, 1983

Question No. Q280.20

Verify that the fire pumps and their controllers are UL listed and installed in accordance with NFPA 20 requirements. The fire pumps start-up setpoints should be adjusted such that both fire pumps do not start simultaneously (at least a 5 to 10 second delay between pump start-ups is required by NFPA 20) (BTP CMEB 9.5-1 Section C 6.B)

Response:

Millstone Unit 3 fire water supply will be provided by the Millstone Station fire pumps.

Millstone's fire pumps have been installed in accordance with the applicable requirements of the National Fire Protection Association (NFPA) Standard 20 (Installation of centrifugal fire pumps).

The fire pumps, drive units, controllers and accessories are all UL listed components.

Refer to Millstone Unit 3 Fire Protection Evaluation Report, Appendix A, Section E.2.c, for the explanation of fire pump startup setpoints.

NRC Letter: May 3, 1983

Question No. Q280.22

It is our position that the reactor coolant pumps be equipped with an oil collection system in conformance with Section C.7.a of BTP CMEB 9.5-1. Provide the design description of this system.

Response:

Refer to revised FSAR Section 9.5.11 for the response to this question.

housing and the shaft. The flow splits with the major portion flowing down the shaft through the radial bearing and into the reactor coolant system. The remaining seal injection flow passes up the shaft through the seals.

Component cooling water (Section 9.2.2.1) is provided to the thermal barrier heat exchanger. During normal operation, the thermal barrier limits the heat transfer from hot reactor coolant to the radial bearing and to the seals. In addition, if a loss of seal injection flow should occur, the thermal barrier heat exchanger cools the reactor coolant to an acceptable level before it enters the bearing and seal area.

The reactor coolant pump motor oil lubricated bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearing is a double-acting Kingsbury type. Component cooling water is supplied to the external upper bearing oil cooler and to the integral lower bearing oil cooler. Each RCP motor is
280.22 equipped with an oil collection system to mitigate the consequences of oil leaks. Section 9.5.11 describes this system in detail.

The motor is a drip-proof, squirrel-cage, induction motor with Class B thermalastic epoxy insulation, and fitted with external water/air coolers. The rotor and stator are of standard construction and are cooled by air. Six resistance temperature detectors are imbedded in the stator windings to sense stator temperature. A flywheel and an anti-reverse rotation device are located at the top of the motor.

The internal parts of the motor are cooled by air. Integral vanes on each end of the rotor draw air in through cooling slots in the motor frame. This air passes through the motor with particular emphasis on the stator end turns. It is then routed to the external water/air heat exchangers, which are supplied with chilled water (Section 9.2.2.2). Each motor has two such coolers, mounted diametrically opposed to each other. Coolers are sized to maintain optimum motor operating temperature. The air is finally exhausted to the containment environment.

Each of the reactor coolant pump assemblies is equipped for continous monitoring of reactor coolant pump shaft and frame vibration levels. Shaft vibration is measured by two relative motion shaft probes mounted on top of the pump seal housing: the probes are located 90 degrees apart in the same horizontal plane and mounted near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90 degrees apart in the same horizontal plane and mounted at the top of the motor support stand. Proximeters and converters linearize the probe output which is displayed on monitor meters in the control room. The monitor meters automatically indicate the highest output from the relative probes and seismoprobes; manual selection allows monitoring of individual probes. Indicator lights display caution and danger limits of vibration.

The spool piece, a removable shaft segment, is located between the motor coupling flange and the pump coupling flange. The spool piece allows removal of the pump seals with the motor in place. The pump

applicable start signal. System design provides the capability to perform Type C testing as specified by Appendix J of 10CFR50.

9.5.10.5 Instrumentation Requirements

Containment air pressure instrumentation is part of the containment leakage monitoring system (Section 6.2.6.1.5). Details of this instrumentation are discussed in Section 7.6.10.

The containment vacuum pumps are manually activated from the control room and are inte.locked with the two containment vacuum system isolation valves. When either isolation valve in a pump suction line is closed, the vacuum pump is stopped automatically. The containment vacuum system isolation valves close automatically on a Phase A containment isolation signal. The containment vacuum pumps are also stopped automatically by a pump discharge temperature signal greater than 300°F.

The containment vacuum system isolation valves have control switches and indicator lights on the main control board. Open and closed positions are monitored by the plant computer. Engineered safety feature status lights indicate on the main control board when an isolation valve is open.

The containment isolation valve for the vacuum system ejector suction is manually controlled from the main control board. The valve is provided with a keylock control switch and indicator light. The key can be removed only in the closed position.

Annunciators are provided on the main control board that alarm when the following conditions exist:

1. Containment Vacuum Pump A - discharge temperature High
2. Containment Vacuum Pump B - discharge temperature High
3. Any MCC load power not available (Status lights on rear of main control board indicate which MCC is without power.)

A local total flow indicator is installed in the combined discharge line for the containment vacuum pumps. The indicator is used to monitor the containment structure leakage rate (Section 6.2.6.1).

9.5.11 Reactor Coolant Pump (RCP) Oil Collection System

The RCP oil collection system incorporates enclosures having drip pans and splash guards at potential oil leakage sites to reduce the possibility of oil fires caused by ignition of oil leakage by hot RCS components and to maintain cleanliness of area.

280.22

9.5.11.1 Design Bases

The RCP oil collection system is designed in accordance with the following:

1. General Design Criteria 2 for structures housing the system and system components to withstand effects of natural phenomena such as earthquakes, tornados, and floods without loss of function

2. Regulatory Guide 1.29 for the seismic classification of system components.

3. Paragraph C.7.a(1)(e) of BTP CMEB 9.5-1.

9.5.11.2 System Description

The RCP motor OSPS consists of a package of splash guards, drip pans, and enclosures assembled as attachments to the RCP motor at strategic locations. These enclosures do not interfere with RCP ventilation or bearing insulation, or the seal maintainenace stand. Shroud enclosures are removable to facilitate maintenance.

The oil collected at the shroud enclosures is gravity drained to four oil collection tanks, one for each RCP. Each oil collection tank has a capacity of approximately 320 gallons and is vented to containment through a flame arrestor/vent assembly. Removal of oil from the collection tank is accomplished via a hose connection on the tank and a portable pump. 280.22

The collection and control package consists of:

1. Oil Cooler and Oil Cooler Piping Enclosure

   The motor oil cooler has a number of flanged connections which represent potential sources of oil leaks. The entire oil cooler and connecting oil piping are therefore provided with an enclosure which will collect any leaks which occur. This enclosure will be designed to provide maximum access to the oil cooler through the use of multiple piece removable construction. Handles will be provided as necessary for pieces which would be difficult to install and remove without them. Pieces will be of such a size and configuration that they can be handled by one man without hoists or lifts. A drain suitable for draining the leakage oil will be provided.

2. Upper Oil Level Alarm Enclosure

   A drip pan will be placed under the upper oil level alarm detector to collect any oil that may leak from the associated piping fittings. The pan will have deep, removable sides to protect against atomizing of the leakage oil by the air currents around the motor. A viewing window will be provided for reading the oil level sight glass. A drain connection is included.

3. Upper Oil Fill and Drain Pipe Enclosure

A drip pan will be placed under the oil fill and drain valve to collect any leaks from the valve. The pan will have deep, removable sides to protect against atomizing of the leakage oil by air currents around the motor. A drain connection is provided.

4. Upper RTD Conduit Box Enclosure

A drip pan will be placed under the upper RTD conduit box. This pan will also have deep, removable sides. A drain connection is provided.

5. Oil Lift System Enclosure

The oil lift system provides high-pressure oil to the motor thrust bearings during startup. A leak in this system could result in oil being sprayed on hot system components. The oil lift system enclosure isolates the high-pressure oil from the environment in the event that the system should leak during its operation.

The enclosure will be designed to provide maximum access to the oil lift pump and motor through the use of multiple piece removable construction. A viewing window will be provided in the enclosure. The pieces of the enclosure will be of a size and configuration such that they can be handled by one man without hoists and lifts. Handles will be provided where appropriate. A drain connection will be provided.

280.22

6. Lower Bearing Oil Pot Drip Pan

This catch basin is located immediately below the lower bearing oil pot and is removable. The pan surrounds the shaft and extends to the lower bracket edge thus protecting the entire underside of the lower oil pot.

7. Upper Bearing Oil Pot Drip Pan

This catch basin is an integral part of the upper bracket. It surrounds the shaft and would catch oil which might come over the standpipe. A drain connection runs to a point external to the upper bracket.

9.5.11.3 Safety Evaluation

There are no moving parts where failure could jeopardize system function in the oil collection system.

Earthquakes and fires are the only natural and postulated phenomena which might affect the operation of this system. The RCP oil collection system is seismically supported.

NRC Letter: May 3, 1983

Question No. Q280.24

Verify that smoke detectors have been provided in all control room cabinets and consoles in accordance with BTP CMEB 9.5-1 Section C.7.b.

Response:

Fire Protection Evaluation Report, Appendix B, Item 7.6 and FSAR Table 1.9-2, SRP BTP CMEB 9.5-1, Item 8 identify the deviation from BTP CMEB 9.5-1 Section C.7.b, and provide justification for the fact that there are no detectors located within the cabinets and consoles in the control room.

NRC Letter: May 3, 1983

Question No. Q280.26

Verify that the loss of ventilation in the safety related battery rooms is alarmed in accordance with BTP CMEB 9.5-1 Section C.7.g.

Response:

See revised FSAR Section 9.4.1.5 for response to this question.

10. Expansion tank chilled water level

11. Chilled water system A or B trouble

12. Chilled water pump A or B flow Low

13. Service water pump A or B flow Low

14. Air flow battery rooms 1, 3, and 5 280.26

15. Air flow battery rooms 2 and 4

Indicators

1. Differential pressure between chiller equipment space and control room

2. Hydrogen level for each battery room 280.26

3. Pressure for each air storage tank

4. Air storage tank reduced pressure

The following instrumentation and controls are located on the main control board:

Annunciators

1. Any motor control center power not available

2. Control building isolation signal bypass Train A and bypass Train B

3. Fire - Control building inlet ventilation smoke

4. Vibration monitor (common)

Power not available status lights are provided on the rear of the main control board for each motor control center.

A smoke detection status light is provided on the fire protection panel for the control building.

All radiation monitor alarms annunciate in the control room.

9.4.2 Fuel Building Ventilation System

The fuel building ventilation system (Figure 9.4-2) removes heat generated by equipment and water vapor from fuel pool evaporation, prevents moisture condensation on interior walls, provides a suitable environment for equipment operation and personnel. It also limits potential radioactive release to the atmosphere during normal operation or anticipated operational transients, and following a postulated fuel handling accident (FHA).

NRC Letter: May 3, 1983

Question No. Q281.6 (Section 9.3.2)

You did not indicate in the FSAR that the chemical additive tank in the CVCS system will be sampled. Confirm that these tanks will be sampled according to Standard Review Plan 9.3.2.

Response:

The function of the chemical mixing tank is to provide a means of adding a known concentration of either hydrazine or lithrium hydroxide to the reactor coolant system. The amount of solution to be injected is based on an analysis of a reactor coolant sample. A prepared solution of a known concentration is added to the chemical mixing tank, and the tank is then completely filled with primary grade water. The entire contents of the 5 gallon volume chemical mixing tank will be discharged into the RCS for each pH adjusting operation. The solution is flushed through the tank to the charging pump suction and into the reactor coolant system. After about one hour, the reactor coolant system is sampled to check the chemical addition for the desired results.

A prepared solution of known concentration is used in the chemical mixing tank. Therefore, the capability to sample the tank is not necessary.

NRC Letter: May 3, 1983

Question No. Q281.11 (10.4.1)

The FSAR states on page 10.4-3, second paragraph, that the tube sheet material in the condensers is aluminum-bronze. On page 10.4-19, second paragraph, line 10, the tube sheet material in the condenser is identified as copper-nickel. Verify the correct material. The information is needed for the dissimilar metal junction capability evaluation.

Response:

The Millstone 3 condenser tube sheet material is aluminum bronze ASTM B-171, CDA alloy No. 613.

See revised FSAR Section 10.4.5.3.

equipment in other systems required for safe shutdown of the facilities and/or required to limit the consequences of an accident is suitably protected from potential flooding caused by rupture of the non-Category I pipe and components of the circulating water system. Internal plant protection is provided by watertight areas specifically designed for such flooding conditions, by elevation, or by system and component design to ensure that such failures are not possible, as discussed under Circulating Water Expansion Joint Rupture, in this section.

Leakage of seawater from the circulating water system into interfacing systems is minimized by the use of highly corrosion resistant materials as barriers between the circulating water system and its interfacing systems. The most prominent of these barriers is the condenser (Section 10.4.1) with tubes fabricated from titanium, which is highly reliable for seawater application. The tube material minimizes erosion due to high steam entrance velocity and protects the insides of the tubes from damage which could be caused by local high velocity circulating water. To further prevent leakage of seawater into the condenser hotwell, aluminum-bronze tube sheets of | 281.11
double integral design are provided. Seal water from the condensate system (Section 10.4.7) is injected into each inlet and outlet tube sheet at a pressure which is greater than the operating pressure of the circulating water system.

Should any seawater leakage occur from the circulating water system into interfacing systems, this leakage is detected in the condenser hotwell by the turbine plant sampling system (Section 9.3.2.2). Section 10.4.1 discusses the detection of seawater leakage into the condenser.

The circulating water system is protected from excessive pressure transients caused by multiple circulating water pump trips deriving from loss of all electrical power. Vacuum breaker valves, located on each condenser inlet and outlet water box, are automatically opened by any two circulating water pumps tripping within one minute of each other. Hydraulic transient analyses were performed on the circulating water system to determine the most critical operating conditions which would yield the most severe transient pressures (both positive and negative) within the system. It was determined that a loss of power which leads to a simultaneous loss of all six circulating water pumps would produce the most severe pressures. The design transient pressures for the circulating water system, based on loss of power and opening of the vacuum breaker valves, fall within the maximum design pressure/vacuum envelope of the circulating water system.

Circulating Water Expansion Joint Rupture

There are no essential systems or components required for safe shutdown or to mitigate the effects of an accident, located within the turbine building which could be affected by flooding due to a circulating water pipe or expansion joint rupture. In addition, there are no passageways, pipe chases, or cableways that could be

NRC Letter: May 3, 1983

Question No. Q281.12 (9.1.2)

Provide a description of any materials monitoring program for the spent fuel pool. In particular provide information on the frequency of inspection and type of samples used in the monitoring program.

Response:

Refer to revised FSAR Section 9.1.2.3 for the response to this question.

9.1.2.3 Safety Evaluation

The design and safety evaluation of the spent fuel racks is in accordance with the NRC position paper, "Review and Acceptance of Spent Fuel Storage and Handling Applications, April 1978."

The racks are designated ANS Safety Class 3 and Seismic Category I and are designed to withstand normal and postulated dead loads, live loads, loads due to thermal effects, and loads caused by the operating basis earthquakes and safe shutdown earthquake events.

The design of the racks is such that the K $\leq$0.95 under all conditions including fuel handling accidents. Due to the close spacing of the cells, it is impossible to insert a fuel assembly in other than design locations. The space between the rack periphery and the pool wall is also sufficiently small to preclude inadvertent insertion of a fuel assembly.

The racks are also designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of the spent fuel bridge and hoist. The new fuel handling crane, which is capable of carrying loads heavier than a fuel assembly, is prevented by interlocks or administrative controls, or both, from bringing the load over the spent fuel pool. The fuel storage racks can withstand an uplift force equal to the uplift capability of the spent fuel bridge and hoist.

All materials used in construction are compatible with the spent fuel pool environment and all surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. All the materials are corrosion resistant and will not contaminate the fuel assemblies or pool environment.

In order to monitor the effectiveness of neutron absorber material, design provisions have been made for a materials monitoring program. The program consists of two surveillance coupon assemblies. Each assembly has eight packets, each containing three neutron absorbing
281.12 material coupons of the same neutron absorbing material used in the full racks. The packets are attached to each other by hanger rods which allow removal and periodic inspection to verify the effectiveness of the neutron absorbing material.

One assembly is used for short term testing and the other for long term surveillance. The short term assembly has one packet removed and analyzed after each of the first eight refuelings; the long term assembly has one packet removed and analyzed every five years over the 40 year life of the plant.

After each refueling, the short term assembly is moved to a location adjacent to a newly removed spent fuel assembly in order to assure a conservative evaluation of the neutron absorbing material. This phase of the materials monitoring program is completed after eight refuelings.

There are three coupons in each packet to provide a statistical significance to the test. 281.12

Design of the facility in accordance with Regulatory Guide 1.13 ensures adequate safety under normal and postulated accident conditions.

The methodology used in the criticality analysis is discussed in Section 4.3.2.6.

9.1.3 Fuel Pool Cooling and Purification System

The fuel pool cooling and purification system (Figure 9.1-6) removes decay heat from spent fuel stored in the fuel pool and provides adequate clarification and purification of water in the fuel pool, refueling cavity, and refueling water storage tank. Table 9.1-1 lists the principal component design characteristics for the system. Table 9.1-2 gives the fuel pool cooling system performance characteristics. Figure 3.8-63 shows equipment locations.

9.1.3.1 Design Bases

The fuel pool cooling and purification system is designed in accordance with the following criteria:

NRC Letter: May 3, 1983

Question No. Q410.14 (Section 9.1.3)

Is there any portion of the spent fuel pool cooling and cleanup system designed to nonseismic category requirements? If so, verify that failure of the nonseismic Category I portion in an earthquake will not affect the operation of the cooling trains.

Response:

Refer to revised FSAR Section 9.1.3.3, and Figure 9.1-6 for the response to this question.

other nickel bearing alloys are sources of soluble radionuclides. These impurities and radionuclides enter the fuel pool through the fuel transfer tube in the form of a hydrated film adhered to the spent fuel assemblies.

Borated water from the RWST is used to fill the fuel pool at a concentration matching that used in the refueling cavity during refueling operations.

Normal makeup to the fuel pool, necessitated by losses due to evaporation, is primary grade water from the primary grade water system (Section 9.2.8) or borated water from the refueling water storage tank (Section 6.2.2), a Seismic Category I tank. Periodic sampling from local sample connections is performed to check the boron concentration of the fuel pool water. Boric acid is added manually, if required, from the dry boric acid inventory to maintain the minimum boron concentration of 1,950 ppm in the fuel pool water. Water from the safety related service water system can be used as an emergency supply to the spent fuel pool. A permanently installed service water pipe will be provided, terminating in the fuel pool area. Should an emergency arise, a temporary piping connection can be made to provide service water to the fuel pool. In addition, water from the fire protection system is available.

Drain lines to the purification pumps are provided at low points in the refueling cavity to remove the water remaining below the reactor vessel flange following refueling. A tap line from the drain lines leads to the containment sump. This arrangement makes it possible for water from the quench spray system and containment recirculation system which falls into the reactor cavity to feed the containment recirculation system. The valves on the tap line are open during plant operation and closed during refueling. The purification pumps transfer the water from the refueling cavity to the RWST. The transfer canal dewatering pump transfers water from the transfer canal to the fuel pool. The spent fuel cask pool has a drain line to the purification pumps. A blank flanged, permanently installed, piping arrangement terminates in the spent fuel shipping cask storage area. Should this piping arrangement be needed, a temporary flanged spool piece can be inserted in the line to enable one of the fuel pool purification pumps to pump the water within the spent fuel shipping cask storage area either through the prefilters or through the prefilters, demineralizer, and postfilter to the boron recovery tanks (Section 9.3.5). Administrative procedures are followed to assure that the cask storage area gate is inserted in the transfer slot in the wall separating the fuel pool from the spent fuel shipping cask storage area before pumping commences. However, the design of the gate is such that even with the gate open, the fuel pool cannot be drained below the top of the active fuel region of the fuel assemblies. 410.14

Piping, valves, and components of this system making contact with the fuel pool water are austenitic stainless steel which is corrosion-resistant to the boric acid solution.

A sample connection is provided downstream of the fuel pool demineralizer for sample removal to check the gross activity, particulate matter, boric acid concentration, and component performance.

9.1.3.3 Safety Evaluation

Two full-size fuel pool cooling pumps and two full-size fuel pool coolers will be provided to ensure 100-percent redundant cooling capacity. This portion of the system is Seismic Category I and Safety Class 3. The Seismic Category I cooling portion of the fuel pool cooling and purification system is independent of the nonseismic purification portion. Failure of the purification portion in an earthquake will not affect the operation of the cooling trains. 410.14

Each pipe which enters the fuel pool has either a 1/2 inch vent hole drilled into the pipe to act as an anti-siphoning device or terminates at an elevation above these vent holes. These provisions prevent siphoning of the fuel pool water to uncover the spent fuel (see Figure 9.1-6). 410.14

One pump and one cooler are sufficient to maintain the pool temperatures as indicated in Table 9.1-2.

The seismic Category I cooling portion of the fuel pool cooling and purification system is independent of the non-seismic purification portion. Failure of the purification portion in an earthquake will not affect the operation of the cooling trains.

Each pipe which enters the fuel pool has either a 1/2 inch vent hole drilled into the pipe to act as an anti-siphoning device or terminates at an elevation above these vent holes. These provisions prevent siphoning of the fuel pool water to uncover the spent fuel.

An evaluation of the capabilities of the spent fuel pool cooling system has been performed for normal and abnormal conditions. A range of possible fuel pool loading scenarios was evaluated and a conservative heat loading was chosen. Heats for normal refueling and emergency core offload are shown on Figures 9.1-7 and 9.1-8, respectively.

The normal refueling evaluation assumed removal of one-third core (64 assemblies) into a loaded fuel pool with remaining capacity for one and one-third cores. This evaluation was made using heat loads at 132 hours after shutdown and resulted in a maximum temperature of 125°F. The emergency core offload evaluation assumed the complete removal of a full core (193 assemblies) into an otherwise loaded fuel pool. This evaluation was made using heat load at 10 days after shutdown and resulted in a maximum temperature of 149°F.

Following a design basis accident with loss of power, the reactor plant component cooling water system is not available to cool the spent fuel pool coolers until 4 hours after the accident. Power from the emergency generators is not immediately available due to loading

considerations. A loss of cooling evaluation has been performed which shows that the spent fuel pool temperature reaches a temperature of 200°F in approximately 12.5 hours. This provides sufficient time to manually initiate pool cooling. Once the cooling is restarted, the temperature decreases to 150°F in less than 6 hours. Redundant safety grade fuel pool temperature indication is provided on the main control board. Redundant safety class 3 level instruments are located in the fuel pool which indicate both locally and in the control room. They are set to provide indication before the water level falls below 23 feet above the top of the fuel racks. Piping penetration are at least 11 feet above the top of the spent fuel so that failure of inlets, outlets or accidental piping leaks cannot reduce the water below this level.

Normal makeup water to the spent fuel pool is the primary grade water system (Section 9.2.8). Should primary grade water be unavailable, makeup water can be provided from the refueling water storage tank, a Seismic Category I source (Section 6.2.2). Water can also be provided from the hose station of the fire protection system near the spent fuel pool. In addition, as an additional safety feature for the unlikely event of failure of both cooling trains and loss of the sources above, a Seismic Category I, Safety Class 3 flow path is provided from the service water system (Section 9.2.1). To prevent contamination of the pool from service water during normal conditions, a spool piece is included at the fuel pool end of the piping, with a blind flange normally in place. Sufficient time exists before pool boiling to install the spool piece.

9.1.3.4 Inspection and Testing Requirements

The fuel pool level and temperature instrumentation is tested and calibrated on a periodic basis. The safety related trains will be tested for operability in accordance with the Technical Specifications. Visual inspection of system components and instrumentation is conducted periodically.

Safety related components will receive inservice testing and/or inspection as specified in Sections 3.9.6 and 6.6. In addition, containment isolation valves will be tested as specified in Section 6.2.6.3.

The system is in operation during refueling and whenever spent fuel is stored in the fuel pool. Therefore, system operational tests are not required.

Provisions are made for monitoring the spent fuel pool water for water purity, analyzing boron concentration, pH, and crud level from
281.5 samples taken by the reactor plant sampling system at the demineralizer inlet and outlet. Local differential pressure indicators across the filters and demineralizers are used to indicate when filters and resins should be replaced.

281.5 The pressure indicators are set for the demineralizer at 15 psid and for the filters at 25 psid. All indicators alarm at a local control

panel in the fuel building. These setpoints are based upon operating experience.

The fuel pool water is sampled weekly for pH, conductivity, chloride, fluoride, turbidity, and total gamma activity. Chloride and fluoride levels are 0.15 ppm each, while acceptable pH may vary between 4.2 and 10.5. 281.5

Boron concentration is monitored prior to refueling operations as stated in Section 9.1.4.2.2.

9.1.3.5 Instrumentation Requirements

The fuel pool has redundant safety grade low level alarms and temperature indicators provided in the main control room. Nonsafety grade level indication is provided locally and high and low level alarms are provided both locally and in the main control room.

Local temperature indicators are provided on each fuel pool cooler outlet. Fuel pool cooler inlet and outlet high temperature is alarmed locally. Fuel pool cooler outlet flow is indicated, and low flow alarmed, locally. Fuel pool cooler instrumentation is nonsafety grade.

The fuel pool cooling pumps have control switches and indicating lights in the main control room. The discharges of all pumps have local pressure indicators. Upon a high temperature at the pool, the standby fuel pool cooler is started manually. The cooling pumps can be operated manually either from the control room or the switchgear. The purification pumps are operated locally.

Flow through the fuel pool demineralizer is controlled automatically. Local differential pressure indicators are used across the filters and demineralizer to indicate cleanness.

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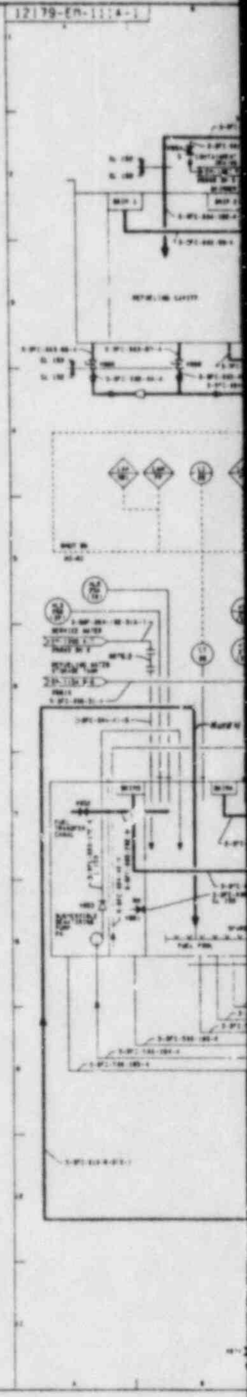
P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1



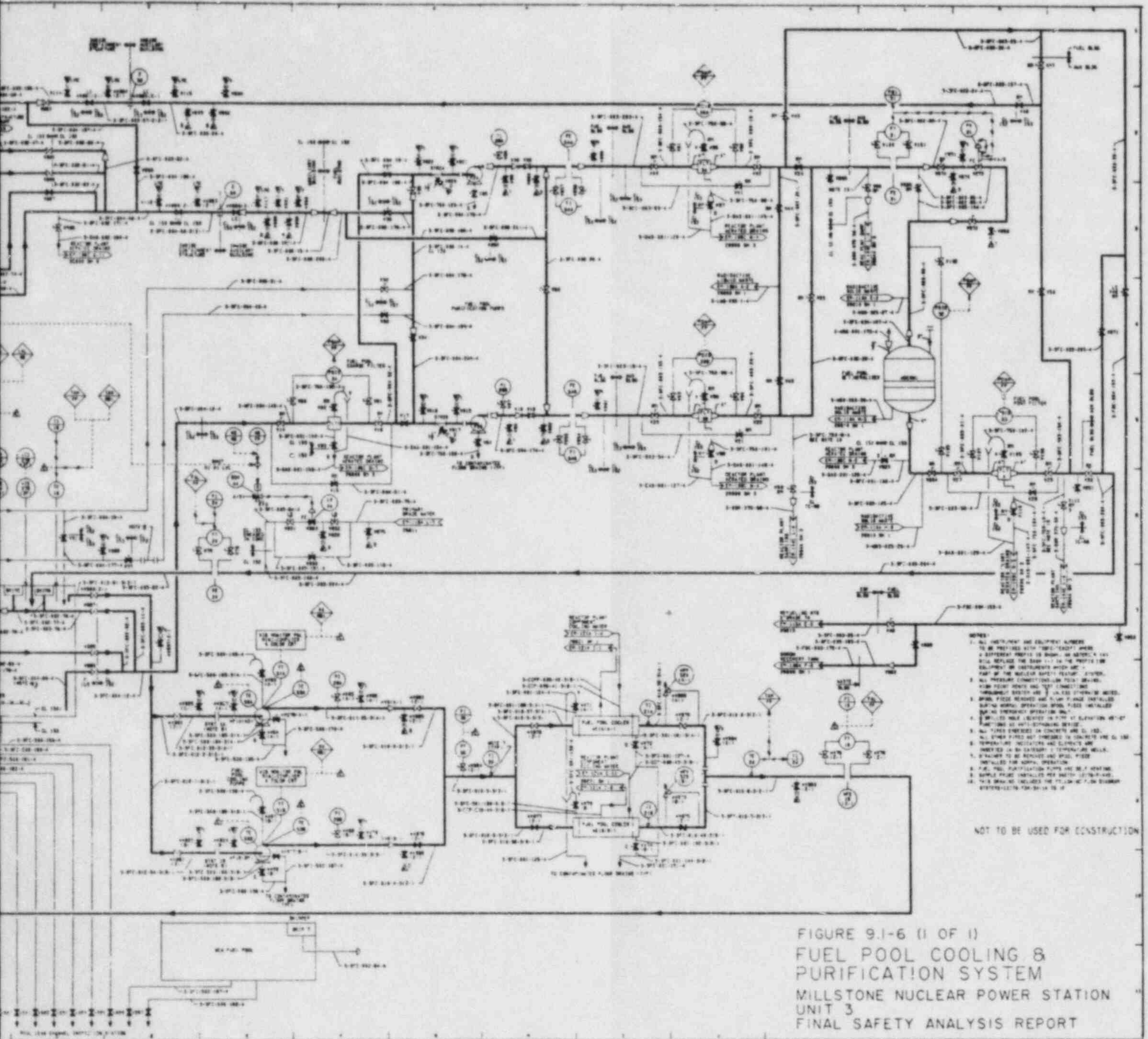


FIGURE 9.1-6 (1 OF 1)
FUEL POOL COOLING &
PURIFICATION SYSTEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Amendment 3 August 1983

Also Available On
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NRC Letter: May 3, 1983

Question No. Q410.18 (Section 9.4)

Some of the licensees have provided measures for detecting and correcting dust accumulation on safety related equipment in order to assure their availability on damand. Verify that dust accumulation does not pose a problem in this plant.

Response:

The ventilation for buildings housing safety related equipment utilize one or more of the following features to preclude dust accumulation from posing a problem.

Filtration Systems

Each air handling unit which continuously supplies outdoor air into safety related building for offsetting the equipment heat loads is provided with filters. Pressure differential switches accross the filters are also provided to monitor the filter conditions. Filters are changed routinely with the Vendor's recommendations as delineated in the Instrumentation Manuals supplied with the air handling equipment. For additional information regarding the type of ventilation systems see FSAR Sections 9.4.1, 9.4.2, 9.4.3, and 9.4.5.

Non-Filtration Systems

Supply air or exhaust air for systems which draw outdoor air into or from the safety related equipment areas such as diesel-generator building, service water pump rooms, etc., as discussed in FSAR Sections 9.4.6, 9.4.8, and 9.4.10, do not use filters.

Heavy concentrations of atmospheric dust are not anticipated to occur in outdoor areas of the plant, due to the lack of heavy industry. See FSAR Section 2.2. In addition, the design features for the non-filtered air intakes employ elevated intakes and low inlet air velocities to minimize dust accumulation in the equipment rooms.

NRC Letter: May 3, 1982

Question No. Q410.19 (Section 9.4)

Describe the effect on the safety function of the essential HVAC systems in the event of a single failure in a fire damper in the ventilation system ducts. It is our position that such a failure not compromise the safety function of the HVAC system.

Response:

Refer to revised FSAR Sections 9.4.1.3, 9.4.2.3, and 9.4.5.3 for response to this question.

9.4.1.3 Safety Evaluation

The control building air-conditioning, ventilation, and chilled water systems are designed to seismic Category I and QA Category I and are enclosed in a Category I missile and tornado protected building.

The control building habitability envelope air bottle pressurization system is seismically supported and designed to ASME B and PV Code Section VIII, Division 1 and ANSI B31.1 standards.

A radiation monitor connected with the makeup air duct of the control room area air-conditioning units will detect and respond to the presence of radioactivity. At the discretion of the operator, the emergency ventilaton system can be started manually and the return air of the control room or the outdoor air supply diverted through the emergency ventilation filter assembly.

High radiation or toxic gases detected by the monitors located in the air intakes will result in control building isolation (Section 6.4). During a control building pressurization, the system emergency ventilation and pressurization will supply breathable air to maintain a positive pressure within the control room envelope. The air intake isolation valves are opened following one hour of air bottle pressurization. These valves are manually operated from the main ventilation panel in the control room. If either valve fails to open, these valves which are located within the habitability zone can be manually opened. The valves have been designed to enable local manual activation with a rack screw operation. This design enables these valves to be opened within one hour following control room isolation.

The storage bottles are refilled from a connection located on the outside wall of the turbine building. An air tank truck can be on site within 3 days for refilling purposes.

Fusible link fire dampers are provided on openings in fire barrier separating fire areas. The dampers automatically isolate the area affected by fire. Fire damper assemblies installed in ventilation ductwork common to redundant portions of this system consist of at least two fire dampers in parallel in order to preclude a single failure of one fire damper from impairing the safety function of the system. Airtight doors, sealed penetrations and fire walls prevent smoke, heat, and carbon dioxide from entering the control room. A carbon dioxide detector is provided to monitor the control room area. A purge system is provided to remove smoke and carbon dioxide from all areas except the chiller room which has 100 percent outside air circulation. The purge system is completely independent of all control building air-conditioning and ventilation systems. The largest area served by the purge system can be ventilated at a rate of one air change per hour.

410.19

9.4.1.4 Inspection and Testing Requirements

The control building air-conditioning and ventilation system is field tested and inspected for air balance and completeness of installation.

During fuel handling, the exhaust air is manually diverted through at least one of the fuel building filtration units, in addition to reducing the supply air to 17,000 cfm, thus maintaining a negative pressure. During periods of high temperature and humidity, it may be necessary to use both fuel building filter exhaust fans to maintain proper atmospheric clarity in the spent fuel pool area.

During a fuel handling accident, the air supply is reduced to 25,000 cfm and one of the two fuel building filtration units is stopped manually, thus maintaining a negative pressure.

9.4.2.3 Safety Evaluation

During normal plant operations, the ventilation air is discharged by one nonnuclear safety related exhaust fan to the atmosphere via the ventilation vent. A particulate and gas radiation monitor is provided which samples the exhaust air stream prior to the filtration units as discussed in Section 11.5. On receipt of a high radiation alarm, the exhaust air is manually diverted through one of the fuel building filtration units, the normal exhaust fan is stopped, and the associated safety related fan is started. High radiation signals from radiation monitors located above the spent fuel pool and in the new fuel storage area alarm locally and in the control room.

The ventilation exhaust system, with the exception of the unfiltered air exhaust fan, is safety related. In addition, a single nonnuclear safety related damper is provided in the ventilation supply system to reduce air capacity in the fuel building so that a negative pressure can be maintained in the fuel building. The actuation of this damper occurs simultaneously with the filtration unit used to ensure maintenance of a negative pressure within the building. In the event of a failure in the nonnuclear safety related supply system, the safety related wall-mounted backdraft dampers shall admit the required makeup air. This operation prevents potentially contaminated air from leaving the spent fuel pool area. The filtered exhaust system is provided with redundant 100-percent capacity fans, dampers, and filtration units.

All ventilation exhaust ductwork is seismically supported. Ventilation supply ductwork located above the spent fuel pool and portions that compromise the integrity of safety related systems are also seismically supported. The ventilation exhaust system components, excluding the unfiltered air exhaust fan, are QA Category I and Seismic Category I. The damper in the ventilation supply system to the fuel building is QA Category II and Seismic Category II. The wall-mounted backdraft dampers are QA Category I and Seismic Category I. These categories are discussed in Section 3.2.

A standby redundant safety related fuel building ventilation exhaust system is provided to assure that a loss of functional performance capability of the system does not occur due to a single active failure. Upon low flow in the operating exhaust fan discharge line,

the standby system is automatically started as discussed in Sections 7.3.2 and 9.4.2.5.

410.19 Fire damper assemblies installed in ventilation ductwork common to redundant portions of this system consist of at least two fire dampers in parallel in order to preclude a single failure of one fire damper from impairing the safety function of the system.

9.4.2.4 Tests and Inspections

Inspections and testing of fuel building ventilation filter systems are consistent with the requirements outlined in NRC Regulatory Guide 1.52, Rev. 2.

Test programs consist of predelivery shop and qualification tests, initial in-place acceptance tests, and post-operation surveillance testing.

Filter housing leak tests, performed in accordance with ANSI-N510, are conducted at the shop and during in-place acceptance testing. This test demonstrates leakage rates of less than 0.02 percent of rated design flow at design pressure.

Each HEPA filter is factory tested to demonstrate a minimum efficiency of 99.97 percent when tested with a 0.3 micron DOP aerosol at 100 percent and 20 percent of rated flow. After delivery and installation each HEPA bank is tested with DOP in accordance with ANSI N510 to confirm a penetration of less than 0.05 percent at rated flow.

Carbon media qualification and batch tests for the charcoal filters are performed prior to shipment to demonstrate compliance with Regulatory Guide 1.52, Revision 2 requirements. After the adsorber cells are charged with the qualified carbon, the adsorber section is leak tested with freon in accordance with ANSI N510. This test is performed to confirm that bypass leakage through the adsorber section is less than 0.05 percent.

An airflow distribution test is performed on the upstream HEPA bank. Flow distribution across each HEPA filter will be demonstrated to be within ±20 percent of the average air flow.

Test cannisters are provided to allow periodic removal of carbon samples for laboratory testing to be sure that adequate capacity exists for the collection of radioiodines.

The fans are operationally tested following installation.

System availability is assured by the surveillance requirements imposed by the applicable plant Technical Specifications (Chapter 16).

9.4.2.5 Instrumentation Requirements

A temperature controller mounted in the spent fuel pool area supply ductwork maintains the spent fuel pool area temperature at 85°F by modulating the hot water temperature control valve for the inlet air hot water heater, provided outside air temperature is less than 50°F and spent fuel pool temperature is higher than 100°F. When any one of the above two conditions is not present, the hot water temperature control valve is closed. The control circuit of the valve can also be activated manually with an open-close control switch mounted on the local control panel.

The fuel building normal exhaust fan has a control switch and indicator lights on the main heating and ventilation panel in the control room. The normal exhaust fan starts automatically when any

failure of this nonessential system will not preclude operation of any essential safety related systems.

The ESF building emergency ventilation system is safety related and is required to operate during or after a postulated accident.

All of the safety related ESF building ventilation subsystems are located in a Seismic Category I structure that is tornado, missile, and flood protected. The redundant components are connected to redundant Class 1E buses and can function as required in the event of loss of offsite power. The safety related ESF building ventilation system can withstand a single active component failure or failure of one of its Class 1E electric power sources without degrading the performance of the safety function.

The safety related ESF building ventilation system uses equipment of proven design. All components are specified to provide maximum safety and reliability. Consequences of probable component failures are tabulated in Table 9.4-7.

Each of the redundant safety injection and quench spray pump areas, residual heat removal pump and heat exchanger areas, and the containment recirculation pump and cooler areas has its own ventilation system. The redundant ventilation system ensures that, in the event of a ventilation unit failure, a second train is available. The auxiliary feedwater pumps and mechanical room areas have ventilation systems with two trains of 100-percent capacity supply and exhaust fans with common supply and exhaust ductwork. Fire damper assemblies installed in ventilation ductwork common to redundant portions of this system consist of at least two fire dampers in parallel in order to preclude a single failure of one fire damper from impairing the safety function of the system. These redundant fans ensure the integrity of this duct system. 410.19

During normal plant operation, safety related systems do not operate.

During plant shutdown, the safety injection and quench spray pump areas and the residual heat removal pump and heat exchanger areas safety related ventilation subsystems are required to operate. These ventilation subsystems are designed to automatically start whenever the residual heat removal pumps, quench spray pumps, or safety injection pumps are started.

During a postulated accident, the ESF building emergency ventilation subsystems automatically start whenever any of their respective safety related pumps start. These ventilation systems supply and exhaust air throughout their equipment areas to maintain environmental conditions at which the pumps and coolers can perform their safety functions. Upon a failure of any of the safety related units in one train, the redundant train can maintain the areas at the designed conditions.

All areas in which safety related equipment is located are monitored for high temperature and annunciated in the control room. Upon a

high temperature alarm within one of the areas the operator can switch to the redundant system for backup. Upon receipt of a safety injection signal, the residual heat removal pump and heat exchanger areas, safety injection and quench spray pump areas, and containment

NRC Letter: May 3, 1983

Question No. Q410.20 (Section 9.4.1)

Describe the measures for assuring a proper operating environment for essential control room and ESF switchgear room air handling units when the normal control building HVAC system is not available during emergency conditions.

Response:

The control room HVAC and the control building (ESF) switchgear room HVAC systems are discussed in FSAR Sections 6.4 and 9.4.

For both systems the air handling units used during normal operation are the same air handling units used during emergency conditions. The design of these systems considers the worst case heat loads occuring during normal and emergency plant conditions.

The control room HVAC air handling units are located in the control building HVAC mechanical equipment room, which is part of the control room habitability envelope. The control building switchgear room air handling units are located in the switchgear rooms they serve. Therefore, the proper operating environments are maintained by the units themselves (Refer to Figure 3.8-64).

NRC Letter: May 3, 1983

Question No. Q410.23 (Section 10.3.3)

Describe the purpose of the block valve located downstream of the main steam pressure relieving bypass valve and the reason why it is not locked open.

Response:

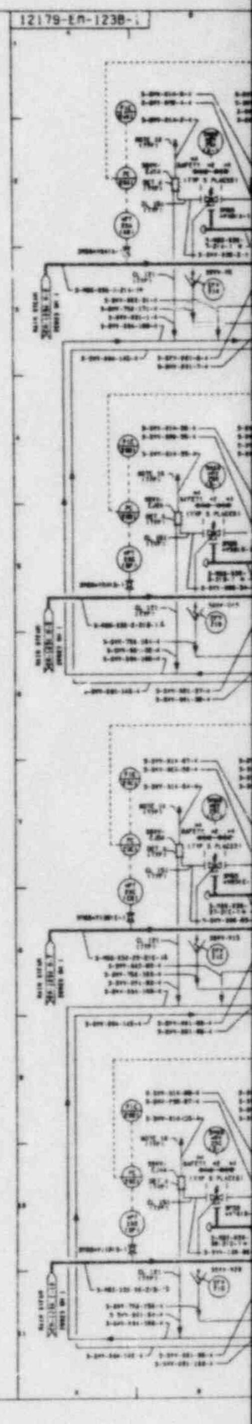
FSAR Figure 10.3-1 has been revised to delete this block valve.

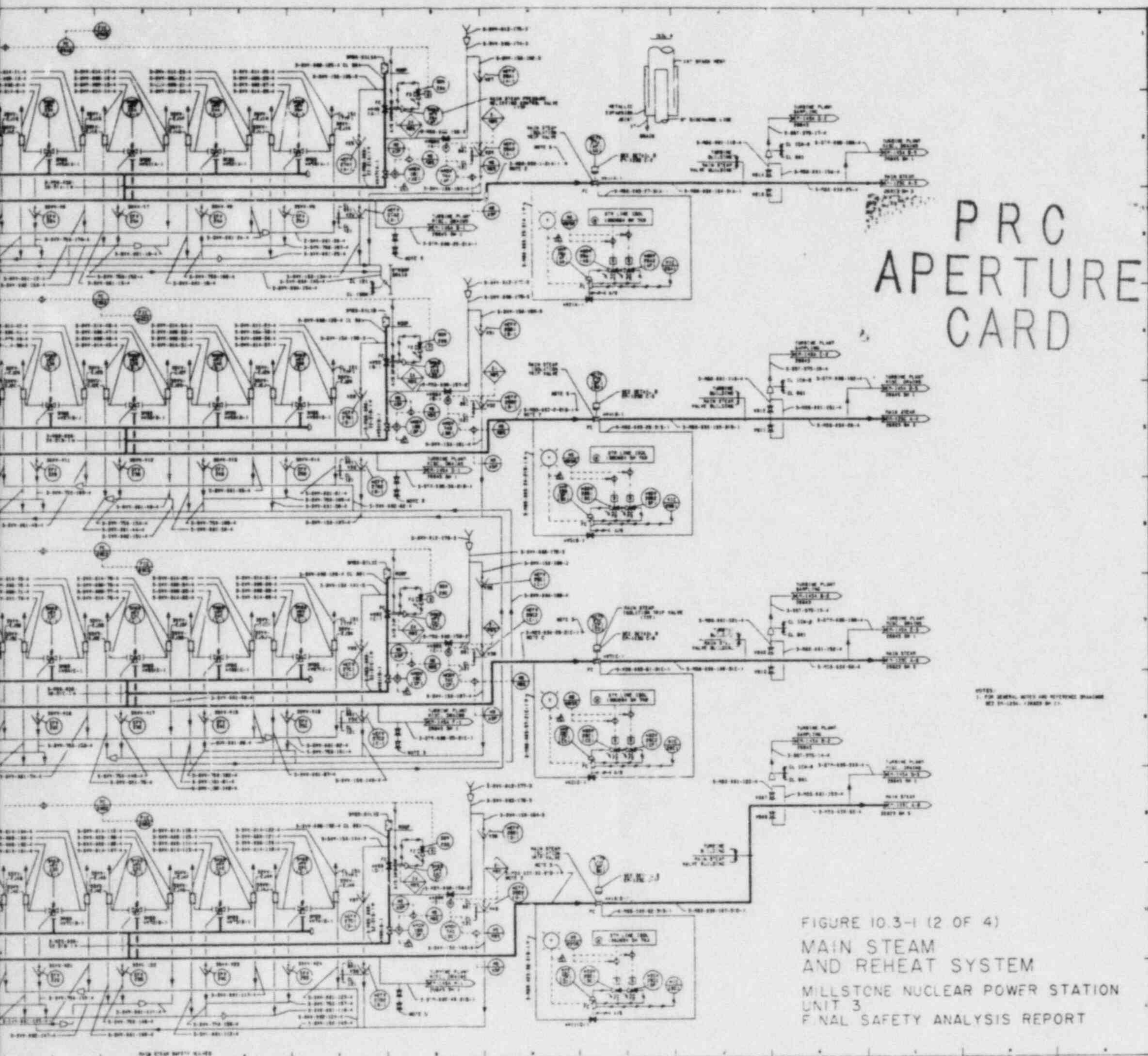
P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1

P&ID NO.	FSAR FIGURE NO.
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3

P&ID NO.	FSAR FIGURE NO.
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1





PRC
APERTURE
CARD

FIGURE 10.3-1 (2 OF 4)
MAIN STEAM
AND REHEAT SYSTEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Amendment 3 August 1983

Also Available On
Aperture Card

8307260197-06

NRC Letter: May 3, 1983

Question No. Q410.25 (Section 10.4.5)

In the evaluation of potential flooding of essential plant areas as a result of a circulating water system failure, credit cannot be taken for operator action to stop the circulating water pump in 15 minutes to contain the spillage water in the turbine building to elevation 21 feet 6 inches. Indicate the water level in the turbine building to which it will eventually rise and verify that this level of water will not affect any essential systems or components.

Response:

Reference Question 10.7 of the Millstone 3 PSAR on the subject of water level in the turbine building. This question included direction from the staff to assume a 15 minute time period before operator action is taken to stop circulating water flow. This number was reflected in FSAR Section 10.4.5.3.

NRC Letter: May 3, 1983

Question No. Q410.29

The Applicant has stated that the unit is capable of remaining at hot shutdown status for 72 hours. The Applicant should confirm that the unit is capable of attaining cold shutdown status within 72 hours of reactor trip using only onsite power.

Response:

Revised Section 8.2 of the Fire Protection Evaluation Report (FPER) details the ability of the Millstone 3 plant to maintain the secondary heat sink for a minimum of 72 hours using various sources of water. Revised FPER Section 9.2, in conjunction with Section 6.2 and 6.3, shows that Millstone 3 has the capability of being brought to a cold shutdown condition within 72 hours of reactor trip utilizing only onsite power.

SECTION 8

RESOLUTION OF SAFETY SHUTDOWN EVALUATION PROBLEM AREAS

8.1 ALTERNATIVE SHUTDOWN CAPABILITY

In order to assess compliance with the requirements of Appendix R, maximum credit was taken for alternative shutdown capability with the functional flexibility of Millstone 3 system design. The alternative shutdown methods are summarized below:

Function	Alternative Method A	Alternative Method B
Reactor Coolant Letdown (Figure 6-2)	Normal Letdown Path	Reactor Head Vent
Auxiliary Feedwater Injection (Figure 6-3)	Motor-Driven Pump(s)	Turbine-Driven Pump
Steam Release (Decay Heat Removal) (Figure 6-5)	Atmospheric Dump Valves	Code Safety Valves
Boration (Figure 6-6)	Charging Pumps from Boric Acid Tank	High-Head Safety Injection Pump from RWST
Reactor Coolant System Depressurization (Figure 6-7)	Auxiliary Spray Line	Pressurizer Power Operated Relief Valves

The methods are described in more detail in Section 6.2.

8.2 LONG TERM (72-HOUR) HOT SHUTDOWN

To maintain secondary heat sink, the auxiliary feedwater system supplies water to the steam generators. This allows removal of heat from the reactor coolant system.

A minimum of 800,000 gallons must be made available to the auxiliary feedwater system in order to provide the required water volume for 72 hours of hot standby followed by a 6-hour cooldown.

The auxiliary feedwater and condensate makeup and drawoff system designs provide 340,000 useable gallons in the DWST and 200,000 gallons in the condensate storage tank. The two tanks are normally supplied from the water treating storage tank (3WTS-T1) using the water treating supply pumps (3WTS-P1A, B). Since these pumps are nonsafety related, it is conceivable that they will be unavailable during a fire, which could result in loss of normal ac power. 410.29

An additional 260,000 gallons is supplied directly to the DWST from the domestic water system via a 2 inch line to the DWST fill line.

410.29 This 2 inch line will pass 127 gpm, supplying the additional 260,000 gallons in approximately 34 hours.

8.3 CONTROL SYSTEM ISOLATION FOR CONTROL ROOM/SPREADING ROOM/IRR FIRES

In evaluating the consequences of fires in the control room, instrument rack room, and cable spreading room, it was determined to be necessary to add a controls system to transfer signals from the affected areas to the switchgear rooms. This control system, along with certain manual actions (Section 9), allows the plant to be brought to cold shutdown without the use of the control room, instrument rack room, or cable spreading room. A summary of these additions follows:

1. Two new fire transfer panels, one for each of the emergency switchgear rooms, are provided. Each of these panels contains the required number of control transfer switches, power supplies, and signal conditioning electronics required for safe shutdown of the plant.

   The transfer switch function is to disassociate the ASP controls and indicators from their normal support components, which may be lost by fire, and replace those signals with signals from the new transmitters. Block diagrams of transfer schemes are provided in Figure 8-1.

2. Transfer switches 3RCS*HCV442A/B and 3CHS*MV8105 are relocated from the auxiliary shutdown panel to each of the fire transfer panels.

3. Additional instrumentation for monitoring plant process variables is provided. These include six pressure transmitters, five level transmitters, four flow transmitters, two neutron sensors, and neutron flux processing racks.

   Two environmentally qualified neutron detectors will be installed in the spare wells in the neutron shield tank. Two qualified electronic channels will be installed in the new fire transfer panels. The electronics will be provided from a vital instrument bus.

4. Key lock control switches at the local motor control centers are added for the following: 3RCS*MV8000B, 3HVC*ACU3A,B and 4A,B. These are provided for local administrative control at the MCCs in the event of fire in the main control room.

8.4 FIRE PROTECTION OF CABLE

As the safe shutdown functions were evaluated, cabling which supplied power to all required components was also considered (e.g., motor valves and equipment). The analysis of the plant design concluded that only a few cables needed protection. Cables for 3RCS*PCV455A and 3RCS*PCV456 will be protected by a 1-hour fire barrier inside the

auxiliary building. Cables for 3NMS-N131C and 3MNI-N135C will be protected with 1-hour fire barriers in the auxiliary building, service building tunnel, and control building.

8.5 AUXILIARY BUILDING ELEVATION 24 FEET-6 INCHES

Major components of the two systems considered in the safe shutdown evaluation are located at auxiliary building elevation 24 feet-6 inches These are the three charging pumps (CHS) and the three component cooling pumps (CCP).

Normally, both of these systems supply cooling water to the reactor coolant pump seals during the hot shutdown period. However, only one source of cooling is needed when the pumps are not in operation. In addition, CCP is required for cold shutdown to remove heat from the residual heat removal system. Because these systems are within the same fire area, Millstone 3 has installed automatic suppression and detection in this area.

The CCP pumps are approximately 60 feet from the charging pump cubicle. In addition to this separation, manual hose stations are located throughout the area at elevation 24 feet-6 inches If all of the component cooling water pumps were damaged, Millstone 3 has the onsite capability to repair a train of component cooling within 72 hours. These precautions and modifications strengthen the position that both the charging pumps and the component cooling pumps will not be simultaneously damaged by a fire.

8.6 REACTOR CONTAINMENT

Although the design of Millstone 3 containment does not meet the letter of Appendix R requirements, sufficient design features exist to support a request for an exemption in the area, based upon the following consideration:

> The Millstone containment structure is subatmospheric, normally unmanned, and equipped with both fire detection and suppression. Manually initiated suppression consists of a water sprinkler system for the electrical penetration area. The reactor coolant pumps are equipped with a seismic oil collection system which is capable of collecting the entire reactor coolant pump motor lubrication system oil volume. Additional suppression is provided by fire hose stations and portable dry chemical extinguishers. Electrical cables are separated in orange and purple cable trays; these trays are approximately 16 feet apart with a low quantity of combustibles between the trays.

We have concluded that the above described protection features provided is equivalent to the requirements of Appenndix R.

SECTION 9

OPERATOR ACTIONS AVAILABLE FOLLOWING A FIRE

9.1 OPERATOR GUIDELINES

Following a fire, equipment normally used to bring the plant down to cold shutdown conditions may be inoperable. Table 9-1 identifies plant functions which may be affected by a fire in different fire areas, and describes the alternative operator actions available to fulfill these functions. The information in Table 9-1 will be considered during the preparation of Millstone Unit 3 operating procedures.

9.2 REPAIR OF EQUIPMENT

After a fire, some equipment may have to be repaired before achieving and maintaining cold shutdown. The safe shutdown evaluation concluded that there is only one area of the Millstone 3 plant where any major repairs could be required. This would occur if all three of the component coolant water pumps were damaged by fire in Area AB-1, Auxiliary Building, elevation 24 feet-6 inches.

The Millstone 3 plant has the capability to repair or replace one pump motor in either train of component cooling water using onsite material (e.g., spare motor and cables) and still achieve cold shutdown conditions within 72 hours of reactor trip using only onsite power. This capability allows Millstone 3 to fully comply with Appendix R, III.G.1.b. Other minor repairs, such as replacement of fuses or circuit breakers, can be accomplished well within the 72-hour requirement. No repairs are necessary to achieve hot standby or hot shutdown. This is a awmiautomatic function which does not require any manual actions outside of the control room or emergency operating facility. 410.29

NRC Letter: May 3, 1983

Question No. Q410.30

The Applicant should provide reactor coolant hot leg temperature indicator on the auxiliary shutdown panel for direct reading of process variables to control the reactor shutdown.

Response:

Reactor coolant hot leg temperature is provided as indicated in Table 7.4-1, page 1 of 5, of the FSAR.

NRC Letter: May 3, 1983

Question No. Q430.3 (SRP Section 8.1)

Criterion 50 of Appendix A to 10CFR50, IEEE Standard 485, Regulatory Guide 1.63 and Branch Technical Positions ICSB 4, PSB-1 and PSB-2 have not been identified in Table 8.1-2 of the FSAR; thus, a positive statement as to compliance with these criteria and staff guidelines has not been provided in the FSAR. Provide a statement of compliance and justify areas of noncompliance.

Response:

Refer to revised FSAR Section 8.1, Table 8.1-2. As stated in FSAR Section 1.9, Table 1.9-2, PSB-1 is currently under review and will be addressed in a future amendment.

TABLE 8.1-2

ACCEPTANCE CRITERIA FOR ELECTRIC POWER

Criteria	Title	FSAR Section Applicability 8.1	8.2*	8.3.1	8.3.2	Remarks	
1. 10CFR50							
10CFR50.34	Contents of Applications: Technical Information	X	X	X	X		
10CFR50.36	Technical Specifications	X	X	X	X	See Chapter 16	
10CFR50.55a	Codes and Standards	X	X	X	X	See Chapter 3	
2. General Design Criteria (GDC), Appendix A to 10CFR50							
GDC-1	Quality Standards and Records	X	X	X	X	See Section 3.1.2.1	
GDC-2	Design Bases for Protection against Natural Phenomena	X	X	X	X	See Section 3.1.2.2	
GDC-3	Fire Protection	X	X	X	X	See Section 3.1.2.3	
GDC-4	Environmental and Missile Design Bases	X	X	X	X	See Section 3.1.2.4	
GDC-5	Sharing of Structures, Systems, and Components	X	X	X	X	See Section 3.1.2.5	
GDC-13	Instrumentation and Control	X	X	X	X	See Section 3.1.2.13	
GDC-17	Electric Power Systems	X	X	X	X	See Section 3.1.2.17	
GDC-18	Inspection and Testing of Electrical Power Systems	X	X	X	X	See Section 3.1.2.18	
GDC-21	Protection System Reliability and Testability	X	X	X	X	See Section 3.1.2.21	
GDC-22	Protection System Independence	X		X	X	See Section 3.1.2.22	
GDC-33	Reactor Coolant Makeup	X	X	X	X	See Section 3.1.2.33	
GDC-34	Residual Heat Removal	X	X	X	X	See Section 3.1.2.34	
GDC-35	Emergency Core Cooling	X	X	X	X	See Section 3.1.2.35	
GDC-38	Containment Heat Removal	X	X	X	X	See Section 3.1.2.38	
GDC-41	Containment Atmosphere Cleanup	X	X	X	X	See Section 3.1.2.41	
GDC-44	Cooling Water	X	X	X	X	See Section 3.1.2.44	
GDC-50	Containment Design Basis	X	X	X	X	See Section 3.1.2.50	430.3
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:							
IEEE Std 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations	X		X	X	See 10CFR50.55a(h) and Reg. Guide 1.62	
IEEE Std 288-1969 (ANSI C37.92-1972)	Guide for Induction Motor Protection	X		X			
IEEE Std 308-1974	Standard Criteria for Class IE Power Systems for Nuclear Power Generating Stations	X		X	X	See Reg. Guide 1.32	

TABLE 8.1-2 (Cont)

Criteria	Title	FSAR Section Applicability 8.1	8.2*	8.3.1	8.3.2	Remarks
IEEE Std 317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X		X	X	See Reg. Guide 1.63
IEEE Std 323-1974	Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations	X		X	X	See Reg. Guide 1.89 and Section 3.11
IEEE Std 323A-1975	Supplement to the Foreword of IEEE Std 323-1974	X		X	X	
IEEE Std 334-1975	Standard for Type Tests of Continuous Duty Class IE Motors for Nuclear Power Generating Stations	X		X		See Reg. Guide 1.40
IEEE Std 336-1971	Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations	X	X	X	X	See Reg. Guide 1.30
IEEE Std 338-1977	Standard Criteria for the Periodic Testing of Nuclear Power Generating Station	X	X	X	X	See Reg. Guide 1.118
IEEE Std 344-1975	Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations	X		X	X	See Reg. Guide 1.100 and Section 3.10
IEEE Std 379-1972	Application of the Single Failure Criterion to Nuclear Power Generating Station Class IE Systems	X		X	X	See Reg. Guide 1.53
IEEE Std 382-1972 (ANSI N41.6)	Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations	X		X		
IEEE Std 383-1974 (ANSI N41.10-1975)	Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations	X		X	X	See Reg. Guide 1.131
IEEE Std 384-1974	Standard Criteria for Independence of Class IE Equipment and Circuits	X		X	X	See Reg. Guide 1.75

TABLE 8.1-2 (Cont)

Criteria	Title	FSAR Section Applicability 8.1	8.2*	8.3.1	8.3.2	Remarks
IEEE Std 387-1977	Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Stations	X		X		See Reg. Guide 1.108
IEEE Std 415-1976	Guide for Planning of Pre-Operational Testing Programs for Class IE Power Systems for Nuclear Power Generating Stations	X		X	X	
IEEE Std 420-1973 (ANSI N41.17)	Guide for Class IE Control Switchboards for Nuclear Power Generating Stations	X		X	X	
IEEE Std 450-1975	Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations	X			X	See Reg. Guide 1.129
IEEE Std 484-1975	Recommended Practice for Installation Design and Installation of Large Storage Batteries for Generating Stations and Substations	X			X	See Reg. Guide 1.128
IEEE Std 485-1978	Recommended Practice for Sizing Large Lead Storage Batteries for Generating Station and Substations	X			X	
4. Regulatory Guides (RG)						
RG 1.6	Independence between Redundant Standby (Onsite) Power Sources and between Their Distribution Systems	X		X	X	See Section 1.8
RG 1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies	X		X		See Section 1.8
RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	See Section 1.8
RG 1.29	Seismic Design Classification	X		X	X	See Section 1.8
RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	See Section 1.8
RG 1.32	Use of IEEE Std 308, "Criteria for Class IE Electric Systems for Nuclear Power Stations"	X	X	X	X	See Section 1.8

430.3

TABLE 8.1-2 (Cont)

Criteria	Title	FSAR Section Applicability				Remarks	
		8.1	8.2*	8.3.1	8.3.2		
RG 1.40	Qualification Tests of Continuous-Duty Motors Installed inside the Containment of Water-Cooled Nuclear Power Plants	X		X		See Section 1.8	
RG 1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	X	X	X	X	See Section 1.8	
RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	See Section 1.8	
RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X		X	X	See Section 1.8	
RG 1.62	Manual Initiation of Protective Actions	X		X	X	See Section 1.8	430.3
RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Reactors	X		X	X	See Section 1.8	
RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Nuclear Power Plants	X	X	X	X	See Section 1.8	
RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 3	X	X	X	X	See Section 1.8	
RG 1.73	Qualification Tests of Electric Valve Operators Installed inside the Containment of Nuclear Power Plants	X		X		See Section 1.8	
RG 1.75	Physical Independence of Electric Systems	X		X	X	See Section 1.8	
RG 1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	X		X	X	Use in conjunction with BIP ICSB-7. See Section 1.8	
RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X		X	X	See Section 1.8 and 3.11	
RG 1.93	Availability of Electric Power Sources	X	X	X	X	See Section 1.8	
RG 1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	X		X	X	See Section 1.8 and 3.10	
RG 1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	X		X		See Section 1.8	

TABLE 8.1-2 (Cont)

Criteria	Title	FSAR Section Applicability 8.1	8.2*	8.3.1	8.3.2	Remarks	
RG 1.108	Periodic Testing of Diesel Generators Used as Onsite Electric Power Stations at Nuclear Power Plants	X		X		See Section 1.8	
RG 1.118	Periodic Testing of Electric Power for Protection System		X	X	X	See Section 1.8	
RG 1.120	Fire Protection Guidelines for Nuclear Power Plants	X	X	X	X	See Section 1.8	
RG 1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	X			X	See Section 1.8	
RG 1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	X			X	See Section 1.8	
RG 1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light Water-Cooled Nuclear Power Plants	X		X	X	See Section 1.8	
5. Branch Technical Positions (BTP) EICSB							
BTP ICSB 1(PSB) - Rev. 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X		X	X		
BTP ICSB 2(PSB) - Rev. 1	Diesel Generator Reliability Qualification Testing	X		X			
BTP ICSB 4(PSB) - Rev. 1	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines					See Section 7.6.4	430.3
BTP ICSB 8 (PSB) - Rev. 1	Use of Diesel Generator Sets for Peaking	X		X			
BTP ICSB 11(PSB) - Rev. 1	Stability of Offsite Power Systems	X	X				
BTP ICSB 15(PSB) - Rev. 1	Reactor Coolant Pump Breaker Qualification	X	X	X			
BTP ICSB 17(PSB) - Rev. 1	Diesel Generator Protective Trip Circuit Bypasses	X		X			
BTP ICSB 18(PSB) - Rev. 1	Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves	X		X			
BTP ICSB 21(PSB) - Rev. 1	Guidance for Application of Reg. Guide 1.47	X	X	X	X		430.3
BTP PSB 2	Criteria for Alarms and Indications Associated with Diesel Generator Unit Bypassed and Inoperable Status	X		X			

TABLE 8.1-2 (Cont)

Criteria	Title	FSAR Section Applicability 8.1	8.2*	8.3.1	8.3.2	Remarks
6. American National Standards Institute (ANSI)**						
ANSI C37	Power Switchgear	X		X	X	
ANSI C50	Rotating Electrical Machinery	X		X		
ANSI C57	Transformer, Regulators, and Reactors	X		X		
7. Insulated Cable Engineers Association (ICEA)**						
ICEA P-46-426	Power Cable Ampacities	X	X	X	X	
ICEA P-54-440	Standard Publication "Ampacities-Cables in Open Top Trays"	X	X	X	X	
ICEA S-61-402	Thermoplastic - Insulated Thermoplastic - Jacketed Cables	X	X	X	X	
ICEA S-68-516	Ozone Resistant Ethylene Propylene Rubber Insulation	X	X	X	X	
ICEA S-66-524	Crosslinked Thermosetting Polylene Cables	X	X	X	X	
ICEA S-19-81	Applicable Test Power Cable Insulation and Jacket	X	X	X	X	
ICEA S-67-401	Metallic and Associated Coverings for Impregnated-Paper - Insulated Cables	X	X	X	X	
ICEA S-56-434	Polyethylene-Insulated Thermoplastic Jacketed Cables	X	X	X	X	
8. National Electrical Manufacturers Association (NEMA)						
NEMA AB-1	Molded Case Circuit Breakers	X		X	X	
NEMA EI2	Instrument Transformers	X		X		
NEMA FU1	Low-Voltage Cartridge Fuses,	X		X	X	
NEMA ICS	Industrial Controls, and Systems	X		X	X	
NEMA PB-1	Panelboards	X		X	X	
NEMA PB-2	Dead-Front Distribution Switchboards	X		X	X	
NEMA PV-5	Constant-Potential Type Electric Utility (Semiconductor Static Converter) Battery Chargers	X			X	
NEMA SG3	Low Voltage Power Circuit Breakers	X		X		
NEMA SG4	AC High Voltage Power Circuit Breaker	X		X		
NEMA SG5	Power Switchgear Assemblies	X		X		

TABLE 8.1-2 (Cont)

Criteria	Title	FSAR Section Applicability 8.1	8.2*	8.3.1	8.3.2	Remarks
NEMA SG6	Power Switching Equipment	X		X		
NEMA TR-1	Transformers, Regulators, and Reactors	X		X		
NEMA MG1	Motors and Generators	X		X	X	
NEMA WC5	Thermoplastic - Insulated Wire and Cable	X	X	X	X	
NEMA VE-1	Cable Tray Systems	X				
9. Miscellaneous**						
MIL C-17	Coaxial Cable	X		X	X	
NFPA No. 70	National Electric Code	X	X	X	X	
NFPA No. 78	Lightning Protection Code	X	X	X		
UL Standard 96A	Installation Requirements - Master Labeled Lightning Protection System	X	X	X		

NOTES:

* The preferred power system is not a Class 1E system and is designed as a normal system based on good engineering practice and experience. The intent is to consider, where applicable, non-Class 1E systems, the GDC, IEEE Standards, Regulatory Guides, and Branch Technical Positions as indicated

** The issue, including Addenda, in effect on the date of the Request for Proposal for purchase of the specific equipment

NRC Letter: May 3, 1983

Question No. Q430.5 (SRP Section 8.2)

The Millstone design provides two immediate access offsite circuits between the switchyard and the 4.15 kV Class IE buses. It is the staff position that these two circuits be physically separate and independent such that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. The physical separation and independence of these two circuits has not been described or analyzed in the FSAR. Provide the description and analysis and justify areas of noncompliance with the above staff position. The analysis should include separation and independence of control and protective relaying circuits as well as the power circuits.

Response:

The design of two offsite circuits from the 345 kV switchyard to the 4.16 kV Class 1E buses is via separate transformers (main/normal station service and reserve station service). FSAR Figure 8.1-1 shows the tie lines, transformer and bus arrangement connections.

The tie lines to the main/normal station service transformers and to the reserve station service transformer are physically separate and electrically independent. The main/normal station service transformers and the reserve station service transformers are located at opposite ends of the plant. The connection from the transformers to the 4.16 kV Class 1E buses are presented in the response to Question 430.8.

The control power for these buses is from different dc panels and batteries. The breakers in the Class 1E buses (34C and D) are independently protected with separate relaying.

These circuits are completely redundant and separated so that no single failure can disable both offsite power supplies to the Class 1E buses.

NRC Letter: May 3, 1983

Question No. Q430.6 (SRP Section 8.2)

The Millstone design arrangement provides two immediate access offsite circuits. One of these circuits utilizes a generator circuit breaker to isolate the turbine generator from the main and normal station service transformers. Other facilities that utilize generator circuit breakers have been required to perform verification testing. Provide a verification test program with results to demonstrate the breaker's ability to perform its intended function during steady-state operation, power system transients, and major faults.

Response:

The capabilities of the generator circuit breaker have been demonstrated by design tests and conformance tests made on similar breakers supplied to US users. The breaker capabilities will also be verified by certain production tests. The testing complies with ANSI C37.09 - 1979 as well as the more specific proposed standard Test Procedure for AC High Voltage Generator Circuit Breakers Rated on a Symmetrical Current Basis, C37.09b.1/D1, presently being developed by a Working Group of the IEEE Switchgear Committee. A discussion of the various tests follows:

A. Dielectric Tests

CERDA Test Report 1738A documents the design dielectric tests (Duke Power Breaker). Low frequency (50 HZ) withstand tests of each pole will be conducted in the factory at 0 psig air pressure.

The breaker will have the dielectric capabilities for a rating of 36 kV maximum, 170 kV BIL, even though the application for Millstone Unit 3 requires 25.2 kV and 150 kV, respectively.

B. Load Current Switching

CERDA Test Report 2090A describes a test of 40 load current switching operations performed for Public Service Company of New Hampshire.

C. Fault Current Interrupting Capability

KEMA Test Report 292-81A describes short circuit tests performed on one pole of a TVA breaker with the same interrupting rating of 275 kA symmetrical. The maximum interrupting rating required is 230.9 kA symmetrical.

D. Maximum Rate of Rise of Recovery Voltage

The same KEMA test report, 292-81A, demonstrated RRRV capability of about 5 kV/microsecond. A Duke Power Breaker was tested with an applied RRRV of 12 kV/microsecond.

E. Short-Time Current Carrying Capability

The one-second short time current capability of 275 kA RMS was demonstrated for Duke Power as shown in KEMA Test Report 2283-74A.

F. Momentary Current Carrying Capability

The momentary (close and latch) capability of 1000 kA peak was demonstrated on a Duke Power Breaker. See KEMA Test Report 2945-78A.

G. Transformer Magnetizing Current Interruption

This capability was demonstrated for Duke Power, as shown in CERDA Test Report 2000A.

H. Thermal Capability

EdF Test Report HM 51 02 806A documents tests made for TVA.

One pole of the Millstone 3 generator circuit breaker will be subjected to heat run tests to measure the temperature rises, both with normal cooling systems operating and with various losses of cooling equipment simulated.

The nameplate capability of NUSCo's breaker will be 37.5 kA continuous, even though the maximum continuous current will be 34.4 kA.

I. Mechanical Operation Test

One pole of a Duke Power generator breaker was subjected to 2000 no-load operations. 200 of these were done at -20°C ambient temperature, and 200 operations were performed with the hottest spot of the breaker at 105°C.

In factory tests, NUSCo's breaker will perform 20 operations with various conditions of low, normal, and high control voltage and air pressure. It will operate a minimum of six times during timing tests. Then it will be operated 100 more times.

NRC Letter: May 3, 1983

Question No. Q430.8 (SRP Sections 8.2 and 8.3.1)

Each of the 4.16 kV Class 1E buses at Millstone is supplied power from preferred offsite and standby onsite circuits. It is the staff position that these circuits should not have common failure modes. Physical separation and independence of these circuits has not been described or analyzed in the FSAR. Provide a description and analysis in accordance with Section 5.2.1(5) of IEEE Standard 308-1974.

Response:

Refer to revised FSAR Sections 8.3.1.1 and 8.3.1.2.1, for the response to the question.

8.3 ONSITE POWER SYSTEMS

8.3.1 AC Power Systems

The ac power systems (Figure 8.1-1) are required to distribute power for unit station service loads. The ac power systems are designed to distribute power reliably to all station auxiliaries required for startup, normal operation, normal shutdown, and emergency shutdown of the unit.

8.3.1.1 Description

The onsite ac power systems consist of the normal and Class IE systems. The normal system supplies nonsafety related equipment. The Class IE system has the redundancy, capacity, capability, and reliability to supply power to all safety related loads. This system ensures a safe plant shutdown to mitigate accident effects, even in the event of a single failure in accordance with General Design Criteria 17, 33, 34, 35, 38, 41, and 44 (Table 8.1-2).

The one-line diagram (Figure 8.1-1) illustrates the connections of the preferred normal and alternate offsite circuits, the standby onsite circuits, power supply feeders, busing arrangements, and electrical separation of safety and nonsafety related systems. General physical separation of systems in the plant is shown on Figure 8.3-1 and 8.3-7. 430.8

The safety related equipment is divided into two redundant and independent load groups with each group capable of safely shutting down the plant. Equipment associated with each load group is identified by color code to allow easy identification.

The Class IE onsite power systems have independence such that no single failure or common mode failure (including single protective relay, interlock or switchgear failure) causing loss of offsite power, limits the Class IE power system in accomplishing its intended function.

The offsite power sources have independence such that no single failure (including loss of one source) limits the Class IE power system in accomplishing its intended function.

8.3.1.1.1 Normal AC Power System

The normal ac power system consists of station service transformers, 6.9 kV buses, 4.16 kV buses, 480 V load centers, and 480 V motor control centers. The normal 120 V ac instrument power requirements are met by inverters fed from the stub 480 V motor control center (Figure 8.3-2).

The station service transformer system consists of two normal station service 3-winding tranformers and two reserve station service 3-winding transformers. Normal station service transformer A is rated 24/32/40 MVA OA/FA/FA 22.8 kV/4.16 kV/4.16 kV; normal station service

transformer B is rated 30/40/50 MVA OA/FA/FA 22.8 kV/6.9 kV/6.9 kV; reserve station service transformer A is rated 27/36/45 MVA OA/FOA/FOA 345 kV/4.16 kV/4.16 kV; and reserve station service transformer B is rated 30/40/50 MVA OA/FOA/FOA 345 kv/6.9 kv/6.9 kV.

During normal operation, power is supplied through the normal station service transformers from the unit generator via the isolated phase bus duct, with the generator breaker closed. Normal station service transformer A supplies power to normal 4.16 kV buses 34 A and B. Normal station service transformer B supplies power to normal 6.9 kV buses 35 A, B, C, and D.

The normal station service transformers have the capacity to supply normal auxiliaries and those emergency auxiliaries (both load groups) required during normal operation up to the full output of the main generator.

In the event of a unit trip (i.e. turbine, reactor, or generator trip), the generator breaker opens immediately (5 cycles), thus ensuring continuous power to buses 34 A and B and 35 A, B, C, and D via the normal offsite power source through the normal station service transformers (Section 8.1).

In the event of loss of the normal offsite power source, the alternate offsite power source supplies power through the reserve station service transformers from the 345 kV switchyard. Reserve station service transformer A supplies power to normal 4.16 kV buses 34 A and B via emergency buses 34C (Train A) and 34D (Train B), respectively. Reserve station service transformer B supplies power to normal 6.9 kV buses 35 A, B, C, and D. Upon loss of the normal offsite source of power, an automatic high-speed transfer (6 cycles) is initiated to the alternate offsite source, thus ensuring continuous power to buses 34 A and B and 35 A, B, C, and D.

Both the normal and alternate offsite power sources have the capacity to supply normal auxiliaries required for an orderly shutdown together with emergency auxiliaries (both load groups) required for a safe shutdown.

Both the normal and alternate offsite power sources have the capacity to provide unit startup power.

The normal 6.9 kV bus systems consist of four independent 6.9 kV buses. The normal 4.16 kV bus system consists of two independent 4.16 kV buses.

The four normal 6.9 kV buses 35A, 35B, 35C, and 35D are each rated 2,000 amp at 6.9 kV. The two normal 4.16 kV buses 34A and 34B are each rated 2,000 amp (with incoming sections rated 3,000 amps) at 4.16 kV.

Each of the normal 6.9 kV and 4.16 kV buses are housed in separate indoor metal-clad switchgear assemblies. The supply and feeder air

circuit breakers are electrically operated drawout types with stored energy mechanisms.

emergency generator, thereby providing a separate power source to each heat tracing circuit on each safety related line.

9. Generator Breaker - The generator breaker is not safety related. It automatically operates only on turbine, reactor, and generator trips. It can be manually operated from the control room. It is used to synchronize the main generator to the offsite system.

8.3.1.2 Analyses

8.3.1.2.1 Compliance Analysis

The preferred normal and alternate offsite circuits and the standby
onsite circuits satisfy GDC 17 and IEEE 308 (Table 8.1-2). The
offsite and onsite ac electrical power circuits permit functioning of
safety-related structures, systems, and components. In addition, the 430.8
offsite and onsite ac electrical power circuits have sufficient
independence to minimize the likelihood of their simultaneous failure
under operating and postulated accident and environmental conditions.

The Class IE power system satisfies GDC 17 and 18, IEEE 308, and Regulatory Guides 1.6 and 1.9 (Table 8.1-2).

The onsite ac electrical power systems provided permit functioning of
safety-related structures, systems, and components. In addition, the 430.8
onsite ac electrical power sources and the onsite ac electrical
system have sufficient independence, redundancy, and testability to
perform their safety functions assuming a single failure to meet the
requirements of General Design Criterion 17.

The ac electrical power systems important to safety are designed to permit periodic inspection and testing to assess continuity of the systems and the condition of their components, to meet the requirements of General Design Criterion 18.

8.3.1.2.2 Bus System Analysis

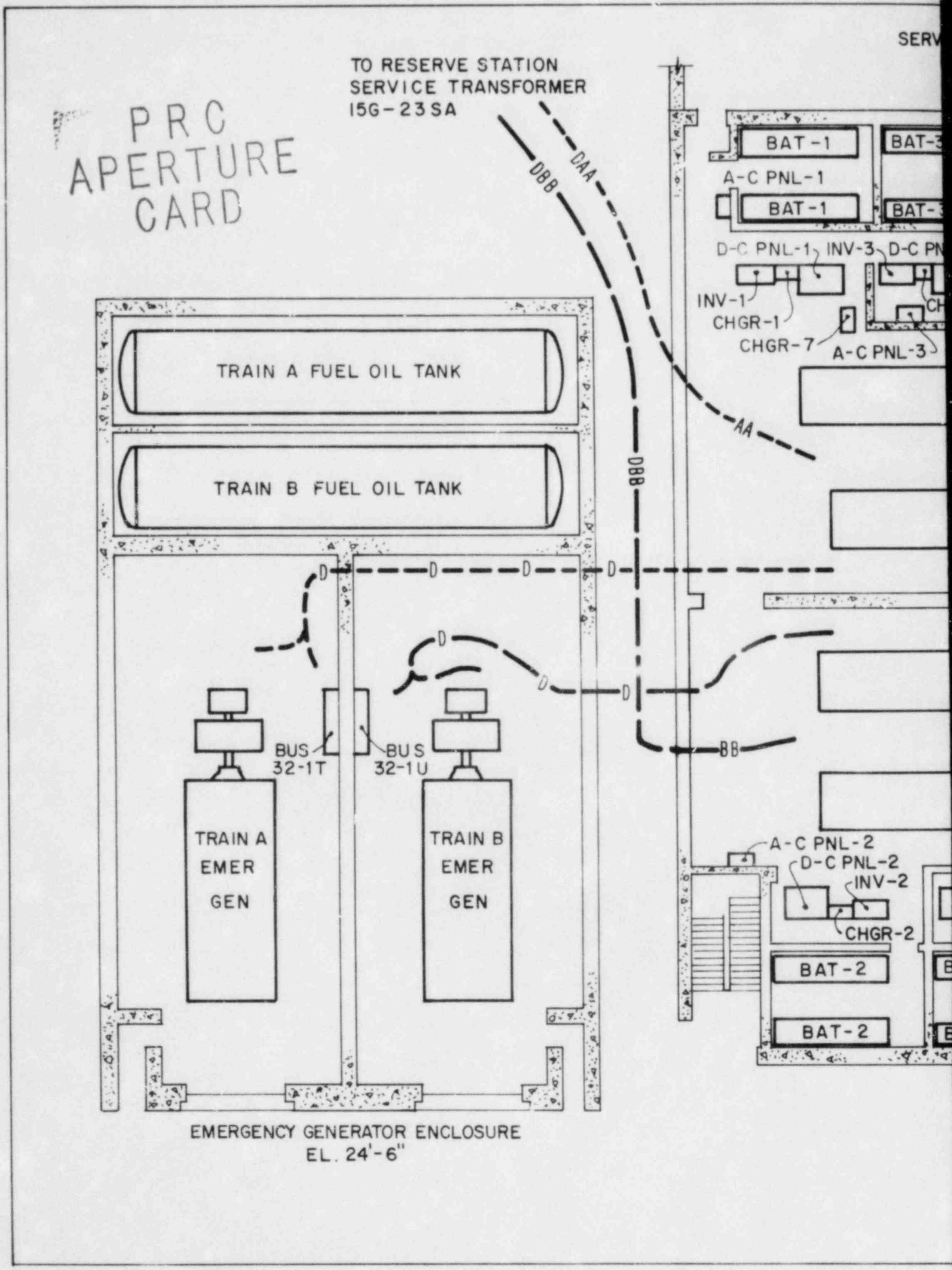
The emergency 4.16 kV buses 34C (Train A) and 34D (Train B) and associated emergency 480 V unit substations are designed to distribute the ac power required to safely shut down the reactor, maintain a safe-shutdown condition, and operate all auxiliaries required for safety under all normal, transient, and accident conditions. The design bases for the ac emergency buses are:

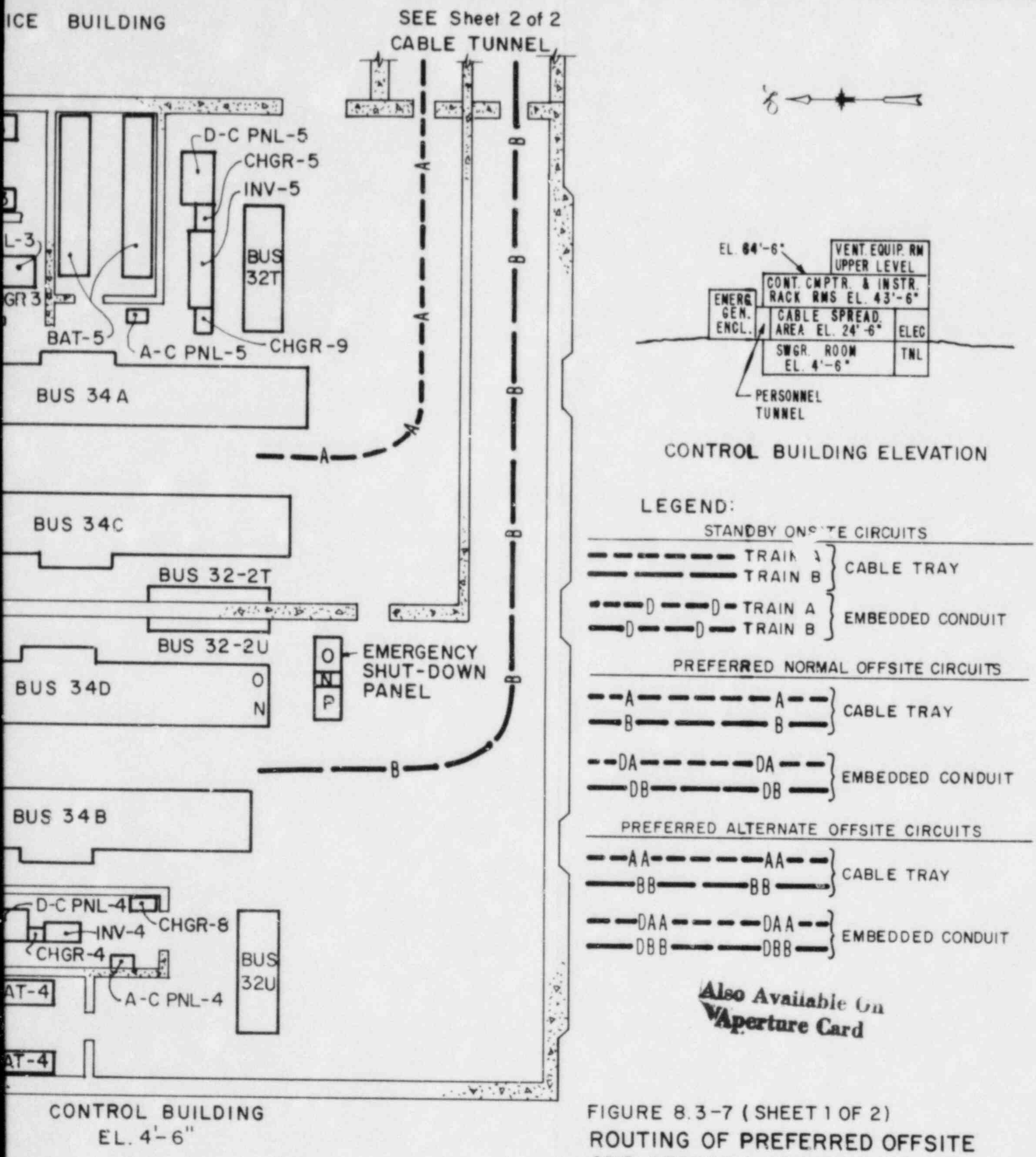
1. The emergency portion of the station service ac power system distributes power to all loads which are essential for safety (Table 8.3-1).

2. The normal and emergency portions of the ac power system are so arranged that a single failure will not prevent safety related systems from performing their intended safety functions (Figure 8.1-1).

3. The emergency 4.16 kV buses of the ac power system are arranged so that they can be supplied from either the normal or alternate offsite ac power sources or the respective onsite ac power source.

4. The emergency portion of the ac power system is designed in accordance with the IEEE 308 (Table 8.1-2).


TO RESERVE STATION
SERVICE TRANSFORMER
15G-23SA
SERV
CAA
DBB
AA
BB
D
BAT-1
BAT-3
A-C PNL-1
D-C PNL-1
INV-3
INV-1
CHGR-1
CHGR-7
A-C PNL-3
TRAIN A FUEL OIL TANK
TRAIN B FUEL OIL TANK
BUS
32-1T
BUS
32-1U
TRAIN A
EMER
GEN
TRAIN B
EMER
GEN
A-C PNL-2
D-C PNL-2
INV-2
CHGR-2
BAT-2
EMERGENCY GENERATOR ENCLOSURE
EL. 24'-6"



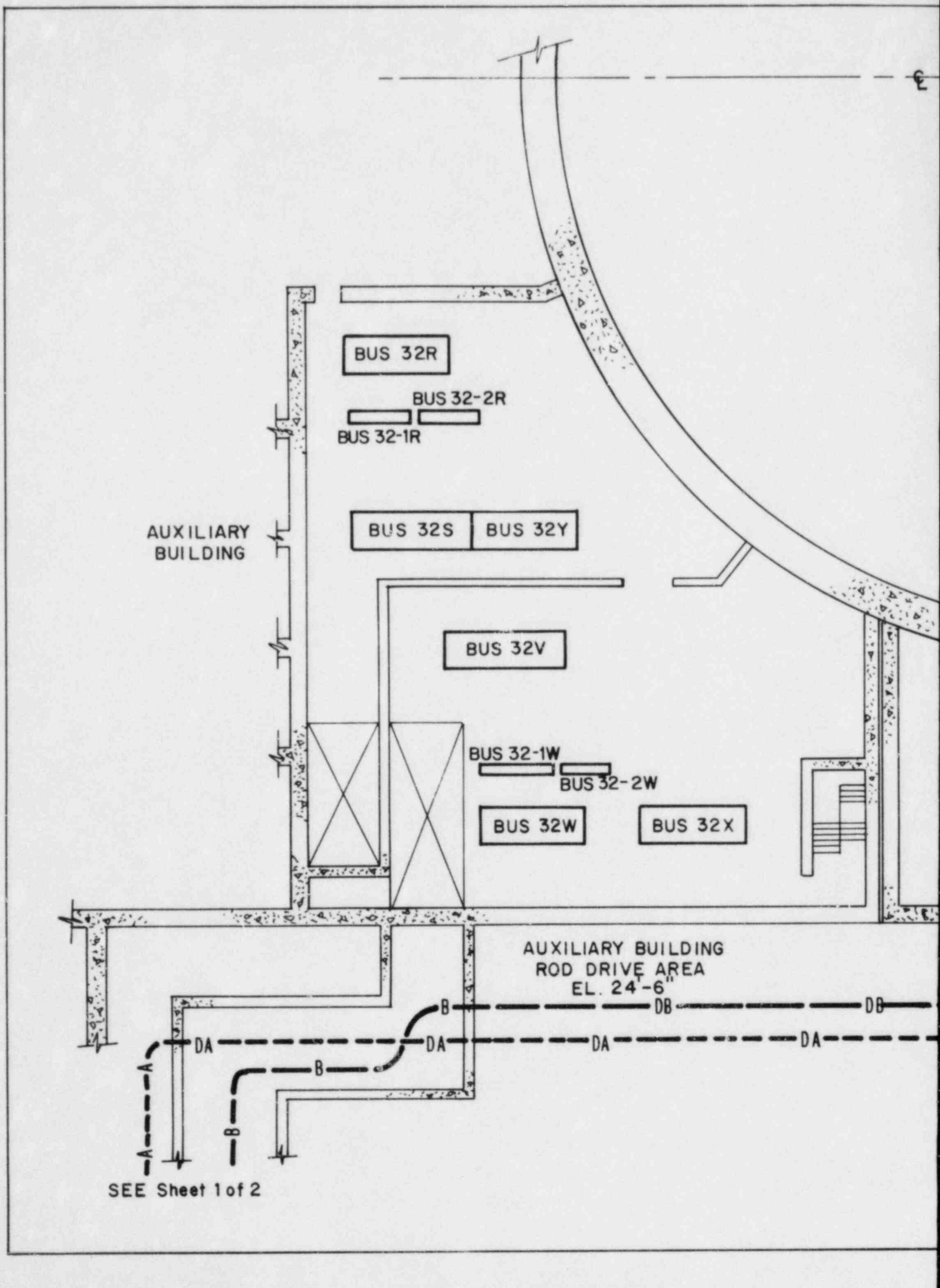
Also Available On Aperture Card

FIGURE 8.3-7 (SHEET 1 OF 2)

ROUTING OF PREFERRED OFFSITE AND STANDBY ONSITE CIRCUITS

MILLSTONE NUCLEAR POWER STATION UNIT 3

FINAL SAFETY ANALYSIS REPORT


BUS 32R
BUS 32-2R
BUS 32-1R
AUXILIARY BUILDING
BUS 32S
BUS 32Y
BUS 32V
BUS 32-1W
BUS 32-2W
BUS 32W
BUS 32X
AUXILIARY BUILDING ROD DRIVE AREA EL. 24'-6"
B
DB
DB
DA
DA
DA
DA
A
B
A
B
SEE Sheet 1 of 2

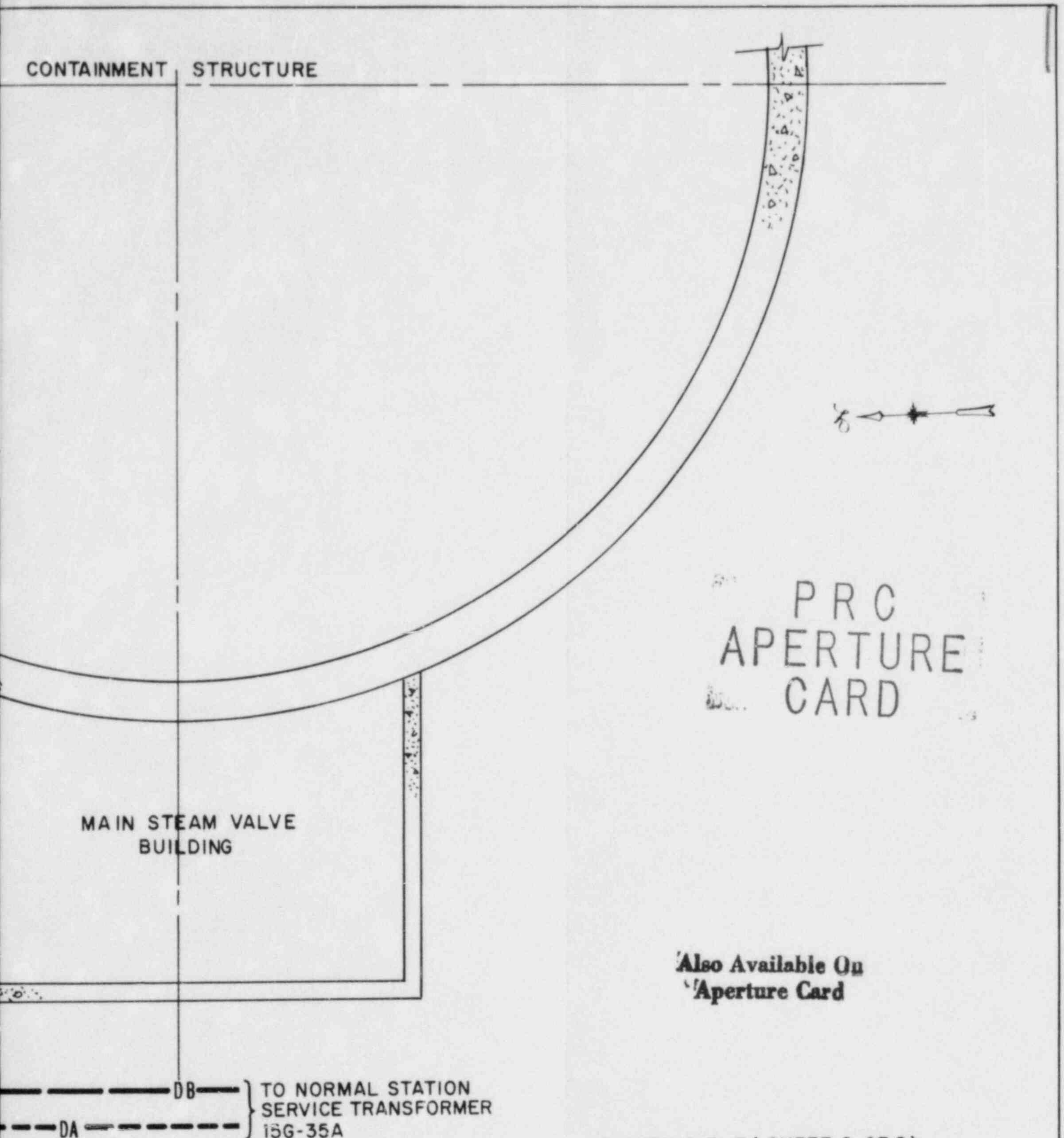


FIGURE 8.3-7 (SHEET 2 OF 2)
ROUTING OF PREFERRED OFFSITE AND STANDBY ONSITE CIRCUITS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: May 3, 1983

Question No. Q430.13 (SRP Section 8.3.1, Appendix 8A)

Section 8.3.1.1.3 of the FSAR indicates that diesel generator protective relaying is bypassed under accident condition in accordance with Branch Technical Position ICSB 17. Provide drawing reference numbers that describe the design of the bypass circuitry, the two-out-of-three logic circuitry, and relaying that is not bypassed under accident conditions.

Response:

This information is described in the following logic and elmentary sketches (Ref. FSAR 1.7) and manufacturer's drawings.

SWEC Logic	12179-LSK-24-9.2D
	12179-LSK-24-9.3J
SWEC Elementary	12179-ESK-8KD
	12179-ESK-8KG
	12179-ESK-5DS
	12179-ESK-5DR
	12179-ESK-7Q
	12179-ESK-7R

Manufacturer's Drawings:

Colt Drawing No. 11869137 (Sheets 1 through 4)
(SWEC File 2447.300-241-011 through 014

NRC Letter: May 3, 1983

Question No. Q430.14 (SRP Section 8.3.1)

Section 8.1.7 of the FSAR indicates that the diesel generator voltage (prior to connection of the first load block) may drop below the 75 percent minimum level permitted by position 4 of Regulatory Guide 1.9 (Revision 2). Provide justification for this exception to Regulatory Guide 1.9 and correct inconsistency between statements of compliance found in Sections 1.8 and 8.3.1.2.6 of the FSAR.

Response:

Refer to revised FSAR Table 1.8-1 and revised FSAR Section 8.3.1.2.6.

TABLE 1.8-1

NRC REGULATORY GUIDES

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.6	Independence between Redundant Standby (Onsite) Power Sources and Their Distribution Systems (Rev. 0, March 10, 1971)	Comply	8.3.1.4 8.3.2
1.7*	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev. 2, November 1978)	Comply	6.2.5
1.8	Personnel Selection and Training (Rev. 1-R, May 1977)	Comply	13.1.3 13.2 17.2
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies (Rev. 2, December 1979)**	Comply, with the following exceptions: The magnetizing inrush current due to the four 4,160-480V load center transformers may cause a momentary (3 to 5 cycles) dip in voltage prior to the first load block. This momentary voltage dip to levels outside that allowed by the Regulatory Guide for load sequencing is considered inconsequential to the successful loading of the standby generator unit. 430.14 C.11 Section 6.5 "Site Acceptance Testing" and Section 6.6 "Periodic Testing" of IEEE Std. 387-1977 should be supplemented by R.G. 1.108. The Millstone 3 position on R.G. 1.108 has several clarifications. (See R.G. 1.108).	8.3.1
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Rev. 1, January 2, 1973)	Withdrawn: Withdrawal of this guide does not alter any prior or existing licensing commitments based on its use. A position statement follows: 1. Reinforcing bars with a radius curve of 60 feet-0 inches or greater are tested at the sampling frequency specified in paragraph C4a. Reinforcing bars with a radius of curvature of less than 60 feet-0 inches are tested using only sister splices with the following frequency for each splicing crew:	3.8.1.6.2

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		One sister splice for the first 10 production splices. Four sister splices for the next 90 production splices. Three sister splices for the next and subsequent units of 100 production splices. If any sister splice used for tensile testing fails to equal or exceed 125 percent of the	

1. Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences

2. The reactor core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents

The selection of the emergency generator sets complies with Regulatory Guide 1.9 (Table 8.1-2).

The first load block in the generator loading sequence is made as large as practicable in order to minimize the length of the loading sequence.

430.14 The magnetizing inrush current due to the four 4,160-480 V load center transformers may cause a momentary (3 to 5 cycles) dip in voltage prior to the first load block. This momentary voltage dip to levels outside that allowed by the regulatory guide for load sequencing is considered inconsequential to the successful loading of the standby generator unit.

430.14 At no other time during the loading sequence does the voltage decrease below 75 percent of nominal voltage at the emergency generator.

The emergency ac power system consists of two completely redundant systems which are electrically and physically independent. Each emergency generator is capable of starting, operating, and carrying the required load continuously under postulated accident conditions.

The basic design criterion is that no single failure can prevent both emergency generator power systems from functioning. One emergency generator is adequate to supply power to all required emergency equipment. Surveillance instrumentation is provided in the control room to warn the operator during normal station operation of detectable inadequacies or failures which could lead to loss of function of the emergency generator and its power supply system. The emergency generator load is indicated in the control room. Information on load requirements of equipment that can be connected to the bus served by the emergency generator is contained in the equipment manual and operating procedures. Adequate information is available for the operator to make the proper decisions to keep from overloading the emergency generator. Components whose correct functioning can be verified only during operation of the emergency generator system are tested periodically. These tests demonstrate that no failures, which would prevent proper functioning, have occurred.

Assurance of the independence of the redundant onsite emergency power sources is obtained by means of the following:

1. Mechanically operated key interlocks are provided to prevent any two emergency generator buses from being operated in

parallel. The mechanically operated key interlocks are manually operated under strict administrative control.

2. Failure of an interlock which could enable an emergency generator bus to remain tied to a normal bus or to the offsite power source only results in a failure of that bus system and in no way affect the correct operation of the remaining onsite emergency power source. Should this event

NRC Letter: May 3, 1983

Question No. Q430.20 (SRP Section 8.3.1)

In accordance with Section 5.6.2.2 of IEEE Standard 387-1977, Section 5 of IEEE Standard 338-1977, and position C2a(8) of Regulatory Guide 1.108, it is the staff position that the diesel generator, when in the test mode and parallel with the offsite power system, be capable of responding to an accident signal. Describe how the Millstone design meets the staff position and justify areas of noncompliance.

Response:

The Millstone Design meets the staff position as follows:

An accident signal trips the emergency generator supply breaker, removes the governor from parallel operating mode and places it in the unit operating mode. The diesel generator continues to run and the emergency generator start signal is blocked by emergency generator speed greater than 300 rpm. If a subsequent loss of offsite power should occur, the connection and loading of the emergency generation is accomplished as described in FSAR Section 8.3.1.1.3.

NRC Letter: May 3, 1983

Question No. Q430.25 (SRP Sections 8.3.1 and 8.3.2)

In Section 1.8 of the FSAR, you identifed the following exception or clarification to Position C4 of Regulatory Guide 1.75:

> "Associated circuits are identified by the same color code as the Class 1E circuit with which they are associated. This color code exists up to and including an isolation device."

Position C.4 of Regulatory Guide 1.75 requires that associated circuits (up to and including an isolation device) be subject to all requirements placed on Class 1E circuits unless it can be demonstrated that the absence of such requirements cannot significantly reduce the availability of Class 1E circuits. The Applicant's clarification or exception implies that associated circuits meet only the color code requirements versus all requirements of Class 1E circuits. Provide justification for the implied exception to Position C4.

Response:

Refer to FSAR Section 1.8, revised Table 1.8-1, Compliance Position C.4 of Regulatory Guide 1.75.

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.75*	Physical Independence of Electric Systems (Rev. 2, September 1978)	Comply, with the following exceptions and clarifications: 1. General (Clarification) For separation purposes, location of cable entry/exit from cable tray is considered to be equivalent to perpendicular cable tray crossings. Ventilated tray covers are considered equivalent to solid tray covers. 2. Position C.1 The non-Class 1E pressurizer heaters, control rod drive mechanism cooling fans, and containment air recirculation fans connected to Class 1E power sources are provided with two separate Class 1E breakers connected in series. In addition, the interconnecting cables (i.e., from power source to load) are identified by the same color code as the Class 1E power source to which they are connected. Other non-Class 1E equipment connected to Class 1E power sources are provided with two separate Class 1E breakers or fuses connected in series. In addition, the interconnecting cables are identified by the same color code as the Class 1E power source to which they are connected (i.e., from power source to the load or up to and including the second breaker). Cable from the second breaker to the load are routed in rigid steel conduit. Coordination between the two series connected Class 1E breakers is not required. 430.36 3. Position C.4 (Clarification) Associated circuits are identified by the same color code as the Class 1E circuit with which they are associated. This color code exists up to and including an isolation device, except as discussed under Position C.1. Associated circuits meet all other requirements of Class 1E circuits up to and including the 430.25	7.1 8.3.1.4

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		isolation device. 4. Position C.6 (Clarification) Analyses of potential hazards in Section 5.1.1.1 of IEEE-384 are accomplished as follows: 1) The high pressure piping and missile analyses are described in FSAR Sections 3.6 and 3.5 respectively. 2) The fire protection analyses are outlined in FSAR Section 9.5.1. and the Fire Protection Evaluation Report. 3) Cable that is not flame retardant is enclosed in a dedicated raceway for the entire length of the run. 4) The building design for flooding is described in FSAR Section 3.4. 5. Position C.7 (Section 4.6 of IEEE-384) Minimum separation between Class 1E and non-Class 1E circuits are as specified in Sections 5.1.3, 5.1.4, or 5.6.2 of IEEE-384, except as discussed under Position C.16. Where plant arrangement in the control room, instrument rack room, or cable spreading room precludes the minimum vertical separation, but a spacing of 12 inches (i.e., the nominal vertical tray spacing) is maintained, a tray cover on the lower tray, a tray bottom on the upper tray, or a barrier interposed between the Class 1E and the non-Class 1E circuits provides the necessary separation. Where plant arrangement in the control room, instrument rack room, or cable spreading room, precludes the above referenced minimum 12 inches vertical separation or the minimum horizontal separation, either the non-Class 1E circuit(s) or the Class 1E circuit(s) are run in an enclosed raceway (i.e., conduit or tray with covers top and bottom to barrier is interposed between the non-Class 1E circuit(s) and Class 1E circuit(s). The minimum distance between cable and	

430.25

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		an enclosed raceway or cable and a barrier is one inch. 6. Position C.10 Class 1E cable and raceways shall be marked at intervals not exceeding 15 feet. The 5 foot requirement is a typographical error which has been confirmed by the NRC. 7. Position C.12 1) Power cables that supply power to instrument and control room distribution panels which must traverse the cable spreading room are enclosed in rigid steel conduit. 2) Power cable that traverses the cable spreading room are enclosed in rigid steel conduit. 3) The Millstone 3 design utilizes a single cable spreading room. 8. Position C.16 (Section 5.6.2 of IEEE-384) The minimum 6 inch separation (or a barrier) applies to spacing between exposed terminals, contacts, and equipment of redundant Class 1E circuits or Class 1E and non-Class 1E circuits for testing and maintenance purposes. A minimum of 1 inch separation (or a barrier) is required between redundant wire bundles or Class 1E and non-Class 1E wire bundles. The minimum of 1 inch separation is sufficient since the control boards are protected from and/or are not subject to hazards such as external fire, flooding, high energy piping, and missiles. Internal electrical fires are not considered a hazard due to fire retardant materials and low energy application.	
1.76	Design Basis Tornado for Nuclear Power Plants (Rev. 0, April 1974)	Comply	3.3.2.1
1.77*	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	Comply	15.4.2 15.4.7

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release (Rev. 0, June 1974)	Comply, with the following clarifications: The assumptions used for identifying chemicals potentially hazardous to the control room are in accordance with Regulatory Guide 1.78, dated June 1974. Chemicals not known or projected to be present within a 5 mile radius of the reactor facility are not considered in the evaluation and no specific design features are provided for the chemicals listed in Table C-1 of the Regulatory Guide. Hazardous chemicals that are known or projected to be used, transported, or stored within 5 miles of the reactor facility are considered in the evaluation of control room habitability. If the potential buildup of a specific hazardous chemical is slow, so that the time from detection to incapacitation is greater than 2 minutes, self-contained air breathing apparatus is provided. Redundant non-seismic detectors or human detecting are used as appropriate. If the potential buildup of a specific hazardous chemical exceeds the toxic limit, automatic detection and isolation, low leakage design features, and pressurization, if necessary, are provided to ensure that the control room remains habitable. In this case, specific design features are included: Quick-acting, redundant, non-seismic detectors, automatic control room ventilation isolation, coating of concrete and concrete block surfaces of control room with a suitable surface treatment to reduce leakage due to porosity, cracks, and construction joints in the control room. Sealing of all pipes, ducts, and electrical penetrations into the control room envelope. Compression seals for access doors and equipment removal hatches in the control room. In order to ensure control habitability for design basis accidents, the following are provided: Maintenance of 0.125 inches wg positive pressure. Two tight butterfly dampers in series in each intake. These features provide the control room operator	6.4.4.2

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		with the ability to isolate the control room. Hazards which could threaten control room habitability have been evaluated. The evaluation is consistent with the guide, and all hazards within 5 miles of the site are included. As a result of this evaluation, a commitment to provide chlorine detectors where needed has been made.	
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	Millstone 3 initial startup test program complies with Regulatory Guide 1.79 Rev. 1 with exceptions as stated in FSAR Section 14.2.7.	14.2
1.80	Preoperational Testing of Instrument Air Systems	Withdrawn	

NRC Letter: May 3, 1984

Question No. Q430.27 (SRP Sections 8.3.1 and 8.3.2)

In Section 1.8 of the FSAR, you imply taking exception to Section 5.1.4 of IEEE Standard 384-1974. Where plant arrangements preclude maintaining the minimum separation distance, you state that redundant circuits will be routed in nonsolid raceways. Solid versus nonsolid raceways are required by Section 5.1.4 of IEEE 384. Provide clarification and justification for noncompliance.

Response:

See FSAR Table 1.8-1 for the Millstone 3 Position on Regulatory Guide 1.75, under Position C.7. The NSSS position for Regulatory Guide 1.75 described in FSAR Table 1.8N-1 will be deleted since it is not applicable to Millstone 3.

TABLE 1.8N-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		restrictive requirement for shop fabricaion, where the welders' physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding closely supervised. For field application, the type of qualification should be considered on a case-by-case basis due to the great variety of circumstances encountered.	
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Conainment of Nuclear Power Plants (Rev. 0, January 1974)	The qualification programs for Westinghouse WRD supplied Class IE electric motor operators, solenoid valves, and limit switches described in WCAP-8587 and WCAP-9688 meet the requirements of Regulatory Guide 1.73.	3.11N 8.3.1
1.74	Quality Assurance Terms and Definitions (Rev. 0, February 1974)	The Westinghose position for the WRD NSSS scope of supply on Regulatory Guide 1.74 is presented in WCAP-8370, "WRD Quality Assurance Plant." The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, "NFD Quality Assurance Program Plan."	17.1.2 17.2
1.75	Physical Independ[illegible] of Electric Syst[illegible] (Rev. 2, September [illegible])	Westinghouse takes exception to the Regulatory Guide 1.75 in several areas as discussed below. These issues have been presented to the Regulatory Staff and are not resolved at this time. 1. Isolation Devices (Paragraph 3.8) Regulatory Position: Interrupting devices actuated by fault current are not isolation devices. Westinghouse Position: Interrupting devices actuated by fault current are isolation devices when justified by test or analysis. 430.29 2. Cable Spreading Area and Main Control Room (Paragraph 5.1.3) Regulatory Position: Places additional severe restrictions on equipment in area. 430.27 Westinghose Position: The IEEE draft criteria are adequate. 430.27	7.1 8.3.1.4

TABLE 1.8N-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		3. Instrument Cabinets (Paragraph 5.7) Regulatory Position: Separation requirements for instrument cabinets are the same as those for control boards. Westinghouse Position: Separation requirements should not be the same for instrumentation racks and control boards because functional requirements are different. The IEEE draft criteria are adequate.	
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	The result of the Westinghouse analysis shows compliance with the Regulatory Position given in Section C.1 of Regulatory Guide 1.77. In addition, Westinghouse complies with the intent of the assumptions given in Appendix A of the Regulatory Guide. However, Westinghouse takes exception to Position C.2, which implies that the Rod Ejection Accident should be considered as an emergency condition. Westinghouse considers this a faulted condition as stated in ANSI N18.2. Faulted condition stress limits will be applied for this accident.	15.4.2 15.4.7

NRC Letter: May 3, 1983

Question No. Q430.28 (SRP Sections 8.3.1 and 8.3.2)

In Section 1.8 of the FSAR, you imply taking exception to Position C12 of Regulatory Guide 1.75. Position C12 indicates that:

1. Power supply feeders to instrument and control room distribution installed in enclosed raceways should not be considered acceptable

2. Traversing power circuits separated from other circuits in the cable spreading area by a minimum distance of 3 feet and barriers should not be considered acceptable

3. Traversing power circuits routed in imbedded conduit which in effect removes them from the cable spreading area should be considered acceptable.

Power circuits that traverse the cable spreading area at Millstone are installed in enclosed raceways (rigid steel conduits). In accordance with Position C12 of Regulatory Guide 1.75, the routing should not be considered acceptable. Justify the adequacy of the proposed routing in steel conduit.

Response:

The degree of separation required between power circuits and other circuits varies with the hazards present at any given location. The cable spreading room is a protected area and is not subject to external energetic events such as flood, high energy pipe rupture, missles, etc. Potential electrical fires caused by fault current in the power cables are not considered to be a hazard as such fires, if possible, would be contained in the rigid steel conduit.

NRC Letter: May 3, 1983

Question No. Q430.30 (SRP Sections 8.3.1 and 8.3.2)

In Section 1.8 of the FSAR, you take exception to position C16 of Regulatory Guide 1.75 and Section 5.6.2 of IEEE Standard 384-1974. Minimum separation between redundant Class IE cables or between Class IE, and non-Class IE cables is identified to be 1 inch versus 6 inches inside control switchboards and instrument cabinets. Provide the analysis that demonstrates the adequacy of 1 inch minimum separation.

Response:

The degree of separation required between redundant Class 1E cables or between Class 1E and non-Class 1E cables varies with the hazards present at any given lcation. The control switchboards and instrument cabinets are located in a protected area and not subject to external energetic events such as flood, high energy pipe rupture, missiles, etc. Electrical forms caused by fault current within the control switchboards and instrument cabinets and not considered to be a hazard due to the use of fire retardant materials and low energy cables. The 1 inch is justified because it will prevent interaction between wire bundles due to the hazards of electrical potentials or heated wire bundles caused by electrical faults.

NRC Letter: May 3, 1983

Question No. Q430.31 (SRP Sections 8.3.1 and 8.3.2)

A third or spare charging pump may be connected to either Class 1E bus 34C or 34D. Describe the interlocks that preclude two charging pumps from being powered from the same Class 1E bus and preclude redundant buses from being tied together. Provide a similar description for the third or spare reactor plant component cooling pump.

Response:

Refer to revised FSAR Section 8.3.1.1.2 and revised FSAR Figures 8.3-4 and 8.3-5 for the response to this question.

closed after synchronization, and the emergency diesel generators returned to standby condition. The emergency bus supply breakers have manual synchronizing capability only. All of the above functions are performed from the control room.

The two emergency 4.16 kV buses constitute two redundant and independent power supplies, each supplying power to redundant safety related loads. All safety related loads are fed from the respective color coded emergency buses.

Power for running the third charging pump is supplied from either emergency bus 34C (Train A) or 34D (Train B) (Figures 8.1-1 and 8.3-4). A manually operated transfer switch is provided to connect the pump to the selected bus. A feeder breaker cubicle is provided on each bus for connection to the transfer switch. Normally, these breaker cubicles do not have a breaker installed. Upon failure of or due to required maintenance of one of the two connected charging pumps, its circuit breaker will be removed from its switchgear cubicle and installed in the switchgear cubicle in the same bus for the third pump. Interlocks are provided to ensure that only one of the third pump power sources can be energized at any one time. Thus, at no time can the two emergency buses be tied together. Additional interlocks are provided to ensure that only one pump on an emergency bus can be energized at any one time.

430.31

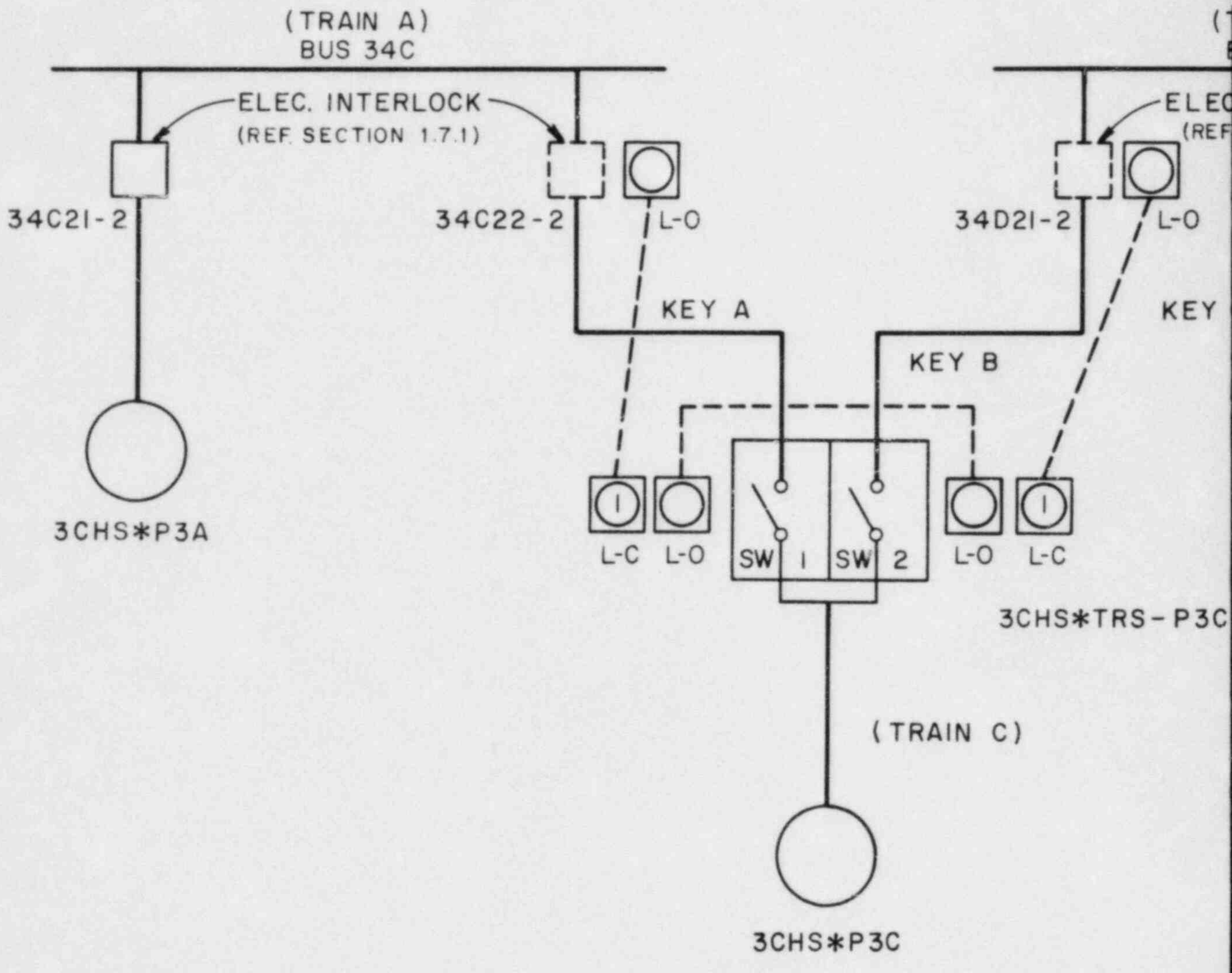
430.31

Power for running the third reactor plant component cooling water pump is supplied from either emergency bus 34C (Train A) or 34D (Train B) (Figures 8.1-1 and 8.3-5). A manually-operated transfer switch is provided to connect the pump to the selected bus. A feeder breaker cubicle is provided on each bus for connection to the transfer switch. Normally, the breaker cubicle in bus 34D will have a breaker installed; however, the breaker cubicle in bus 34C will not have a breaker installed. Upon failure of, or during required maintenance of the Train A connected reactor plant component cooling water pump, the breaker installed in bus 34D must be removed and installed in bus 34C breaker cubicle. Interlocks are provided to ensure that only one of the third pump power sources can be energized at any one time. Thus, at no time can the two emergency buses be tied together. Additional interlocks are provided to ensure that only one pump on an emergency bus can be energized at any one time.

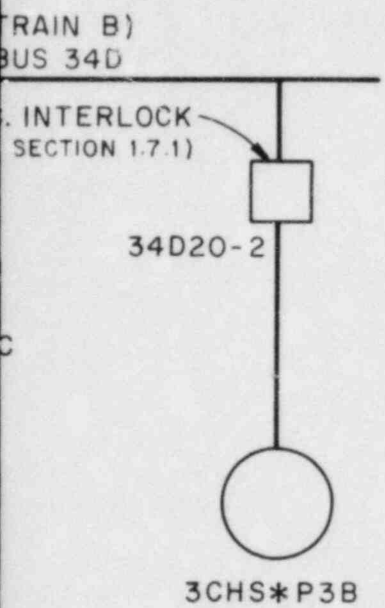
430.31

Each of the redundant emergency 4.16 kV buses is housed in a separate indoor metal-clad switchgear assembly located in separate rooms within a Seismic Category I and tornado missile protected structure. The supply and feeder air-magnetic circuit breakers are of the electrically operated drawout type with stored energy mechanisms. Switchgear breaker control power is supplied from their

respective color coded batteries of the Class 1E dc power system (Section 8.3.2). These buses are physically and electrically separated so that any credible event which



(TRAIN A)
BUS 34C
ELEC. INTERLOCK
(REF. SECTION 1.7.1)
34C21-2
34C22-2
L-O
34D21-2
L-O
KEY A
KEY B
KEY
3CHS*P3A
L-C
L-O
SW 1
SW 2
L-O
L-C
3CHS*TRS-P3C
(TRAIN C)
3CHS*P3C



LEGEND

- BREAKER
- BREAKER CUBICLE
- KEY REMOVED FROM DEVICE IN POSITION SHOWN
- KEY HELD IN LOCK IN DEVICE IN POSITION SHOWN

L-O LOCKED OPEN, KEY REMOVABLE WHEN OPEN

L-C LOCKED CLOSED, KEY REMOVABLE WHEN CLOSED

BREAKER 34C21-2 AND 34C22-2 ELECTRICALLY INTERLOCKED SUCH THAT BOTH CANNOT BE CLOSED AT THE SAME TIME (REFER TO DWG. NO. 12179-ESK-5CS AND CU REFERENCED IN TABLE 1.7-1)

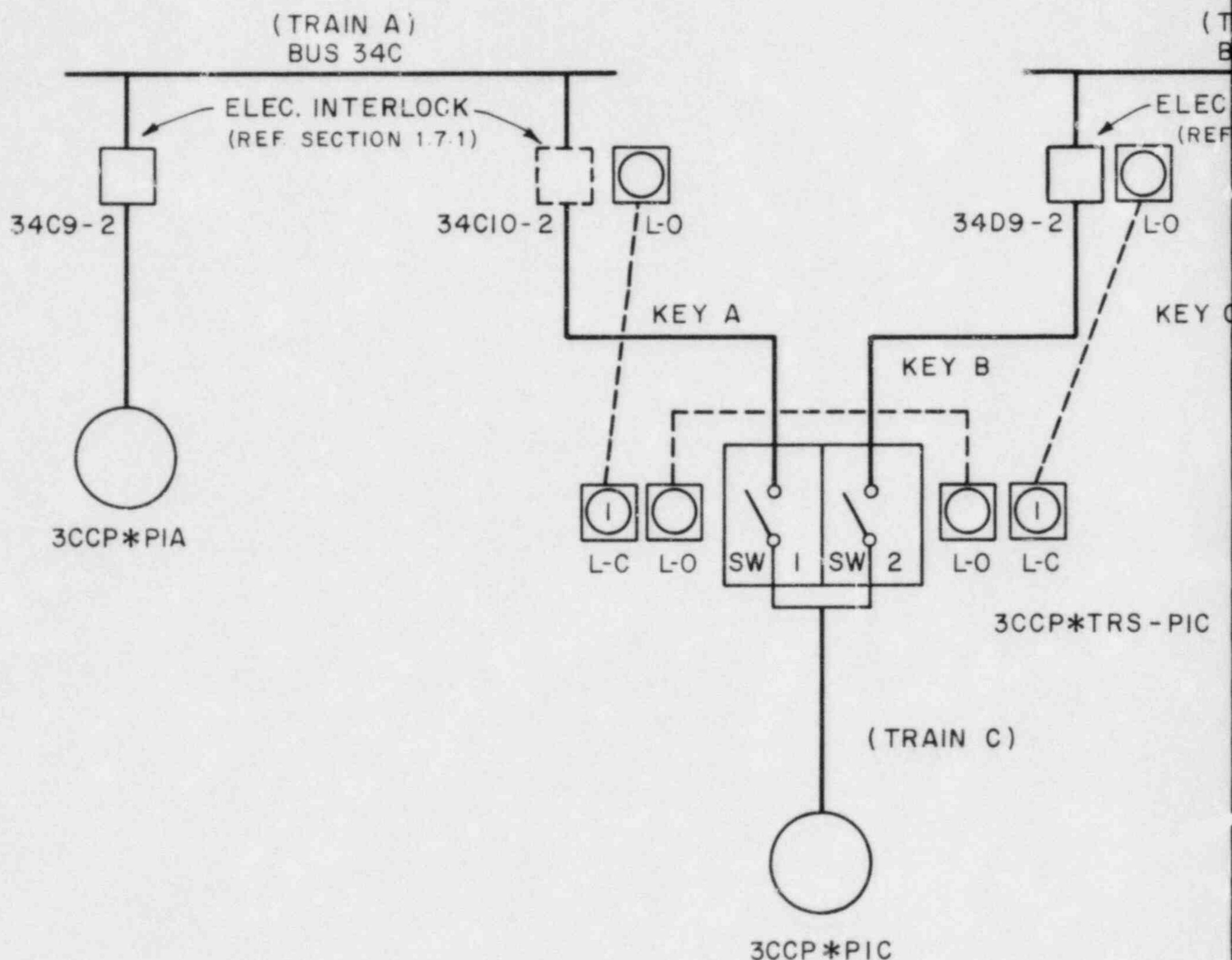
BREAKER 34D20-2 AND 34D21-2 ELECTRICALLY INTERLOCKED SUCH THAT BOTH CANNOT BE CLOSED AT THE SAME TIME (REFER TO DWG. NO. 12179-ESK-5CT AND CV REFERENCED IN TABLE 1.7-1)

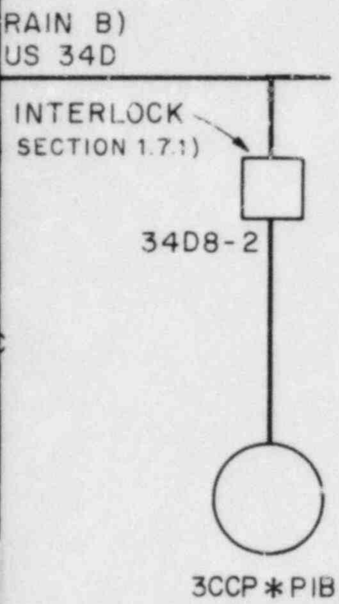
KEY A HELD CAPTIVE IN SW 1 WHEN OPEN; HELD CAPTIVE IN BREAKER 34C22-2 WHEN CLOSED

KEY B HELD CAPTIVE IN EITHER SW 1 OR SW 2 WHEN CLOSED

KEY C HELD CAPTIVE IN SW 2 WHEN OPEN; HELD CAPTIVE IN BREAKER 34D21-2 WHEN CLOSED

FIGURE 8.3-4
POWER SUPPLY
THIRD CHARGING PUMP
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT


(TRAIN A)
BUS 34C
ELEC. INTERLOCK
(REF. SECTION 1.7.1)
34C9-2
34C10-2
L-O
34D9-2
L-O
KEY A
KEY B
L-C
L-O
SW 1
SW 2
L-O
L-C
3CCP*P1A
3CCP*TRS-P1C
(TRAIN C)
3CCP*P1C



LEGEND

BREAKER

BREAKER CUBICLE

KEY REMOVED FROM DEVICE IN POSITION SHOWN

KEY HELD IN LOCK IN DEVICE IN POSITION SHOWN

L-O LOCKED OPEN, KEY REMOVABLE WHEN OPEN

L-C LOCKED CLOSED, KEY REMOVABLE WHEN CLOSED

BREAKER 34C9-2 AND 34C10-2 ELECTRICALLY INTERLOCKED SUCH THAT BOTH CANNOT BE CLOSED AT THE SAME TIME (REFER TO DWG. NO. 12179-ESK-5DA AND DC REFERENCED IN TABLE 1.7-1)

BREAKER 34D8-2 AND 34D9-2 ELECTRICALLY INTERLOCKED SUCH THAT BOTH CANNOT BE CLOSED AT THE SAME TIME (REFER TO DWG. NO. 12179-ESK-5DB AND DD REFERENCED IN TABLE 1.7-1)

KEY A HELD CAPTIVE IN SW 1 WHEN OPEN; HELD CAPTIVE IN BREAKER 34C10-2 WHEN CLOSED

KEY B HELD CAPTIVE IN EITHER SW 1 OR SW 2 WHEN CLOSED

KEY C HELD CAPTIVE IN SW 2 WHEN OPEN; HELD CAPTIVE IN BREAKER 34D9-2 WHEN CLOSED

FIGURE 8.3-5
POWER SUPPLY
THIRD REACTOR PLANT
COMPONENT COOLING PUMP
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: May 3, 1983

Question No. Q430.33 (SRP Sections 8.3.1 and 8.3.2)

The electrical 4.16 kV one line diagrams (No. 12179-1E-1K, 1L, 1M, and 1N) included in Section 1.7 of the FSAR indicate that there is an interconnection between Millstone Units 2 and 3 and between redundant buses 34C and 34D. Figure 8.1-3 and Section 8.0 of the FSAR indicate no interconnection between Units 2 and 3 or between buses 34C and 34D. Provide clarification and justify the interconnection.

Response:

There are no 4.16 kV connections between Millstone Units 2 and 3. There are no 4.16 kV connections between buses 34C and 34D (refer to response to Question No. 430.31). Section 1.7 will be revised to reflect current revision levels of the drawings listed.

NRC Letter: May 3, 1983

Question No. Q430.34 (SRP Sections 8.3.1 and 8.3.2)

Section 8.3.1.1.4(8) of the FSAR indicates that piping subject to freezing or boron precipitation are electrically heat traced. Two heat tracing circuits are provided for each pipe. One heat trace circuit is connected to Class 1E division A while the other circuit is connected through an isolation transformer to division B. Provide a description and justification of the physical and electrical independence between the two heat tracing circuits and between the redundant Class 1E divisions.

Response:

These heating circuits are powered and controlled from two separate heat tracing panels. Power to these panels are connected through isolation transformers to safety related Train A and B, (refer to revised FSAR Section 8.3.1.1.4). These transformers have been qualified and applied as "isolation devices" in accordance with the requirements of IEEE-STD 384 and Regulatory Guide 1.75. The physical and electrical independence of these circuits is assured as they are treated as "associated circuits".

Normally, one train (primary circuit) maintains temperature above its setpoint. Upon failure of the primary circuit, a backup circuit (opposite train) will maintain temperature above its setpoint.

ambient temperature at the motor location, and the hot spot temperature allowance. Large motors (above 200 hp) are generally provided with embedded resistance temperature detectors to monitor stator temperature.

7. Grounding - The design criteria for equipment grounding of safety related systems are:

   a. Equipment hardware, exposed surfaces, and potential induced-voltage hazards are adequately protected to ensure that no danger exists for plant personnel.

   b. A low impedance ground return path is provided to facilitate the operation of ground fault detection or protective devices in the event of ground fault or insulation failure on any electrical lead or circuit.

   All major electrical equipment, including motors of 100 hp and greater, is solidly grounded to the plant grounding grid.

   Intermediate and small size motors, including motors of 60 hp and below, and other electrical devices, such as motor operated valves solenoid operators and lighting fixtures, are grounded in one of two ways. Conduit connections are used as grounding ties to conduit-fed equipment. Other equipment is grounded to the plant grounding grid or to the building steel, which in turn is connected to the plant grounding grid.

   Cable tray, wireway, and metal conduit systems are grounded by copper cable connections to the ground grid or to building steel. Long runs of cable tray are grounded at each end and at intervals not exceeding 100 feet except as otherwise specifically indicated. All cable trays designated for power and control cable carry a No. 2/0 AWG copper ground cable connected to the tray at 50 foot intervals. Where expansion joints are used in tray or conduit runs, flexible copper cable jumpers are used. Metal conduit is grounded by connection to approved grounding bushings or by bolted connection to the tray system.

8. Heat Tracing - The majority of safety related lines and valves are located in heated areas and are not subject to freezing. All safety related lines or valves which are subject to freezing or boron precipitation are electrically heat traced and insulated. Each such line is electrically heat traced by two circuits, each of 100 percent capacity, with one designated as the normal circuit and the other as standby. On any safety related line that is heat traced, each normal and standby circuit is connected through isolation transformers to the Class IE Division "A"; 430.34 Division "B" bus respectively. In the event of a loss of normal ac power, each emergency bus is carried by its own

NRC Letter: May 3, 1983

Question No. Q430.37 (SRP Sections 8.3.1 and 8.3.2)

In Section 8.3.2.1.1 of the FSAR, you state that nonsafety 480 volt stub bus 32-3T (that supplies power to a number of nonsafety dc loads located in a nonseismic building) is powered from a Class 1E bus and is automatically shed upon loss of offsite power. It is the staff position that this stub bus should also be automatically shed upon an accident signal. Provide justification for noncompliance with this staff position.

Response:

Refer to revised FSAR Section 8.3.2.1.1 for the response to this question.

The source of power to each of the two normal 125 V buses is supplied from either its associated battery or its battery charger. The battery chargers are rated to supply normal operative power requirements plus recharge the batteries in 24 hours. All battery chargers are current limiting. The normal 125 V dc bus battery charger for battery 5 is powered from a Class IE emergency bus. Battery charger 5 meets all the requirements of an isolation device. The other normal 125 V dc bus battery charger for battery 6 is powered from a normal 480 V stub bus. The stub bus is powered from an emergency 480 V load center and is automatically shed upon an accident signal or loss of offsite power, but can be restored to service at the operator's discretion. 430.37

A spare swing battery charger is provided as a backup for the two operating battery chargers. This spare battery charger is connected to both buses through normally opened circuit breakers, which are key-interlocked to prevent inadvertent interconnection of both 125 V dc normal buses. The spare battery charger is powered from a normal 480 V ac bus and is located in the control building.

8.3.2.1.2 Class IE 125 V DC Power System

The Class IE 125 V dc power system is a safety related, two-wire, ungrounded bus system. This system is divided into four separate channels (Figure 8.3-3). Two channels are devoted exclusively to supplying the associated regulated 120 V ac vital bus power supply. The other two channels in addition to supplying the associated regulated 120 V ac vital bus loads, also supply other safety related dc loads (Table 8.3-4).

The Class IE 125 V dc power system equipment for each channel consists of one operating battery charger, one spare battery charger shared by two channels of the same train, one 125 V dc battery, and one distribution switchboard. On each of the two channels that also supply other safety related dc loads, additional distribution panels are included.

The batteries of each redundant channel are located in separate rooms in the control building at an elevation of 4 feet-6 inches (Figure 8.3-1). The battery chargers, spare battery charger, distribution switchboards, and distribution panels of each pair of channels are located in the separate emergency switchgear rooms of their associated power train. Barriers are provided between channels to maintain separation.

The four redundant channels are identified by color coding: Channel I (red), Channel II (white), Channel III (blue), and Channel IV (yellow). Equipment have color coded name plates inscribed with the equipment identification number. Cable trays have color coded stickers labelled with their cable tray identification number attached on the side rails of cable tray at intervals of 15 feet. The cables have color coded jackets and identification tags at the termination ends.

NRC Letter: May 3, 1983

Question No. Q430.38 (SRP Sections 8.3.1 and 8.3.2)

Non-Class 1E NSSS loads are connected to the Class 1E 120 V vital ac buses through transformers. You have stated that these transformers are qualified as isolation devices. Provide test results and/or analysis that demonstrates that any failure or combination of failures (including hot short) in the nonsafety circuits will not cause unacceptable influence on any Class 1E circuits. In addition, provide a description of the non-Class 1E load with respect to its size and the capacity and capability of the Class 1E system to supply the non-Class 1E load.

Response:

Testing was performed to demonstrate the adequacy of the transformers as isolation devices. This testing was performed with the station inverter as the power source for the isolation transformer. A short circuit was applied to the output of the isolation transformer. The failure criteria for this testing was either shutdown of the inverter, or unacceptable deviation from the specified inverter output requirements. The inverter exhibited no unacceptable deviation from required output and did not current-limit or shutdown. Moreover, the output loads of these isolating transformers are run in dedicated conduit, thereby precluding the possibility of a short to an external voltage source. The non-Class 1E loads are limited to control and instrument application only and are included in the design of the Class 1E System. The capacity and capability of the Class 1E System is discuss in Sections 8.3.1.1.2 and 8.3.1.1.3.

NRC Letter: May 3, 1983

Question No. Q430.48 (SRP Sections 8.3.1 and 8.3.2)

Describe how the Millstone design complies with the guidelines of NUREG-0737, Items II.E.3.1 and II.G.1, and justify areas of noncompliance.

Response:

These positions are clarified below:

II.E.3.1 Emergency Power Supply for Pressurizer Heaters

Position (1)

One bank of pressurizer backup heaters (PBH) is required to maintain natural circulation at hot standby (Section 5.4.10.3.6). One bank of PBH is normally connected to each safety related train.

Position (2 and 3)

One bank of PBH is required within 60 minutes to maintain natural circulation at hot standby (Section 5.4.10.3.6). Upon loss of power each emergency generator load sequencer permits manual loading of a PBH after 40 seconds onto its respective emergency generator (Table 8.3-1).

Position (4)

PBH connections to the safety related trains meet the requirements of Regulatory Guides 1.63 and 1.75 as discussed in the response to Question 430.55.

II.G.1 Emergency Power for Pressurizer Equipment

Position (1)

There are two power operated relief valves (PORV). PORVs are powered from redundant Class 1E.

Position (2)

There are two PORV block valves. Each PORV block valve is powered from the opposite safety related train from which its associated PORV is powered.

Position (3)

The PORVs and the PORV block valves are class 1E.

Position (4)

The pressurizer level indicators are powered from vital buses (Figure 8.3-3). The vital bus system is discussed in Section 8.3.1.1.2 and analyzed in Section 8.3.1.2.5.

NRC Letter: May 3, 1983

Question No. Q430.52 (SRP Sections 8.3.1 and 8.3.2)

Provide additional information regarding the power sources supplied to the RHR isolation valves. The staff's position is that a single failure of a power supply should not prevent isolation of the RHR when RCS pressure exceeds the design pressure of the RHR system. Additionally, loss of a single power supply should not result in the inability to initiate at least one 100 percent RHR train.

Response:

A single power supply failure will not prevent isolation of RHR or result in the inability to initiate at least one 100 percent RHR train. The description of RHR isolation valve arrangement and the power source for each RHR isolation valve can be found in FSAR Sections 5.4.7.1, 5.4.7.2.1, 5.4.7.2.4, 5.4.7.2.7, and 7.6.2.

NRC Letter: May 31, 1983

Question Q450.6 (SRP Section 15.6.5, Appendix A)

In FSAR Section 6.2.3.3 the SLCRS is discussed and it is stated, "The system starts on receipt of a SIS signal and becomes operative within 15 seconds". Then later in this Section it is stated "...within 60 seconds after the fan gets up to its design speed...". Based on this information, it may be concluded that the reactor enclosure building and contiguous buildings are drawn down to -0.25" water gauge pressure within 75 seconds. Please clarify the time for drawdown of the reactor enclosure building and contiguous buildings.

Response:

Refer to revised FSAR Section 6.2.3.3 for the response to this question.

Each filter bank includes a moisture separator, electric heater, prefilter, upstream HEPA filter, a charcoal adsorber, and downstream HEPA filter.

The charcoal adsorber is of gasketless nontray type and is designed for a 0.25 second residence time per 2 inches depth for gases at a flow velocity of 40 fpm. The actual depth of the absorber is 4 inches. 450.7

The SLCRS collects a portion of the primary containment leakage from the buildings contiguous to the containment, which house the various containment penetrations and the engineered safety features equipment circulating radioactive fluids, filters it, and releases it to atmosphere through the Millstone 1 stack. All leakages from the primary containment following a DBA flow into these areas. A portion of the auxiliary building atmosphere is exhausted via the auxiliary building ventilation system (see Section 9.4.3). In the auxiliary building, hydrogen recombiner building, and engineering safety features building, interior walls serve as the SLCRS boundary, thus separating areas contiguous to the containment from the remainder of these buildings. 450.7

All SLCRS boundaries are established by use of low leakage doors (weather stripped), sealed building joints, sealed piping, conduit cable and ductwork penetrations, and boundary isolation dampers for ventilation systems. Therefore, containment leakage is contained in these areas until filtered by the SLCRS.

6.2.3.3 Safety Evaluation

The SLCRS is not normally in operation. The system starts on receipt of a SIS signal, and is considered operative when the fan gets up to full speed which is 15 seconds after the accident. This includes 10 seconds for startup of the emergency generators. The capacity of each redundant system of 9,500 cfm, is sufficient to induce and maintain a negative pressure of 0.25 inch throughout the enclosure building and contiguous buildings within 45 seconds after the fan gets up to its designed speed, or 60 seconds after the accident assuming a wind velocity of 22 mph. This velocity is not exceeded 95 percent of the time, based upon onsite meteorological records. 450.6 450.6

To ensure protection from loss-of-function due to common events, the filter banks are physically separated, with a barrier (12 inch thick concrete slab) placed between them. Figure 6.2-46 provides indication of the failure position of all air-operated dampers in the SLCRS. The SLCRS is not specifically designed to remain functional following a high energy line break outside the primary containment.

A radiation monitor which monitors the air being processed by the SLCRS, is located downstream of the filter and warns the operator of a potential problem that will require operator action.

6.2.3.4 Inspection and Testing Requirements

All SLCRS components are tested and inspected as separate components and as an integrated system. Instrument readings are taken to ensure that all air systems are balanced to exhaust the required air quantities at design conditions.

Capacity and performance of fans conform to the required conditions and ratings and are in compliance with AMCA test codes and certified ratings program. SLCRS ductwork is leak-tested after installation to ensure against any bypass potentials. The ductwork is of all-welded construction and is pressure tested to 1.25 times the operating pressure. A thermal dioctylphthalate (DOP) smoke test with 0.3 micron smoke particle diameter at 100 percent and 20 percent rated filter air flow is given to each HEPA filter cell before leaving the manufacturer's facilities. A cold DOP test is conducted after filter installation at the site to ensure that there is no leakage from upstream to downstream of the HEPA filter. Provision is made to inject DOP at the inlet of the HEPA filter banks.

Each charcoal adsorber is field tested for leakage using a Refrigerant 11 and air mixture introduced upstream of the charcoal adsorber and a halogen detector of the gas chromatograph type to confirm that the bypass allowables are met. Filter banks are periodically tested for leakage while in place and defective cells are replaced and all leaks eliminated. Test canisters are installed downstream of the adsorber banks to be used for periodic laboratory testing and inspection of the adsorbent. New adsorbent is laboratory tested for acceptance in accordance with Regulatory Guide 1.52.

Fans, air-operated dampers, and controls are tested once a month by automatically starting on a simulated SIS signal and allowing them to reach rated speed with all dampers in the operating position before being shut down. The capability of SLCRS to achieve and maintain a negative pressure of 0.25 inch in the enclosure building and contiguous buildings will be verified by the test conducted as a part of the preoperational phase as described in Section 6.2.6. This test will be performed at intervals not greater than 18 months.

6.2.3.5 Instrumentation Requirements

The SLCRS is actuated on receipt of a SIS. Its logic is described in Section 7.3.

Differential pressure switches indicate pressure drop across each filter section locally and alarm high differential pressure remotely in the control room.

Each filter heater has two temperature switches for high temperature protection of the heater. One protection temperature switch is an automatic reset type while the other has a local manual reset feature. Heater ON and OFF indicator lights are located on the main heating and ventilation panel in the control room. The heater for

each filter bank is interlocked with the respective filter's exhaust fan to de-energize the heater when the fan is stopped.

Relative humidity is monitored upstream of each charcoal filter section and indicated locally.

The discharge air temperature of each charcoal filter section is continuously monitored. When discharge air temperature reaches 190°F, a local amber light at the fire detection panel is

NRC Letter: May 3, 1983

Question No. Q460.6 (Section 11.3)

Acceptance Criterion II.A.6 and Position C.2.c of Regulatory Guide 1.140 call for remote recorded indication of flow rates in normal ventilation systems. Section 11.3, 9.4, and 1.8 of the FSAR excludes flow rate instrumentation. In order to assure representative monitoring/sampling by the offline monitors on the normal ventilation systems, the staff recommends that flow rate instrumentation be provided in lieu of fan curves. Provide your justification for not including this instrumentation in the design.

Response:

Refer to revised FSAR Section 9.4.3.2 for the response to this question.

14. The auxiliary building ventilation system (ABVS) is nonnuclear safety related with the exception of: the building isolation dampers; charging pump, component cooling water pump, and heat exchanger ventilation system; MCC, rod control, and cable vault ventilation system; and the auxiliary building filtration units including fans and dampers all of which are Safety Class 3.

15. The auxiliary building ventilation system is actuated manually. The charging pump, component cooling water pump, and heat exchanger areas ventilation supply and exhaust dampers are actuated by the operation of the pumps. The auxiliary building ventilation isolation dampers are actuated by a safety injection signal (SIS).

9.4.3.2 System Description

The ABVS is comprised of the following subsystems:

1. Auxiliary building general area ventilation

2. Charging pump, component cooling water pump, and heat exchanger areas ventilation

3. MCC, rod control, and cable vault areas ventilation

4. electric cable tunnel area ventilation

The design parameters for the principal components of the auxiliary building ventilation system are given in Table 9.4-4.

The general area ventilation air supply portion includes two 50 percent capacity air handling units, each rated at 33,000 cfm. The air exhaust portion consists of two axial flow fans; one rated at 20,000 cfm and the other rated at 50,000 cfm.

Each air handling unit includes a prefilter, preheat coil, fan, and heating coil. The coils use hot water as a heating medium. Outside air is supplied continuously to all levels of the auxiliary building through ductwork.

The exhaust fans maintain the building at a negative pressure. One exhaust fan draws air from Elevation 66 feet-6 inches and Elevation 43 feet-6 inches, and the other draws air from Elevation 24 feet-6 inches and Elevation 4 feet-6 inches. The air flow path within the auxiliary building is from general areas with lesser potential for contamination to the cubicle area where a greater potential for contamination exists. Exhaust registers are mainly located within the cubicle areas. Once air is drawn from the building space, it is either discharged to the atmosphere through the ventilation vent or diverted to the auxiliary building filtration
460.6 | units, prior to release through the ventilation vent.

The plant ventilation vent is the release point for all ventilation exhaust air from the auxiliary, waste disposal, and fuel buildings, the containment structure and contaminated portions of the service building.

The ventilation vent effluent point of release is at Elevation 157 feet, 133 feet above site grade level, and the discharge velocity is approximately 2,500 fpm. The vent is located in the northeast corner of the turbine building auxiliary bay roof. A radioactive particulate and gaseous detection system is installed in the common duct to monitor effluent and provide visible and audible alarm locally and in the control room. Section 11.5 gives details of the process and effluent radiological monitoring systems. The total air flow through the ventilation vent is also measured and monitored by the RMS computer system. 460.6

Air samples are drawn from several points in the exhaust ductwork for radioactivity analysis upstream of the filtration units. High

Exhaust air is monitored by radiation monitors and high radiation is alarmed locally and in the control room. Air flow is monitored by the RMS computer system; indication is available through the RMS computer system CRT's. 460.6

Engineered safety feature status lights on the main control board indicate the status of the outlet and inlet dampers of the auxiliary building ventilation system. The dampers are closed automatically on receipt of a SIS or loss of power (LOP) signal.

The MCC, rod control, and cable vault air-conditioning air supply units have control switches and indicator lights located on the main heating and ventilation panel in the control room. One air-conditioning unit is normally running with the other in standby. The air-conditioning units are stopped automatically on receipt of a smoke detection signal or a carbon dioxide (CO_2) release signal. Service water booster pumps for each air-conditioning unit are started automatically on receipt of a LOP signal. The service water booster pumps have control switches and indicator lights on the main heating and ventilation panel in the control room.

The electric tunnel area exhaust fan has a control switch and indicator lights on the main heating and ventilation panel in the control room. The exhaust fan is automatically stopped and the exhaust dampers closed on receipt of a CO_2 release signal or a SIS. Engineered safety feature status lights on the main control board indicate when the dampers are closed.

9.4.4 Turbine Building Area Ventilation System

The turbine building area ventilation system, a nonnuclear safety related system, removes the heat dissipated from equipment, piping, and lighting to provide a comfortable environment for personnel and proper function of equipment, instrumentation, and control.

9.4.4.1 Design Bases

The turbine building area ventilation design is based on the following criteria:

1. During the summer, coincident with the outdoor air design temperature, the turbine building temperatures range from 95°F at operating floor to 104°F just below the roof. During the winter, the inside temperature is maintained at a minimum of 65°F.

2. The system is nonsafety related and is designated nonnuclear safety class (NNS).

3. Branch Technical Position ASB 9.5-1, Fire Protection for Nuclear Power Plants.

control board for each emergency ventilation supply fan to indicate when it is running. A bypass alarm is actuated whenever both Train A supply and exhaust fans and Train B supply and exhaust fans are not available for operation.

When the auxiliary feedwater pumps area emergency ventilation system is operating, outside air is used exclusively until the area temperature decreases to 50°F d.b. At this point, the supply, exhaust, and recirculation dampers modulate the incoming outside air with a portion of the recirculated area air to maintain a 50°F d.b area temperature. In addition to the damper position indicating lights on the main ventilation panel, ESF status lights are provided on the main control board to indicate that either the supply or exhaust ventilation damper and the recirculation damper are fully closed.

Temperature switches are provided in each ESF area which actuate an annunciator at the main ventilation panel whenever an area temperature exceeds a predetermined set point. The switches are also monitored by the plant computer.

The ESF building normal ventilation discharge is monitored for particulate and gaseous radiation. The radiation monitors are
460.6 discussed in Section 11.5.2.2. Air flow is measured and monitored by the RMS computer system.

Pressure differential indicators are provided for each inlet filter in the normal ventilation systems and in the inlet filters for self-contained air-conditioning units to monitor the filter condition. The inlet filter for the auxiliary feedwater pumps emergency ventilation system is equipped with a pressure differential switch which energizes an annunciator on the main ventilation panel as a result of high differential pressure. This switch is also monitored by the plant computer.

9.4.6 Emergency Generator Enclosure Ventilation System

The emergency generator enclosure has both safety related and nonsafety related ventilation systems (Figure 9.4-3) which provide a suitable environment for personnel and equipment operation.

9.4.6.1 Design Bases

The design bases for the emergency generator enclosure ventilation system are:

1. The ability to maintain an environment suitable for personnel and equipment and to remove the heat generated by the operation of the emergency generator

2. The outside air summer design dry bulb temperature for the Millstone Point site of 86°F, and the outside air winter design dry bulb temperature of 0°F

associated fans and dampers and one nonnuclear safety related 100-percent capacity exhaust fan. The normal exhaust system provides 41,360 cfm during normal plant operation, while the safety related exhaust system maintains the building at slightly negative pressure during fuel handling operation and accident conditions in relation to the above supply air quantities. The fuel building ventilation exhaust system serves all areas of the fuel building and is
460.6 discharged (unfiltered or filtered) through the radiation monitored ventilation vent.

460.6 The fuel building exhaust filters are described in Section 6.5.1.

The flow of the air within the fuel building is directed from areas of low potential for airborne contamination to areas of greater potential for airborne contamination. The unfiltered exhaust air, not subjected to contamination under normal operation conditions, is discharged through the monitored ventilation vent. (See
460.6 Section 9.4.3.)

The fuel building ventilation exhaust system is capable of detecting and controlling radioactive contamination. On receipt of (1) a containment isolation phase A (CIA) signal or (2) a high radioactivity signal from the particulate and gas monitor which
460.6 samples the ductwork exhaust, the unfiltered ventilation exhaust system is manually isolated by the closing of two safety related dampers located on the supply and exhaust sides of the unfiltered exhaust fan as shown on Figure 9.4-2. In addition, supply air is provided at reduced capacity (17,000 cfm) by the manual actuation of a nonnuclear safety related damper in the supply ductwork, then the exhaust air is manually diverted through one of the two fuel building filtration units. Operational control of the supply air assures the prevention of condensation of water vapor from the spent fuel pool and maintenance of a negative pressure in the fuel building to prevent uncontrolled release of radioactive contamination.

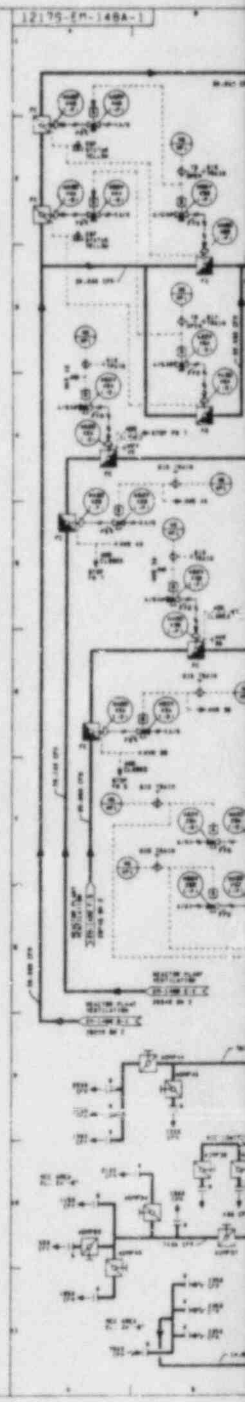
In the event of a failure in the nonnuclear safety related supply air system, the safety related backdraft dampers which are mounted in the fuel building exterior wall shall admit the required makeup air of 17,000 cfm.

During fuel handling, the exhaust air is manually diverted through at least one of the fuel building filtration units, in addition to reducing the supply air to 17,000 cfm, thus maintaining a negative pressure. During periods of high temperature and humidity, it may be necessary to use both fuel building filter exhaust fans to maintain proper atmospheric clarity in the spent fuel pool area. In each
410.21 case, the supply air is 34,000 cfm which is the maximum air flow obtained from a single air handling unit. This condition is maintained by providing manual isolation dampers to isolate the operating supply units from each other.

During fuel handling, the exhaust air is manually diverted through at least one of the fuel building filtration units, in addition to

P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-5 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	10.4-1 SH1
EM110B	10.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-35 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1



PRC
APERTURE
CARD

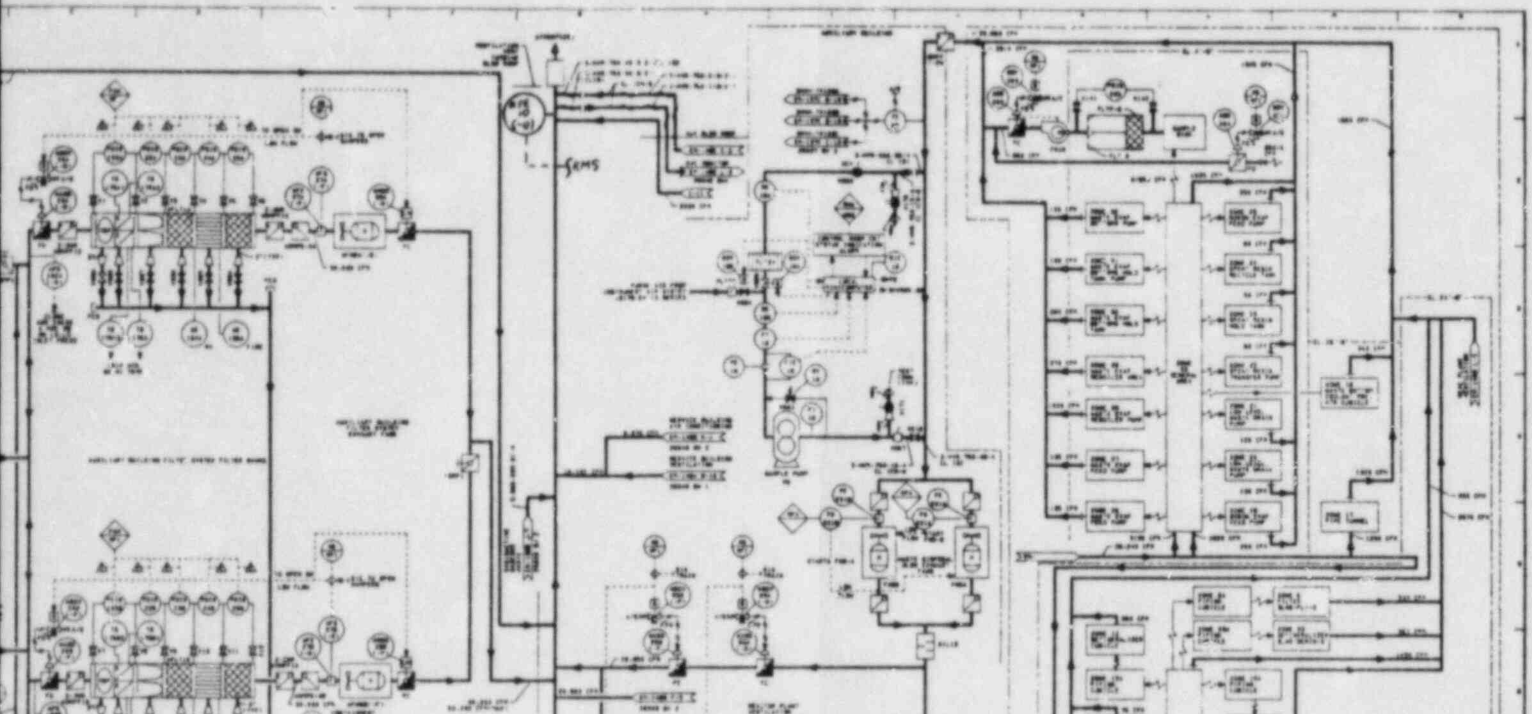


FIGURE 9.4-2 (1 OF 3)
REACTOR PLANT
VENTILATION SYSTEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: May 3, 1983

Question No. Q460.7 (Section 11.1)

Acceptance Criteria, Requirement II.a, calls for meeting the Positions in Regulatory Guide 1.110. Section 11A of the FSAR does not provide the cost of borrowed money. Provide the cost of borrowed money expressed in percent that was used in your cost-benefit analysis for Appendix I to 10CFR50.

Response:

The cost of borrowed money used in the cost-benefit analysis for Appendix I to 10CFR50 was 10 percent. Refer to revised Part III of FSAR Appendix 11A for the response to this question.

PART III - COST-BENEFIT ANALYSIS

This appendix presents the results of cost-benefit analyses performed in accordance with Section II. D of 10CFR50, Appendix I.

Augments to the liquid and gaseous effluent systems and respective potential reductions to the annual population exposure are taken from the U.S. NRC Regulatory Guide 1.110, Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Regulatory Guide 1.110, 1976). The beneficial savings of each augment were calculated by multiplying the calculated dose reduction by $1,000 per man-Rem or $1,000 per man-thyroid-Rem. The cost of borrowed money was conservatively assumed to be 10 percent. The equations and site specific data required for the dose calculations are presented in Part I of this appendix. 460.7

Augments to the Liquid Effluent Treatment System

Table 11A.3-1 presents the calculated base case annual total body dose (man-Rem) and thyroid dose (man-thyroid-Rem) associated with the operation of the plant liquid radwaste system for the population expected to live within an 80-km radius of the plant for the year 2010. Assuming that each augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum benefit to be derived from any augment would be $1,300 for reducing man-Rem exposures to zero and $6,500 for reducing man-thyroid-Rem exposure to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive liquid radwaste augment was found to be $19,000 per year for a plant located in the northeastern United States. Since the benefit from this augment would be less than the corresponding total annualized cost, the cost-benefit ratio is greater than 1. The operation of additional equipment for the purpose of reducing the annual population dose would not be cost effective. Therefore, the most cost-beneficial system has been included in the current plant design.

Augments to the Gaseous Effluent Treatment System

Table 11A.3-2 presents the calculated base case annual total body man-Rem and thyroid man-Rem associated with the operation of the gaseous radwaste system for the 80-km radius population.

Assuming that each augment is capable of reducing the population doses to zero, the maximum benefit to be derived from any augment would be $4,700 for reducing man-Rem exposures to zero and $7,200 for reducing man-thyroid-Rem exposures to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive gaseous radwaste augment was found to be $8,700 per year for a plant located in the northeastern United States. Since the benefit from this

augment would be less than the corresponding total annualized cost, the cost-benefit ratio is greater than 1. The operation of additional equipment for the purpose of reducing the annual population dose would not be cost effective. Therefore, the most cost-beneficial system has been included in the current plant design.

Reference for Appendix 11A Part III - Cost-Benefit Analysis

Regulatory Guide 1.110, 1976. Cost-Benefit Analysis for Radwaste Systems for Light Water-Cooled Nuclear Power Reactor. March 1976.

NRC Letter: May 3, 1983

Question No. Q460.8 (Section 11.1)

Acceptance Criteria, Requirement II.b, calls for meeting the Positions in Regulatory Guide 1.112. Section 11.3.3 of the FSAR states that the release points are indicated on Figure 1.2-1. Clarify if this should be Figure 1.2-2. Compare Figure 1.2-2 with Figure 2.1-5 of the ER. Provide the information requested in Appendix B to Regulatory Guide 1.112, item 6d, for the height and location relative to adjacent structures. Include the turbine building, warehouse, steam relief vents, and outside tank vents, for example. Provide some detail on how the release lines lead to the stack and the reactor plant vent.

Response:

Release points including locations and heights are indicated on revised FSAR Figure 1.2-1 and Figure 1.2-2. Steam vents and turbine building exhausters are shown on revised FSAR Figure 3.8-73. Radioactive tanks are connected to the reactor plant aerated vent system, as indicated in FSAR Section 9.3.3. Air is taken into tanks through the tank vent, and carried out of the tanks by the aerated vent system to the radioactive gaseous waste system (FSAR Section 11.3.3). Ventilation release for the warehouse is discussed in the response to Question 460.15. Information on release lines to the ventilation vent and the Millstone stack is provided in revised FSAR Section 11.3.3.

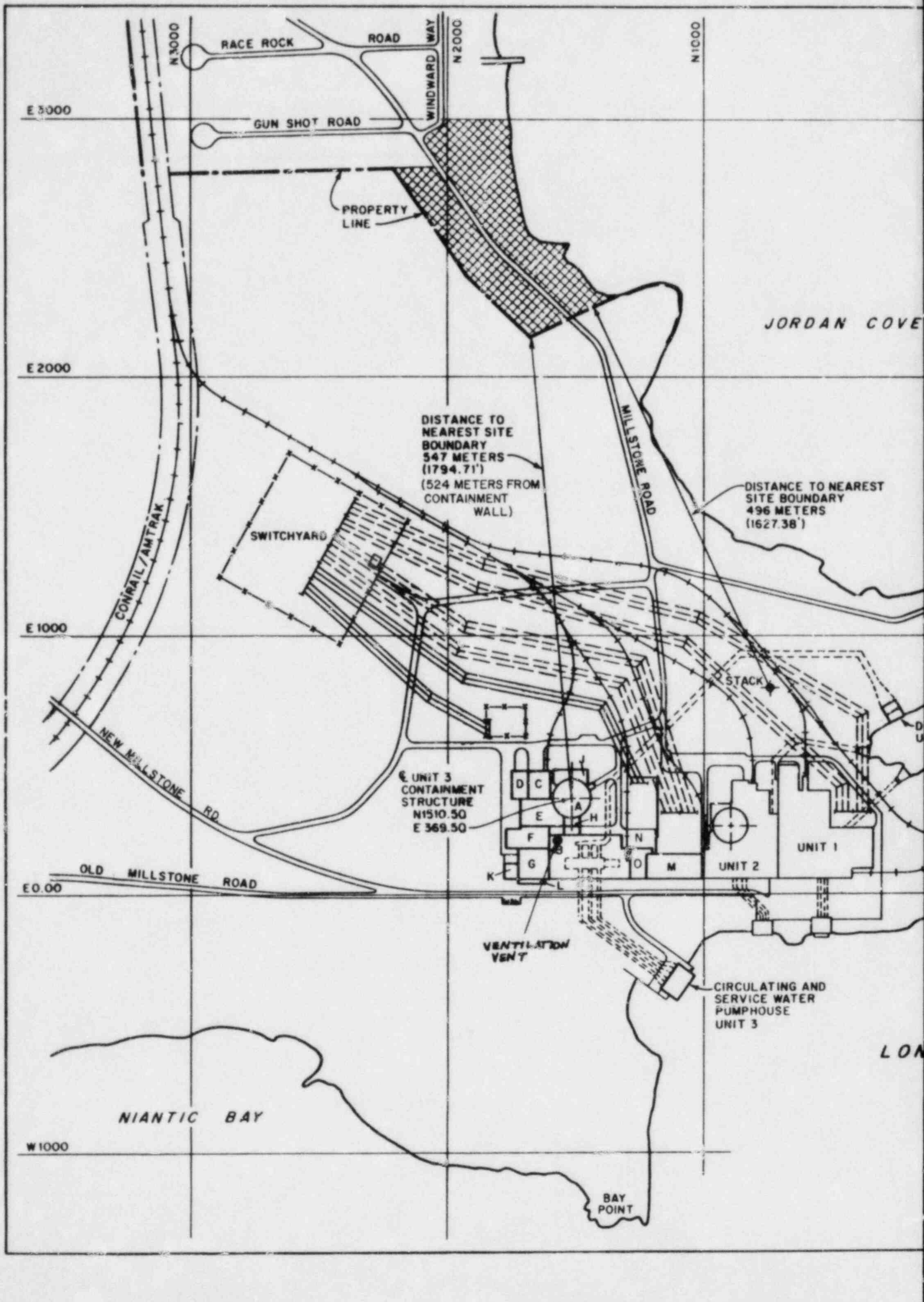
460.8 Release points are identified on Figures 1.2-1 and 1.2-2.

Exhausts from the auxiliary building, the fuel building, the waste disposal building, the containment purge, the service building, and the gaseous waste process vent are released from the reactor plant ventilation vent. This vent is located on the turbine building 133 feet above grade and 157 feet-0 inches above sea level. The base elevation is 75 feet above grade. The square vent cross-sectional dimensions are 10 feet by 10 feet and the discharge velocity is 3,000 feet per minute. The maximum discharge temperature is 104°F. Containment purge is considered an intermittent release. Others are assummed as continuous.

The turbine building ventilation system exhausts through the turbine building roof exhausters, located on the turbine building roof. The exhausters are approximately 114 feet above grade, 138 feet above sea level, and have a base elevation of 107 feet above grade. The system exhausts through a roof mounted mushroom-type hood with dimensions of 14 feet-6 inches by 14 feet-6 inches. The discharge velocity is 1,140 feet per minute and the maximum temperature is 104°F. These releases are assumed to be continuous releases.

The reactor plant aerated vents, the reactor plant gaseous vents,
condenser air removal effluent, radioactive gaseous waste system
discharges, steam generator blowdown tank condenser vent, and
containment vacuum pump discharge are released through the
Millstone 1 stack. The Millstone 1 stack is 375 feet above grade,
389 feet-6 inches above sea level, and has a circular orifice with an
8 foot inside diameter. The stack discharges at a velocity of
5,630 feet per minute at a maximum temperature of 104°F. These
460.8 discharges are assumed as continuous releases. The portion of the
discharge vent between the ESF building and the stack is underground.

The engineered safety features building ventilation system exhausts
through the ESF building vent located on the south wall. The vent is
460.8 23 feet-2 inches above grade, 47 feet-2 inches above sea level. The
vent cross sectional dimensions are 3 feet-8 inches by 7 feet-
8 inches and the discharge velocity is 570 feet per minute. The
maximum discharge temperature is 104°F. The release from the ESF
building is intermittent, considered to occur only during shutdown
periods when the residual heat removal system is in operation. For
calculation purposes, releases from the ESF building are included as
part of the continuous release from the auxiliary building in
Table 11.3-5.


RACE ROCK
ROAD
WINDWARD WAY
N3000
N2000
N1000
E3000
GUN SHOT ROAD
PROPERTY LINE
JORDAN COVE
E2000
DISTANCE TO NEAREST SITE BOUNDARY 547 METERS (1794.71')
(524 METERS FROM CONTAINMENT WALL)
MILLSTONE ROAD
DISTANCE TO NEAREST SITE BOUNDARY 496 METERS (1627.38')
SWITCHYARD
CONRAIL/AMTRAK
E1000
STACK
NEW MILLSTONE RD
℄ UNIT 3 CONTAINMENT STRUCTURE N1510.50 E 369.50
UNIT 1
UNIT 2
OLD MILLSTONE ROAD
E0.00
VENTILATION VENT
CIRCULATING AND SERVICE WATER PUMPHOUSE UNIT 3
NIANTIC BAY
W1000
BAY POINT

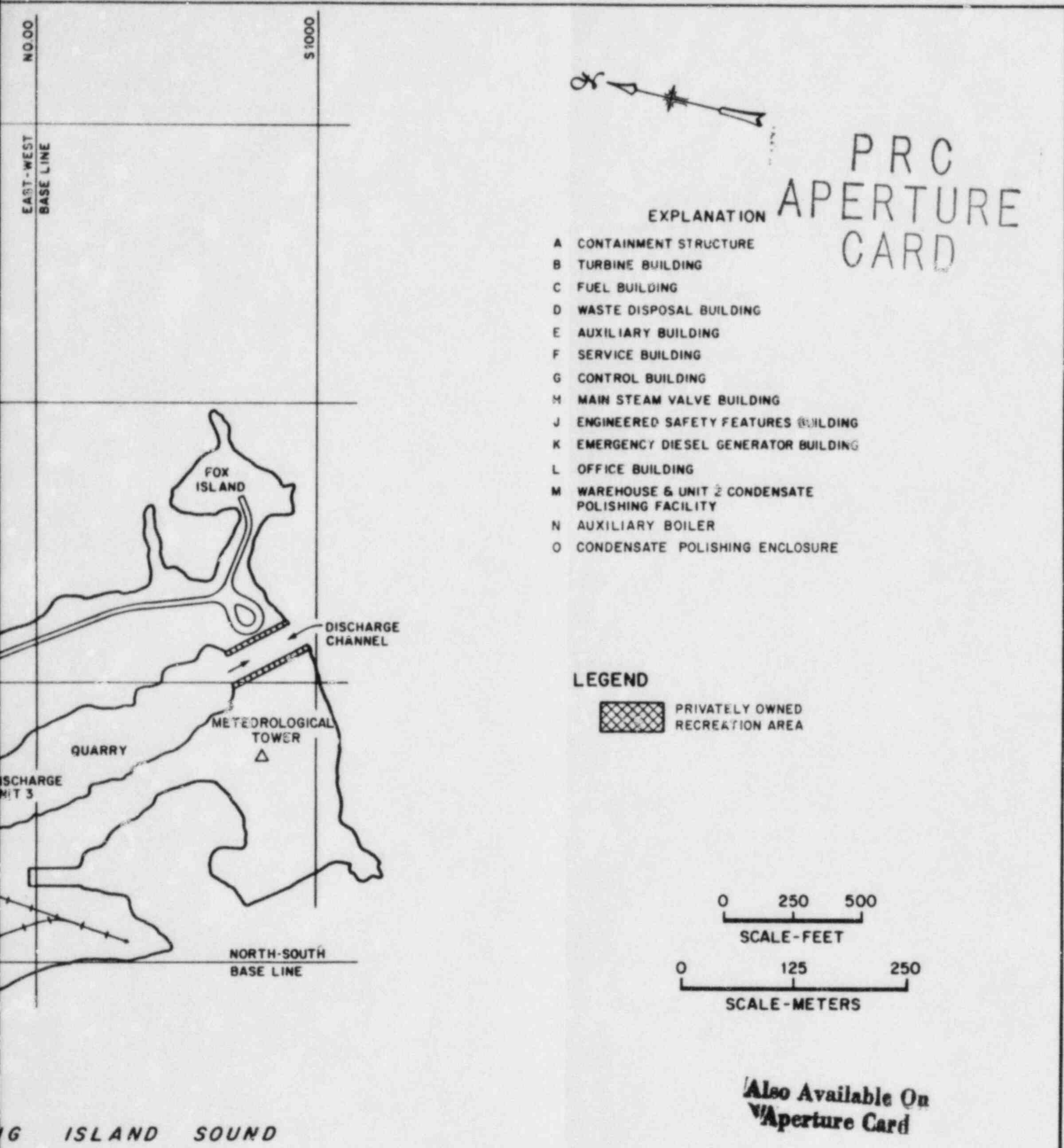
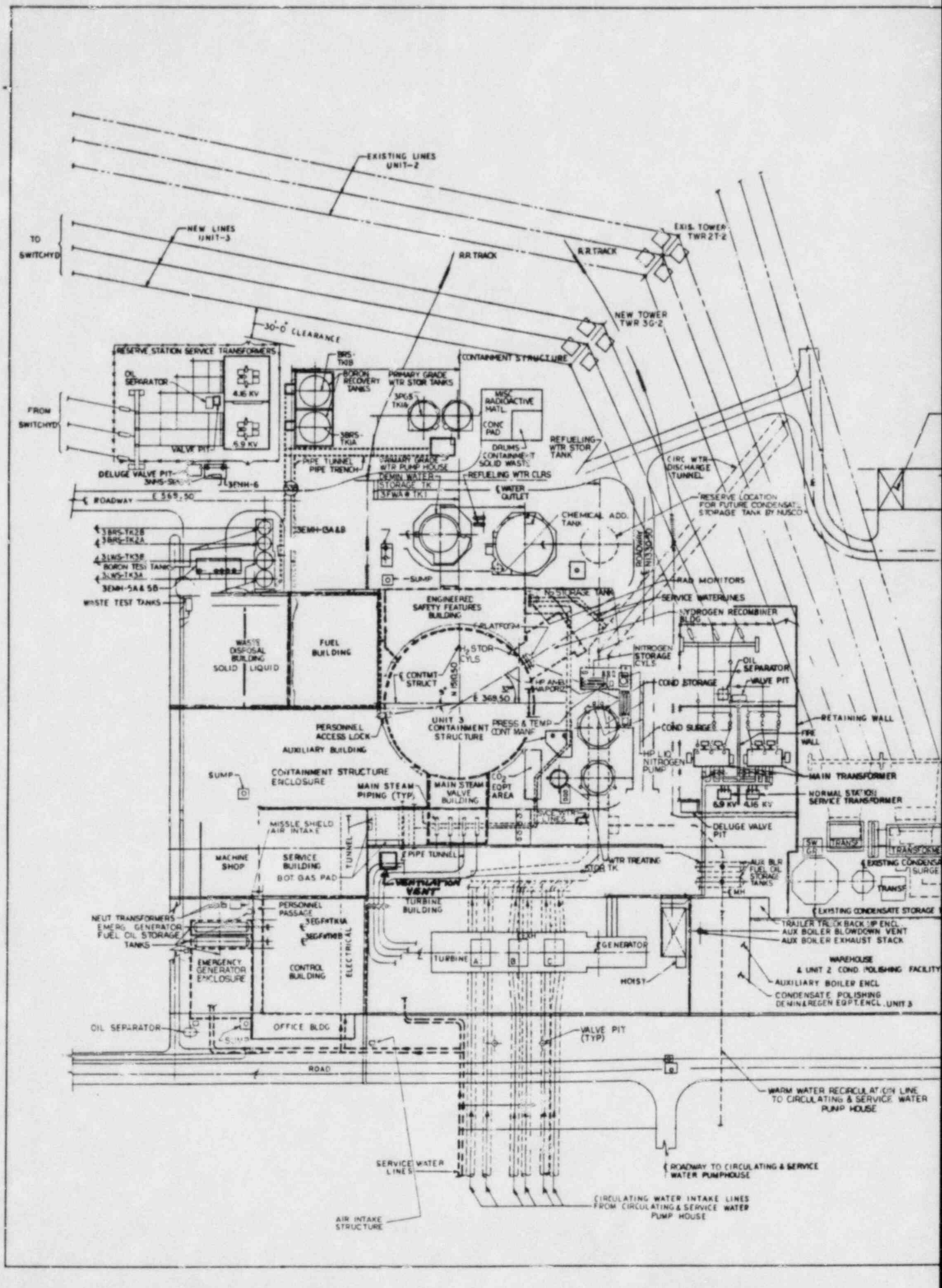


FIGURE 1.2-1
SITE PLAN
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT


EXISTING LINES UNIT-2
NEW LINES UNIT-3
TO SWITCHYD
R.R. TRACK
EXIS. TOWER TWR 2T-2
NEW TOWER TWR 3G-2
30'-0" CLEARANCE
RESERVE STATION SERVICE TRANSFORMERS
OIL SEPARATOR
4.16 KV
6.9 KV
FROM SWITCHYD
VALVE PIT
DELUGE VALVE PIT
BORON RECOVERY TANKS
PRIMARY GRADE WTR STOR TANKS
CONTAINMENT STRUCTURE
MISC RADIOACTIVE MATL.
CONC PAD
DRUMS CONTAINMENT SOLID WASTE
REFUELING WTR STOR TANK
CIRC WTR DISCHARGE TUNNEL
PIPE TUNNEL
PIPE TRENCH
DEMIN WATER STORAGE TK
REFUELING WTR CLRS
WATER OUTLET
ROADWAY
CHEMICAL ADD. TANK
RESERVE LOCATION FOR FUTURE CONDENSATE STORAGE TANK BY NUSCO
BORON TEST TANK
WASTE TEST TANKS
SUMP
RAD MONITORS
SERVICE WATERLINES
HYDROGEN RECOMBINER BLDG
ENGINEERED SAFETY FEATURES BUILDING
PLATFORM
WASTE DISPOSAL BUILDING
SOLID LIQUID
FUEL BUILDING
CONTMT STRUCT
NITROGEN STORAGE CYLS
OIL SEPARATOR
COND STORAGE
VALVE PIT
HP AMB VAPORIZ
RETAINING WALL
PERSONNEL ACCESS LOCK
UNIT 3 CONTAINMENT STRUCTURE
PRESS & TEMP CONT MANIF
COND SURGE
HP LIQ NITROGEN PUMP
FIRE WALL
AUXILIARY BUILDING
MAIN TRANSFORMER
CONTAINMENT STRUCTURE ENCLOSURE
MAIN STEAM PIPING (TYP)
MAIN STEAM VALVE BUILDING
CO_2 EQPT AREA
NORMAL STATION SERVICE TRANSFORMER
MISSLE SHIELD AIR INTAKE
DELUGE VALVE PIT
MACHINE SHOP
SERVICE BUILDING
BOT. GAS PAD
TUNNEL
PIPE TUNNEL
WTR TREATING
STOR TK
AUX BLR FUEL OIL STORAGE TANKS
EXISTING CONDENSATE SURGE
VENTILATION VENT
TURBINE BUILDING
NEUT TRANSFORMERS
EMERG. GENERATOR FUEL OIL STORAGE TANKS
PERSONNEL PASSAGE
EXISTING CONDENSATE STORAGE
TRAILER TRUCK BACK UP ENCL.
AUX BOILER BLOWDOWN VENT
AUX BOILER EXHAUST STACK
GENERATOR
TURBINE
WAREHOUSE & UNIT 2 COND. POLISHING FACILITY
EMERGENCY GENERATOR ENCLOSURE
CONTROL BUILDING
ELECTRICAL
HOIST
AUXILIARY BOILER ENCL
CONDENSATE POLISHING DEMIN & REGEN EQPT. ENCL. UNIT 3
OIL SEPARATOR
SUMP
OFFICE BLDG
VALVE PIT (TYP)
ROAD
WARM WATER RECIRCULATION LINE TO CIRCULATING & SERVICE WATER PUMP HOUSE
SERVICE WATER LINES
ROADWAY TO CIRCULATING & SERVICE WATER PUMPHOUSE
CIRCULATING WATER INTAKE LINES FROM CIRCULATING & SERVICE WATER PUMP HOUSE
AIR INTAKE STRUCTURE

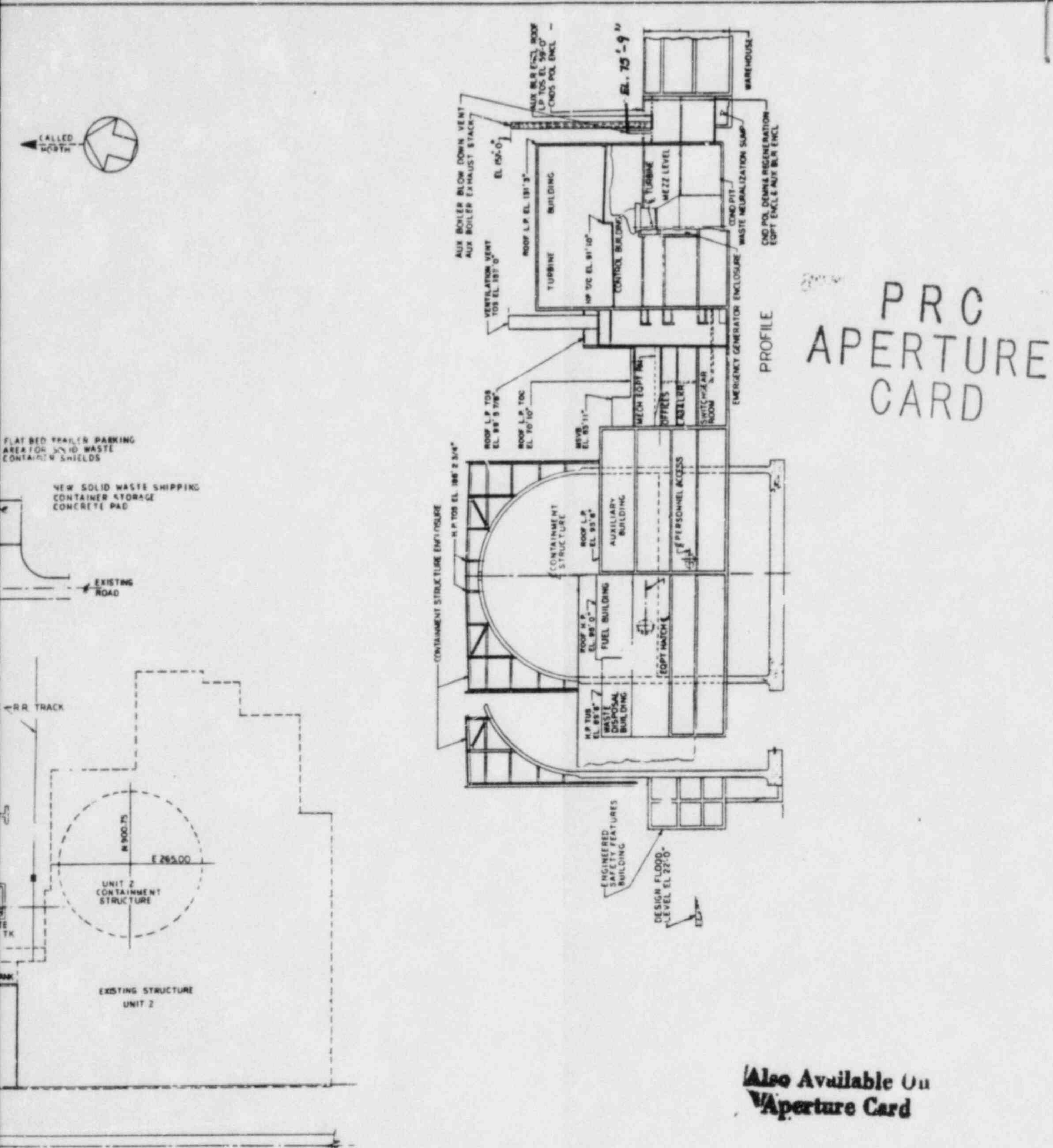
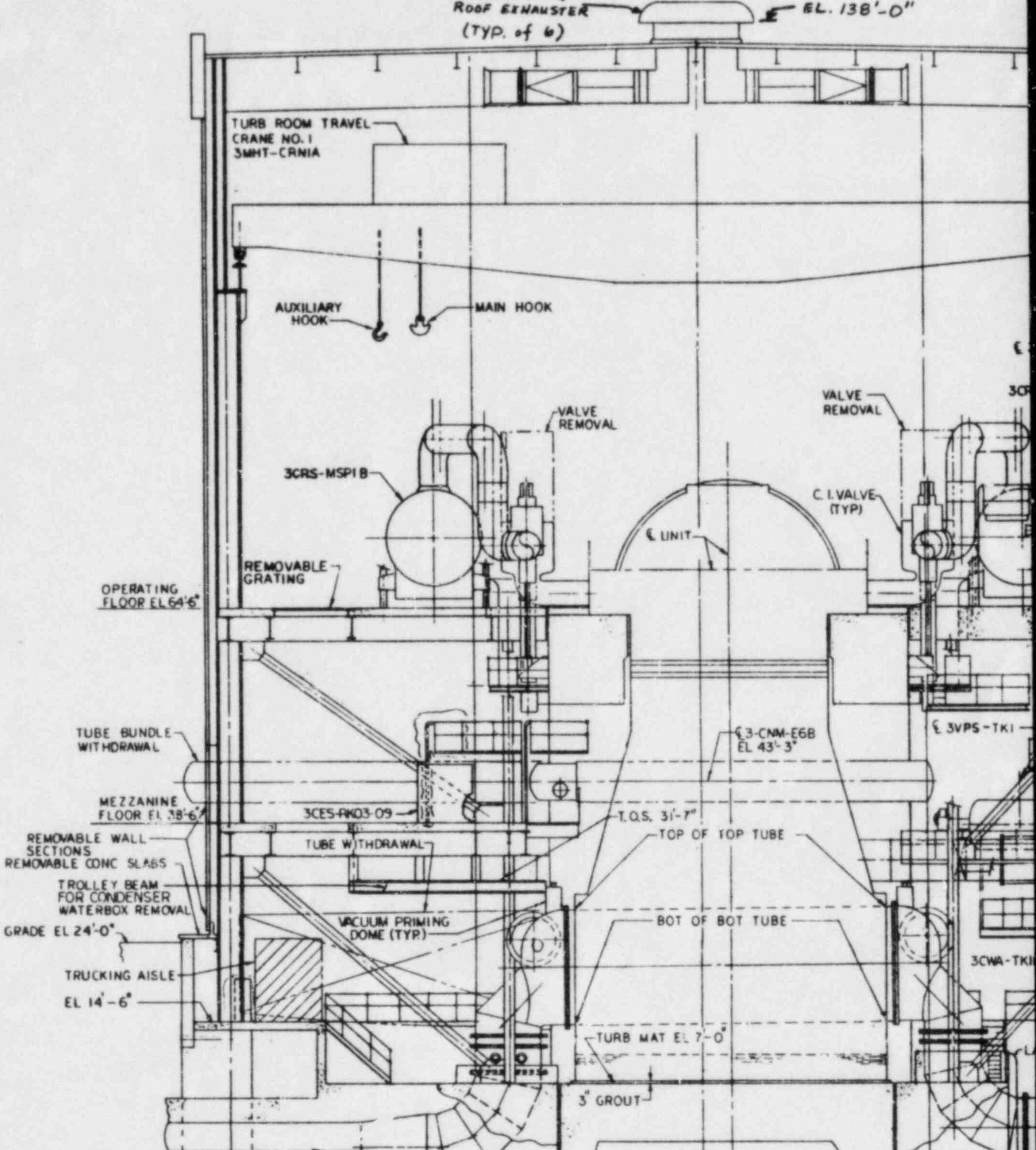


FIGURE 1.2-2
PLOT PLAN
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT


TURBINE BLDG. ROOF EXHAUSTER (TYP. of 6)
EL. 138'-0"
TURB ROOM TRAVEL CRANE NO. 1 3MHT-CRN1A
AUXILIARY HOOK
MAIN HOOK
VALVE REMOVAL
3CRS-MSP1B
C.I. VALVE (TYP)
℄ UNIT
REMOVABLE GRATING
OPERATING FLOOR EL 64'-6"
TUBE BUNDLE WITHDRAWAL
℄ 3-CNM-E6B EL 43'-3"
℄ 3VPS-TK1
MEZZANINE FLOOR EL 38'-6"
3CES-RK03-09
T.O.S. 31'-7"
TOP OF TOP TUBE
REMOVABLE WALL SECTIONS
REMOVABLE CONC SLABS
TUBE WITHDRAWAL
TROLLEY BEAM FOR CONDENSER WATERBOX REMOVAL
GRADE EL 24'-0"
VACUUM PRIMING DOME (TYP.)
BOT OF BOT TUBE
TRUCKING AISLE
EL 14'-6"
TURB MAT EL 7'-0"
3" GROUT
3CWA-STRID

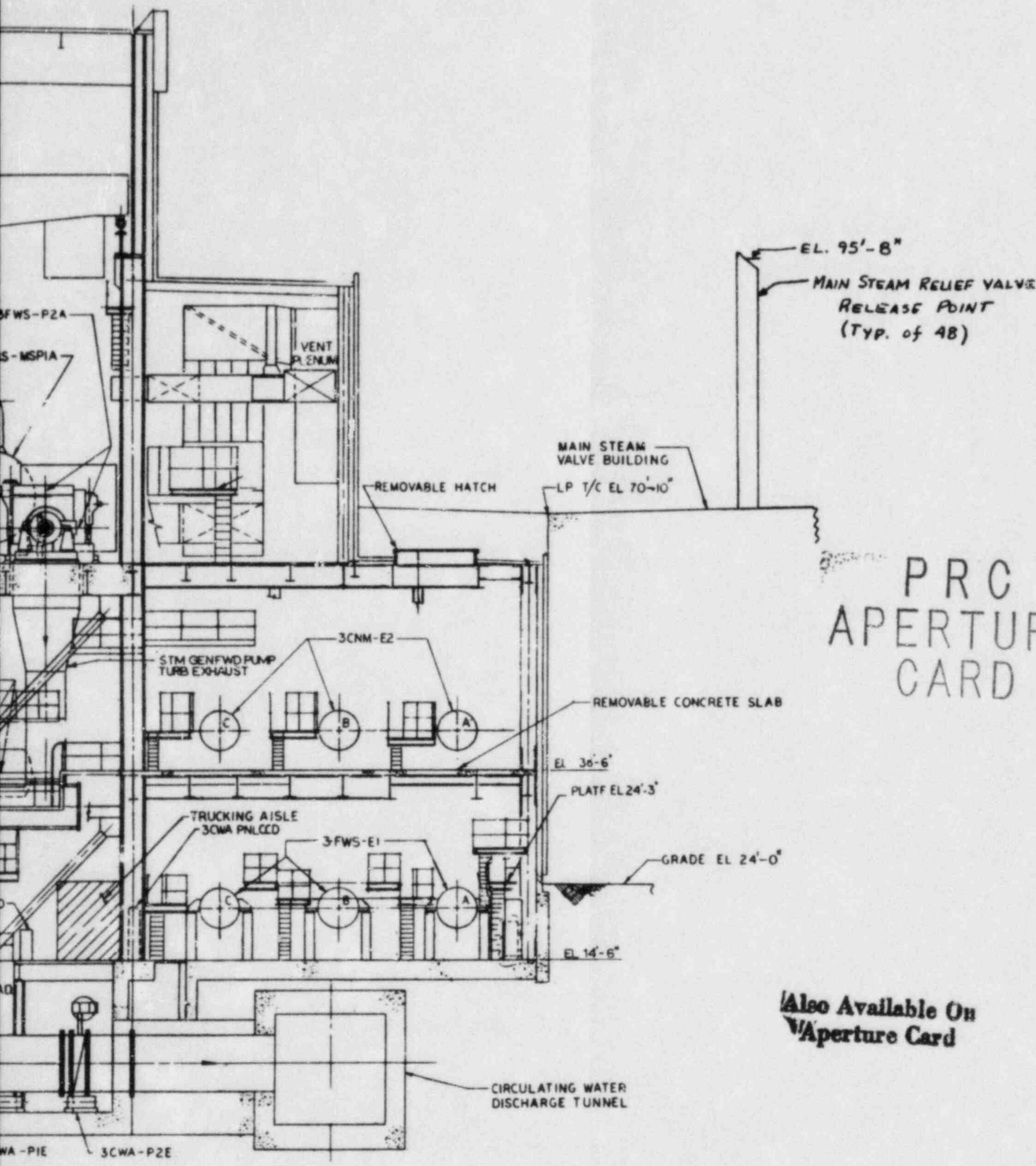


FIGURE 3.8-73 (5 OF 5)
TURBINE BUILDING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: May 3, 1983

Question No. Q460.9 (Section 11.3)

In our review, Subsection I.1 of SRP 11.3, the staff has located several P&ID differences from the description provided in the FSAR.

Provide clarification on the following:

a. The flow direction of the reactor plant gaseous waste and the condenser air removal lines to the radioactive waste line on Figure 9.3-5 (Sheet 1) at K-9 disagrees with Figure 10.4-2 (Sheet 2) at K-2.

b. The waste tank inlet and outlet are connected on Figure 9.3-6 (Sheet 1) at K-4 or Figure 11.2-1.

c. The BRS inlet line EM-109A at E-10 is not shown on Figure 9.3-9 (Sheet 1) at B-2.

d. Figure 11.5-1 indicates monitor 3HVR-REID. Should this be 3HVR-RE10A and for 10B, as given on Figure 3.8-62 (Sheet 4)?

e. Figure 9.4-3 (Sheet 2) at I-1 indicates ventilation release to Unit No. 2. How is this line monitored prior to gas release at Unit No. 3?

Response:

a. The apparent disagreement of Figure 9.3-5 (Sheet No. 1), Figure 10.4-2 (Sheet No. 2) and Figure 11.3-1 (Sheet No. 2) has been resolved in revised P&IDs (EM-107A, EM-109B).

b. The waste tank inlet/outlet connections are shown as being connected on both figures because Figure 9.3-6 and Figure 11.2-1 is the same P&ID for both the radioactive liquid waste (LWS) and aerated drains systems (DAS). Therefore, the information is duplicated for both figures.

c. The connection is located on Figure 9.3-9 (Sheet No. 1) at coordinates A-1. This discrepancy has been resolved in the revised P&IDs (EM-108A, EM-109A).

d. 3HVR*REID should be 3HVR*RE10A,B. This discrepancy has been resolved in the revised Figure 11.5-1.

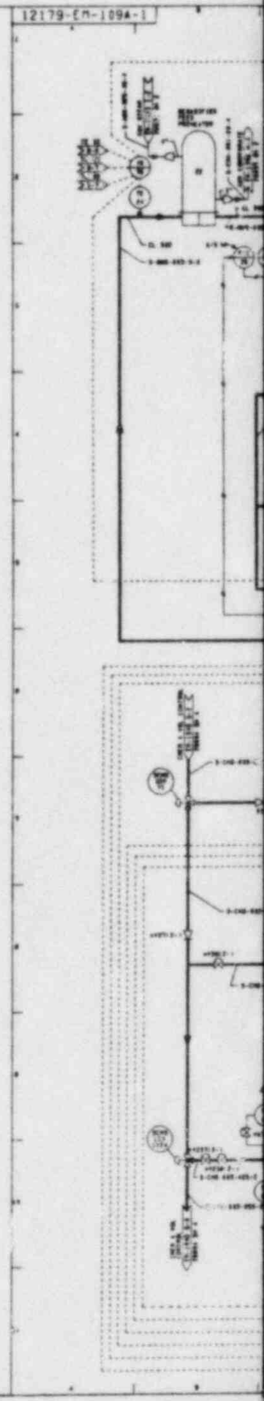
e. This ventilation system is for the warehouse general area, classrooms and offices. It is not exposed to any contaminated or potentially contaminated systems/equipment. Therefore, monitoring prior to gas release at Unit No. 3 is not required.

P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1

P&ID NO.	FSAR FIGURE NO.
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3

P&ID NO.	FSAR FIGURE NO.
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1



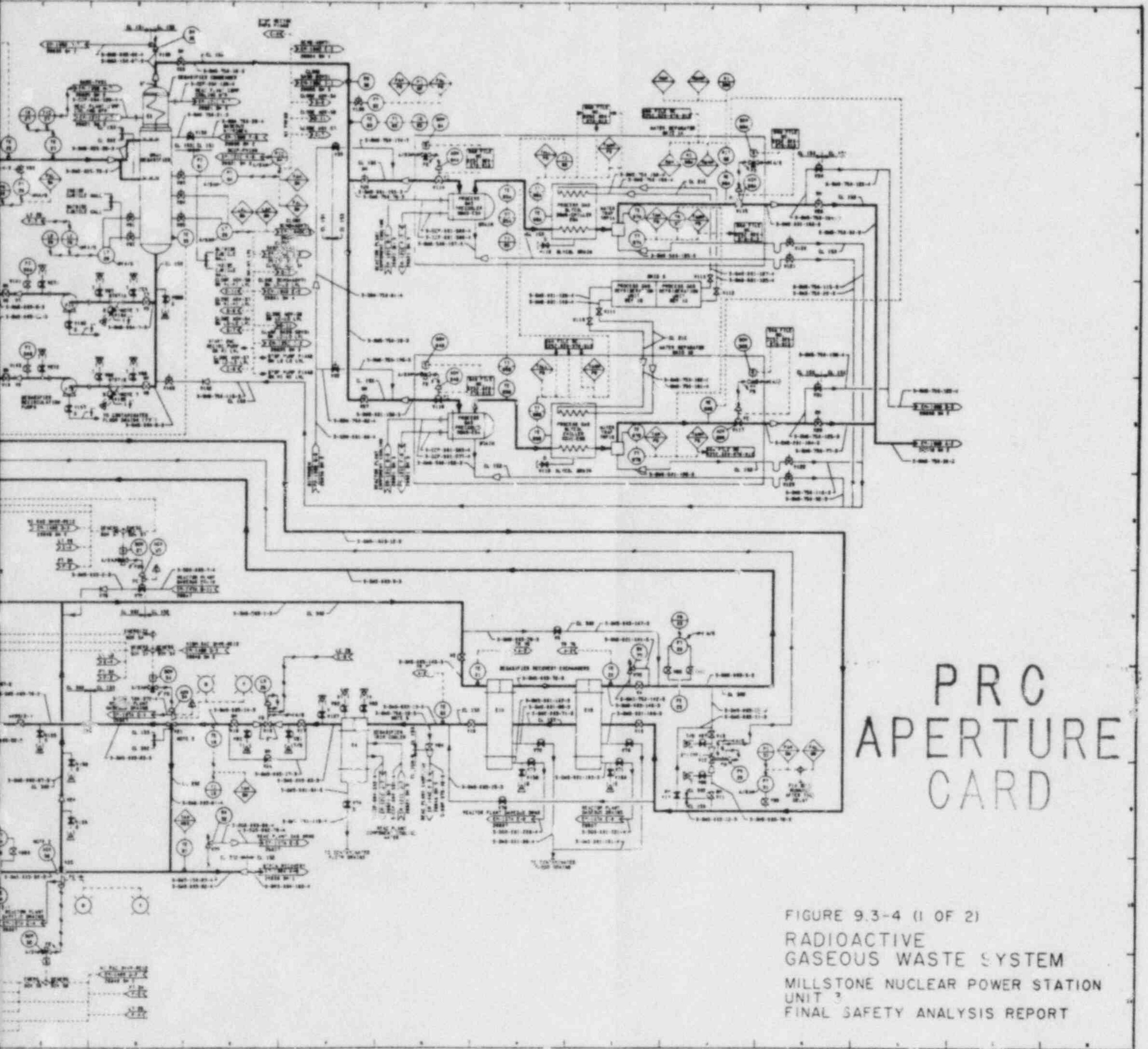
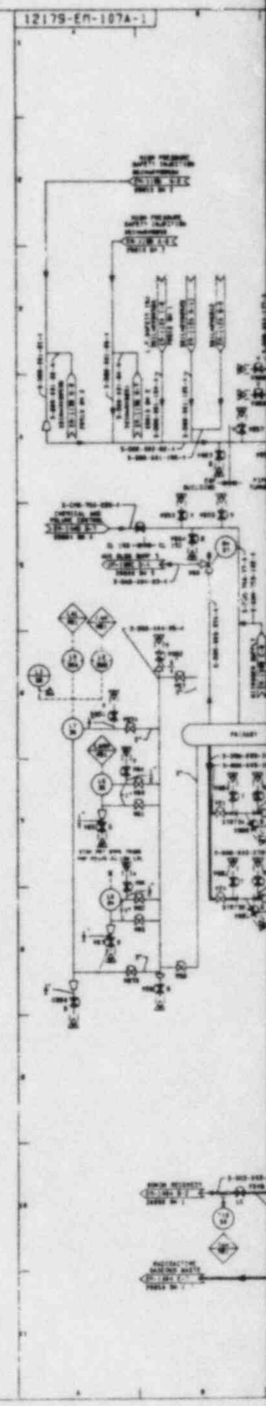


FIGURE 9.3-4 (1 OF 2)
RADIOACTIVE
GASEOUS WASTE SYSTEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Amendment 3 August 1983

P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1



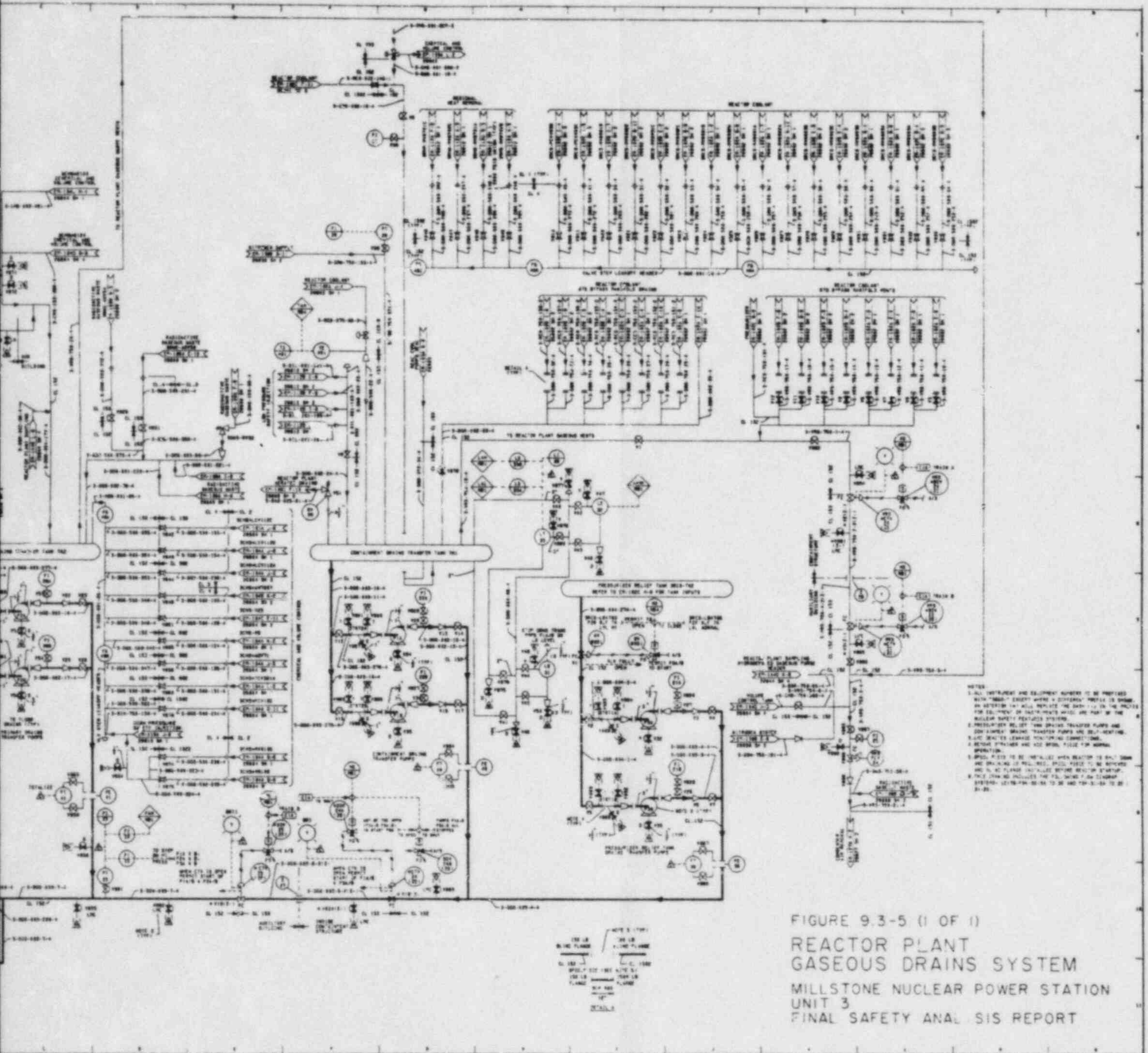


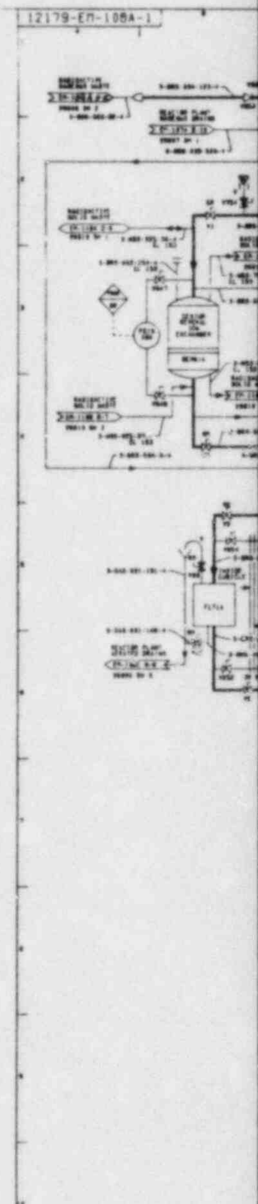
FIGURE 9.3-5 (1 OF 1)

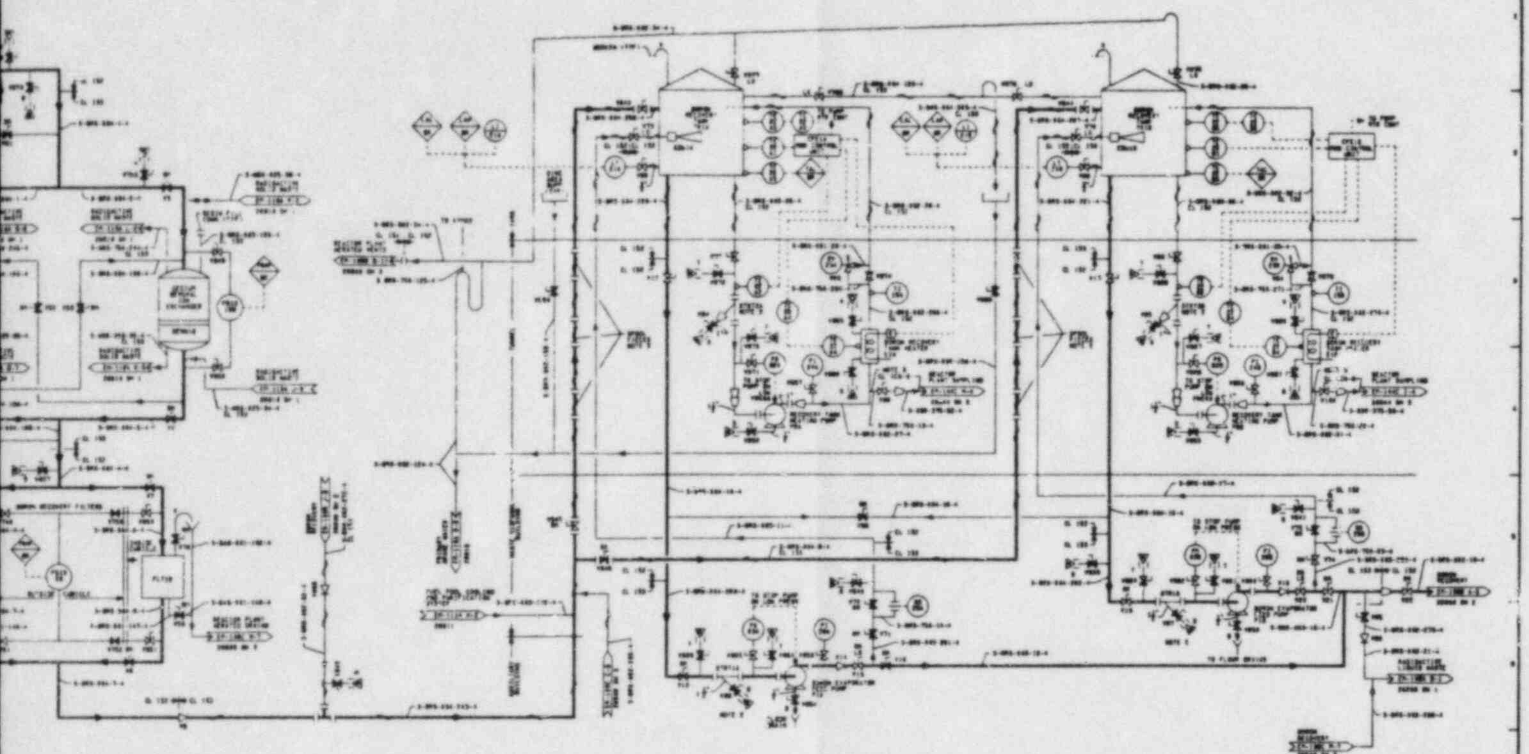
REACTOR PLANT
GASEOUS DRAINS SYSTEM

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1





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APERTURE
CARD

FIGURE 9.3-9 (1 OF 3)
BORON RECOVERY
SYSTEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Amendment 3 August 1983

Also Available On
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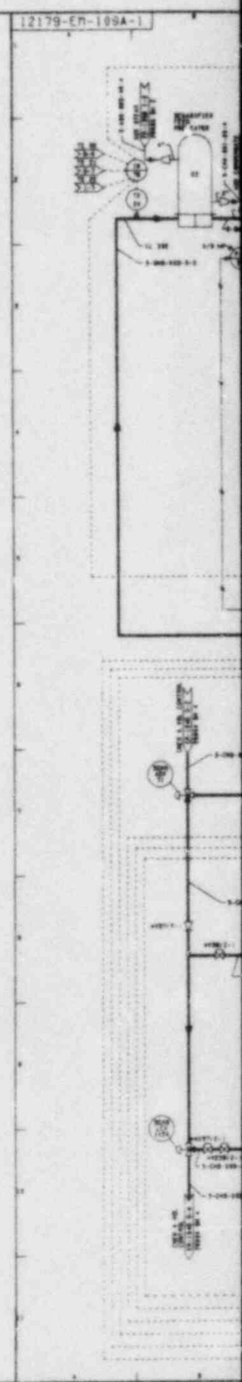
8307260197 -17

P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1

P&ID NO.	FSAR FIGURE NO.
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3

P&ID NO.	FSAR FIGURE NO.
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-6 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1



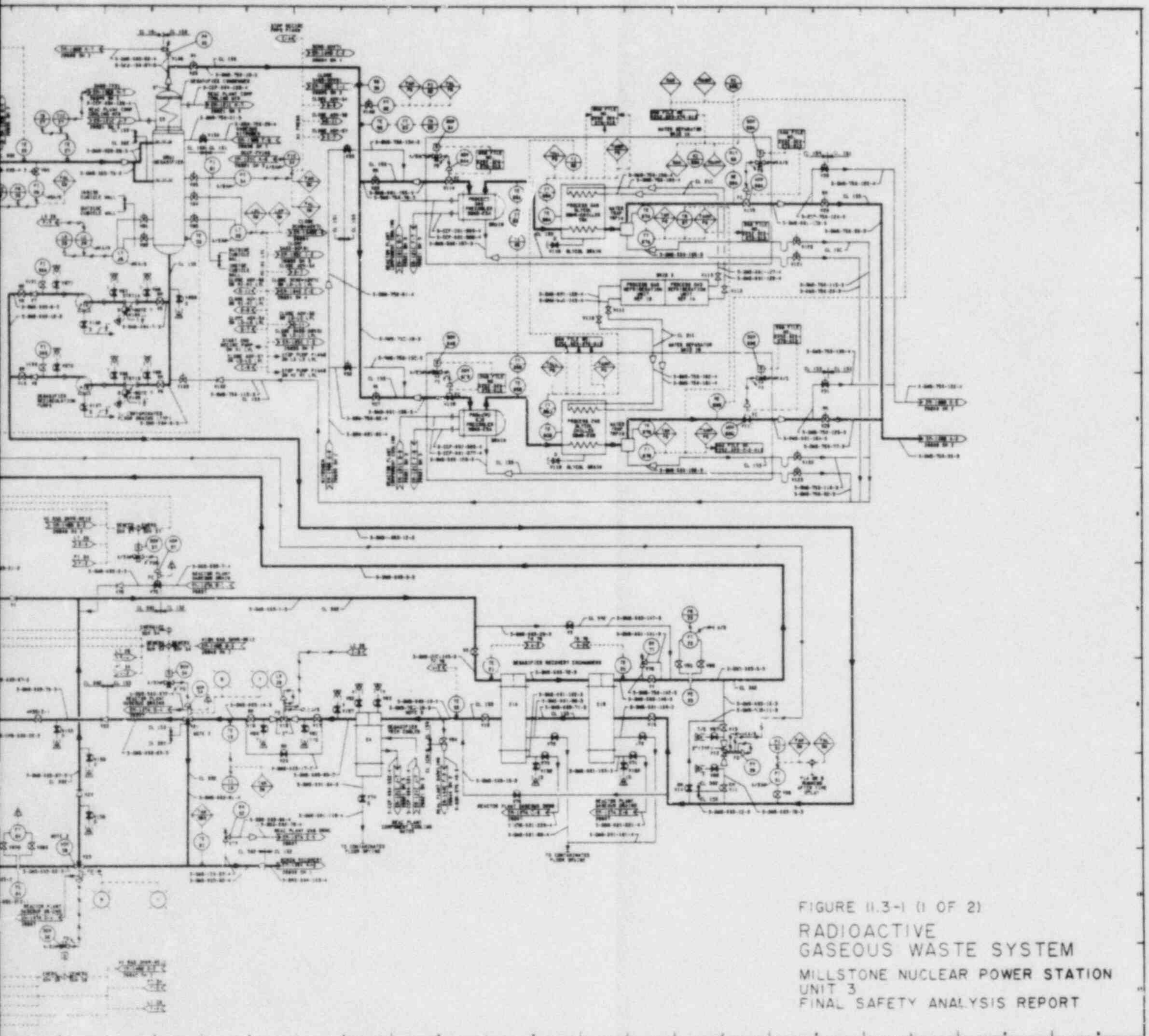


FIGURE 11.3-1 (1 OF 2)

RADIOACTIVE
GASEOUS WASTE SYSTEM

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

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Also Available On
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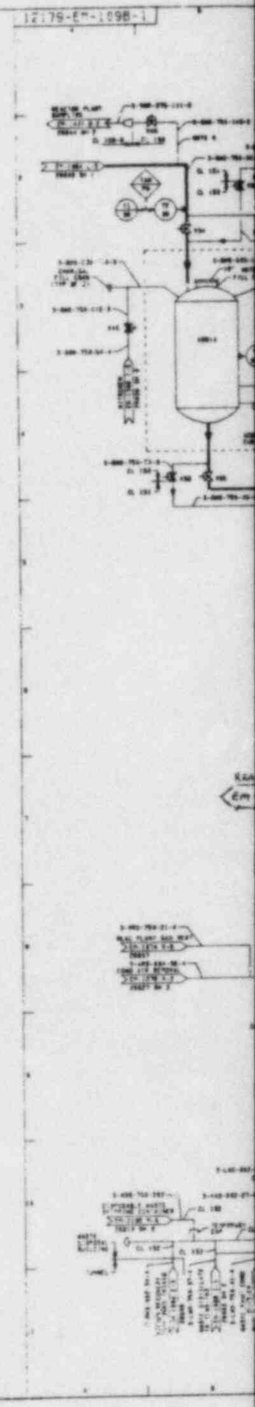
8307260197-18

P&ID-FSAR CROSS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
EM100A	1.2-3 SH1
EM100B	1.2-3 SH2
EM100C	1.2-3 SH3
EM102A	5.1-1 SH1
EM102B	5.1-1 SH2
EM102C	5.1-1 SH3
EM103A	9.3-7 SH1
EM104A	9.3-8 SH1
EM104B	9.3-8 SH2
EM104C	9.3-8 SH3
EM104D	9.3-8 SH4
EM105A	9.2-5 SH1
EM106A	9.3-6 SH1 & 11.2-1 SH1
EM106B	9.3-6 SH2 & 11.2-1 SH2
EM106C	9.3-6 SH3 & 11.2-1 SH3
EM107A	9.3-5 SH1
EM108A	9.3-9 SH1
EM108B	9.3-9 SH2
EM108C	9.3-9 SH3
EM109A	9.3-4 SH1 & 11.3-1 SH1
EM109B	9.3-4 SH2 & 11.3-1 SH2
EM110A	11.4-1 SH1
EM110B	11.4-1 SH2
EM111A	9.1-6 SH1
EM112A	5.4-5 SH1 & 6.2-37 SH1
EM112B	5.4-5 SH2 & 6.2-37 SH2
EM112C	5.4-5 SH3 & 6.2-37 SH3
EM113A	6.3-2 SH1
EM113B	6.3-2 SH2
EM114A	9.2-4 SH1
EM115A	6.2-36 SH1
EM116A	9.5-3 SH1
EM116B	9.5-3 SH2
EM117A	9.5-2 SH1
EM119A	9.2-11 SH1
EM120A	9.2-7 SH1
EM120B	9.2-7 SH2
EM120C	9.2-7 SH3
EM120D	9.2-7 SH4
EM121A	9.2-2 SH1
EM121B	9.2-2 SH2
EM121C	9.2-2 SH3
EM122A	9.2-3 SH1
EM122B	9.2-3 SH2
EM123A	10.3-1 SH1
EM123B	10.3-1 SH2
EM123C	10.3-1 SH3
EM123D	10.3-1 SH4
EM124A	10.4-3 SH1
EM124B	10.4-3 SH2
EM125A	10.4-7 SH1
EM125B	10.4-7 SH2
EM125C	10.4-7 SH3
EM126A	9.2-9 SH1 & 10.4-1 SH1
EM126B	9.2-9 SH2 & 10.4-1 SH2
EM126C	9.2-9 SH3 & 10.4-1 SH3
EM127A	10.4-2 SH1
EM127B	10.4-2 SH2
EM128A	10.4-5 SH1
EM128B	10.4-5 SH2
EM128C	10.4-5 SH3
EM128D	10.4-5 SH4
EM128E	10.4-5 SH5
EM129A	9.2-6 SH1 & 11.2-2 SH1
EM130A	10.4-6 SH1
EM130B	10.4-6 SH2
EM131A	10.3-3 SH1
EM132A	10.4-4 SH1
EM132B	10.4-4 SH2
EM133A	9.2-1 SH1
EM133B	9.2-1 SH2
EM134A	9.2-10 SH1
EM134B	9.2-10 SH2
EM135A	10.4-9 SH1
EM135B	10.4-9 SH2
EM135C	10.4-9 SH3
EM136A	9.4-7 SH1
EM137A	9.4-8 SH1
EM137B	9.4-8 SH2
EM137C	9.4-8 SH3
EM138A	9.3-1 SH1
EM138B	9.3-1 SH2
EM138C	9.3-1 SH3
EM139A	9.5-5 SH1
EM139B	9.5-5 SH2
EM140A	10.2-1 SH1
EM141A	10.2-2 SH1
EM141B	10.2-2 SH2
EM142A	10.2-3 SH1
EM143A	9.3-3 SH1
EM144A	9.3-2 SH1
EM144B	9.3-2 SH2
EM144C	9.3-2 SH3
EM144D	9.3-2 SH4
EM145A	10.3-2 SH1
EM145B	10.3-2 SH2
EM145C	10.3-2 SH3
EM146A	9.5-1 SH1
EM146B	9.5-1 SH2
EM146C	9.5-1 SH3
EM147A	9.2-8 SH1
EM147B	9.2-8 SH2
EM148A	9.4-2 SH1
EM148B	9.4-2 SH2
EM148C	9.4-2 SH3
EM149A	9.4-6 SH1
EM149B	9.4-5 SH2
EM150A	9.4-3 SH1
EM150B	9.4-3 SH2
EM150C	9.4-3 SH3
EM151A	9.4-1 SH1
EM151B	9.4-1 SH2
EM151C	9.4-1 SH3
EM151D	9.4-1 SH4
EM151E	9.4-1 SH5
EM152A	9.4-4 SH1
EM152B	9.4-4 SH2
EM153A	9.4-5 SH1
EM154A	6.2-53 SH1 & 12.3-5 SH1



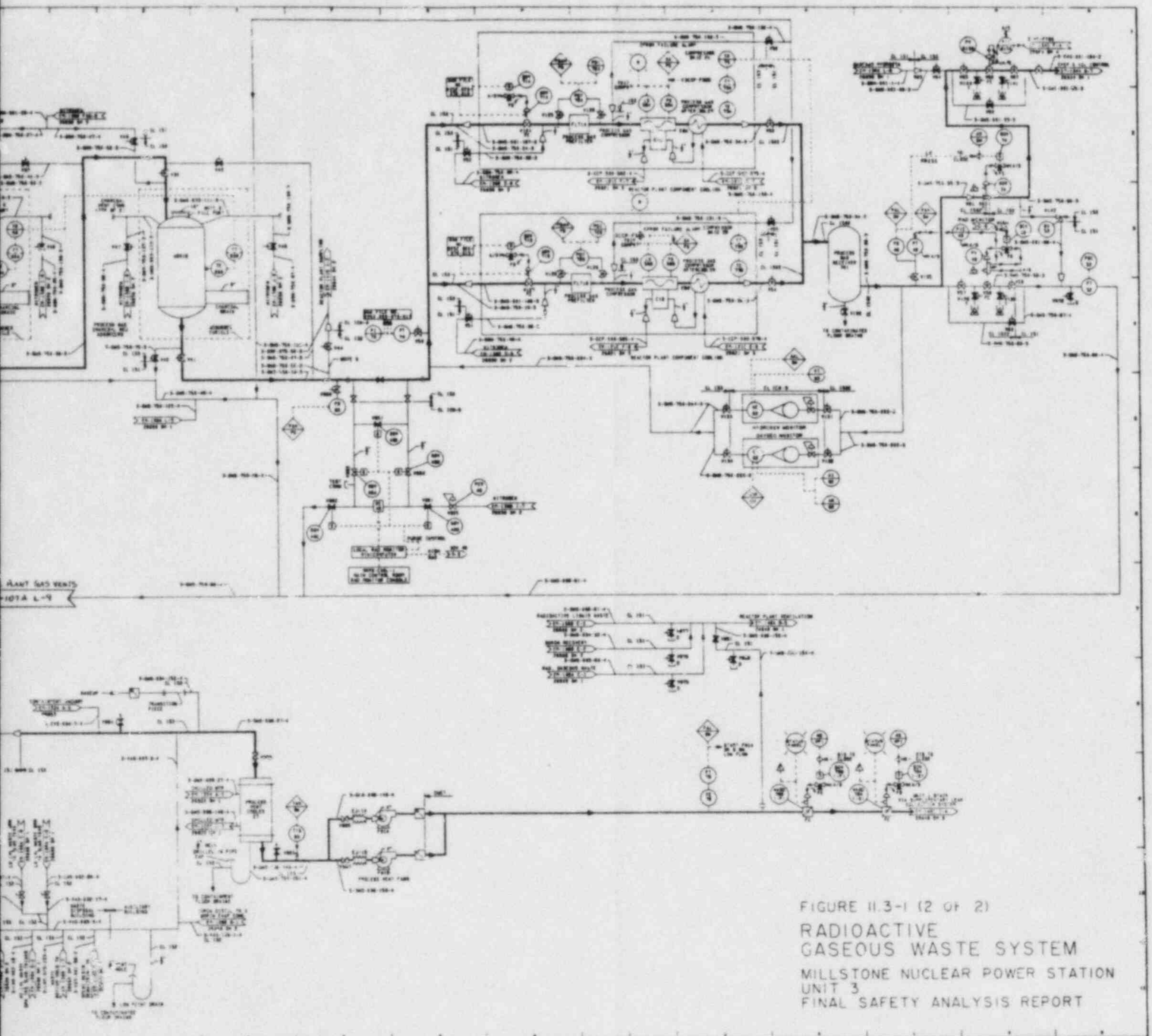
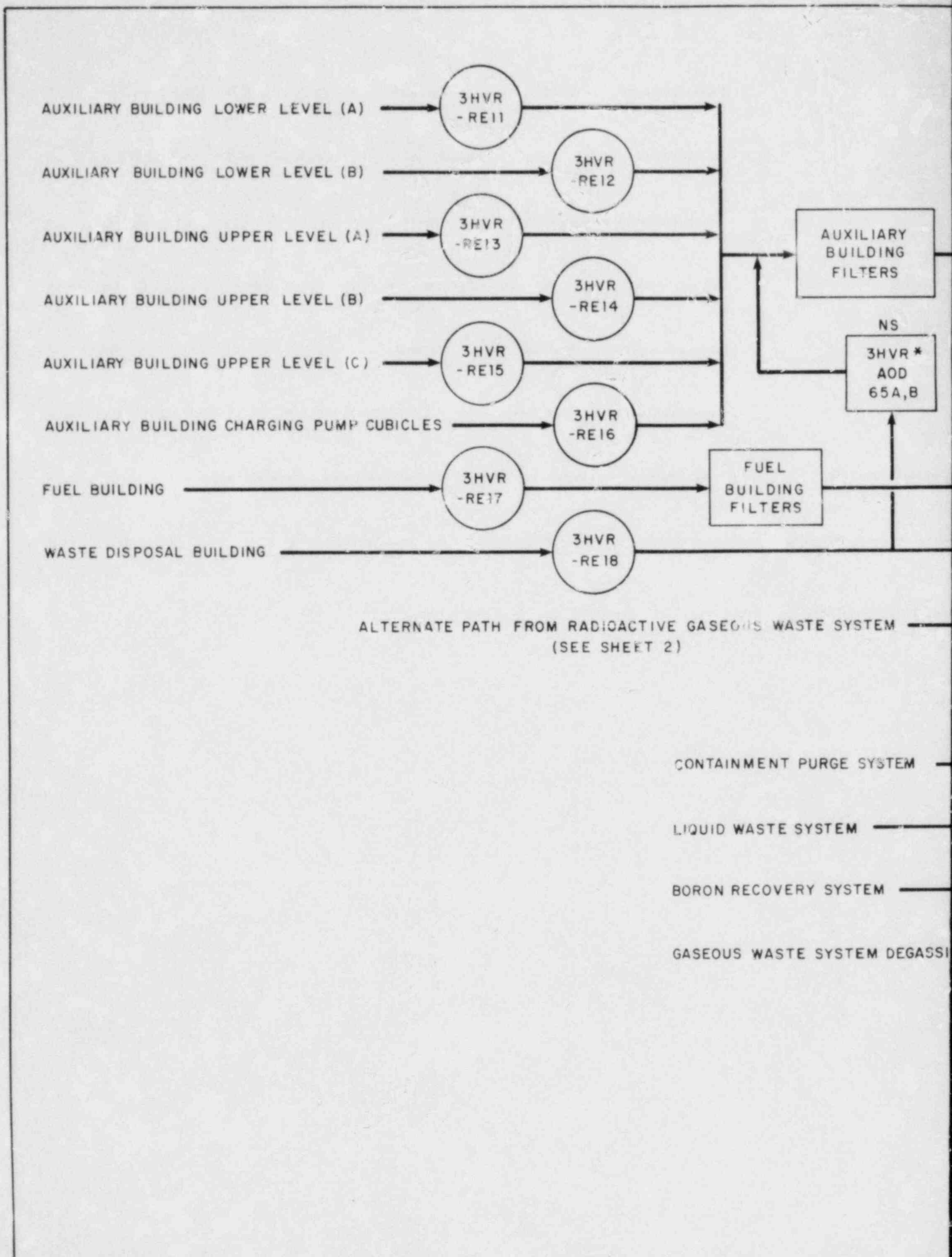


FIGURE 11.3-1 (2 OF 2)
RADIOACTIVE
GASEOUS WASTE SYSTEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT



AUXILIARY BUILDING LOWER LEVEL (A)
3HVR -RE11
AUXILIARY BUILDING LOWER LEVEL (B)
3HVR -RE12
AUXILIARY BUILDING UPPER LEVEL (A)
3HVR -RE13
AUXILIARY BUILDING UPPER LEVEL (B)
3HVR -RE14
AUXILIARY BUILDING UPPER LEVEL (C)
3HVR -RE15
AUXILIARY BUILDING CHARGING PUMP CUBICLES
3HVR -RE16
FUEL BUILDING
3HVR -RE17
WASTE DISPOSAL BUILDING
3HVR -RE18
AUXILIARY BUILDING FILTERS
NS
3HVR* AOD 65A,B
FUEL BUILDING FILTERS
ALTERNATE PATH FROM RADIOACTIVE GASEOUS WASTE SYSTEM
(SEE SHEET 2)
CONTAINMENT PURGE SYSTEM
LIQUID WASTE SYSTEM
BORON RECOVERY SYSTEM
GASEOUS WASTE SYSTEM DEGASSI

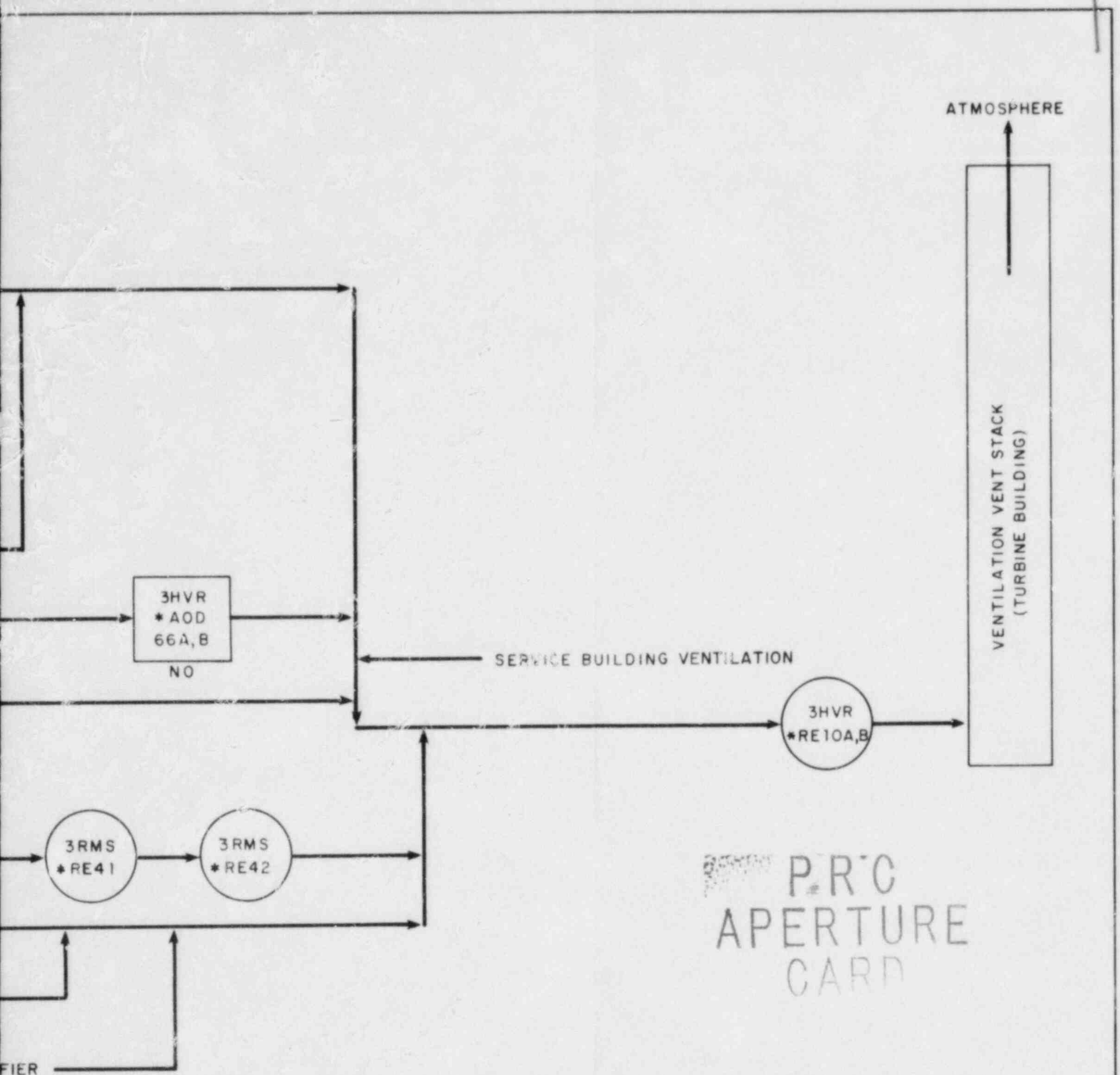


FIGURE 11.5-1 (SHEET 1 OF 4)
RADIATION MONITORS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: May 3, 1983

Question No. Q460.12 (Sections 6.5.1)

Regulatory Guide 1.52 and 1.140 recommend leak testing of dampers used in ESF and non-ESF air filtration systems. In Table 1.8-1, pages 21 and 55, you have taken exception to testing every damper, and propose to test every type of damper. Since leakage is a function of valve size, we recommend that you determine Class B leakage rates for at least one damper of each size and type used in the ESF and non-ESF atmospheric cleanup systems, as an acceptable alternative.

Response:

Except for two backdraft dampers on the supplementary leak collection and release system fan discharge, all other dampers on ESF atmosphere cleanup system air filtration and adsorption units have been tested for leakage rates. The size of the two untested backdraft dampers is bounded by both larger and smaller size dampers which have been satisfactorily tested to 50 percent of allowable leakage rates.

The only dampers associated with non-ESF atmosphere cleanup system air filtration and adsorption units are ANSI N509 leakage Class IV which do not require testing.

NRC Letter: May 3, 1983

Question No. Q460.13 (Sections 11.4)

SRP 11.4 calls for a description and design bases for solid radioactive waste handling systems. Table 1.8-1, page 17, of the FSAR, states that the charcoal adsorber in filtration trains will be replaced using an external vacuum system. Provide the description, P&IDs, bases, and details relative to this equipment and describe the provisions for handling the charcoal adsorber media from the trains to the SWS.

Response:

A description of the vacuum system that is being used for charcoal removal is outlined in CVIs Topical Report No. CVI-TR-7301 dated February 1975 and approved by NRC in their letter of April 30, 1975 (Klecker to H. Parrish). The charcoal is collected in 55-gallon drums and will be treated as low level solid waste as described in revised FSAR Section 11.4.2.1.5.

solidification, and ultimate shipment offsite. The volume of bottoms solution to be shipped offsite is given on Figure 11.4-2. The calculated activity of these bottoms, also shown on Figure 11.4-2, is based on the input of the radioactive liquid waste system (Section 11.2).

11.4.2.1.3 Regenerant Chemical Evaporator Bottoms

The regenerant chemical evaporator is operated so as to discharge bottoms at about 24 percent by weight solids concentration solidification, and ultimate shipment offsite. The volume and activity of the waste is given on Figure 11.4-2.

11.4.2.1.4 Boron Evaporator Bottoms

The boron evaporator in the boron recovery system processes reactor coolant letdown to the boron recovery system in order to recover and recycle boric acid and water for reuse. It is estimated that, approximately once every year, one boron evaporator volume is solidified. The volume given on Figure 11.4-2 for the boron evaporator bottoms is a 1 year average of the contents requiring processing for eventual offsite disposal. The activity shown for these bottoms, given on Figure 11.4-2, is based on the expected performance of the boron recovery system (Section 9.3.5).

11.4.2.1.5 Miscellaneous Radioactive Solid Wastes

It is estimated that approximately 2,000 ft^3 per year of additional waste requiring disposal is processed in the radioactive solid waste system. This volume of spent filters, contaminated cloths, and other radioactive material and their activity, given on Figure 11.4-2, was estimated from operating experience at other nuclear facilities.

460.13 This category includes waste charcoal adsorber media from filtration units described in Section 9.4. Charcoal is removed by an external vacuum system outlined in CVI Topical Report No. CVI-TR-7301 (February 1975). Charcoal is collected in 55 gallon drums.

11.4.2.2 Equipment Description

The radioactive solid waste system equipment is operated on a batch basis. Individual components are designed to support the system's rated capacity.

The system consists of a fill head, a control panel, spent resin dewatering and hold tanks, a storage tank for Dow binder, and the piping, pumps, and process equipment modules required for transfer and dewatering or solidification of the wastes. The solid waste equipment requires a minimum manual action and, in conjunction with the building layout, is designed to minimize occupational radiation exposures.

There are two versions of the shipping container: a high intensity container, used for slurries with an integral dewatering line;

another is used for solidification and has an integral mixing blade assembly.

When slurries are being processed, the fill head will be lifted by the overhead bridge crane and lowered onto the shipping container having the integral dewatering line. Connected to the fill head is a waste fill line, vent line, decontamination lines, and a dewatering line that extends from the fill head. These lines are flexible, with quick disconnect fittings to facilitate periodic replacement or maintenance. When the fill head is lowered onto the shipping container, it is aligned with the internal dewatering line and secured prior to filling.

Waste concentrates from the waste evaporator in the radioactive liquid waste system and any concentrated boric acid discarded from

NRC Letter: May 3, 1983

Question No. Q460.14 (Sections 11.4)

SRP 11.4 calls for a description and design bases for solid radioactive waste handling systems. Although Table 11.4-1 includes spent resins from the condensate demineralizers, we find no provisions for handling these spent resins (Section 10.4.6.2). Provide the information, design bases, and the appropriate P&ID figure(s).

Response:

Refer to the revised FSAR Sections 11.4.2.3 and 10.4.6.2 for the response to this question.

The condensate polishing system ion exchange resins require periodic cleaning and regeneration whenever either a demineralizer effluent quality does not meet acceptable values or the pressure drop across a resin bed is too high. The resin cleaning and/or chemical regeneration is done externally to the demineralizers.

The polisher regeneration equipment consists of a cation regeneration vessel, an anion regeneration vessel, a resin mix and storage vessel, a lime system for regenerating the resin when the polishing system is operated in the ammonia cycle, and an ultrasonic resin cleaner.

Exhausted resins are transferred from the demineralizer to the cation regeneration vessel where the anion and cation resins are separated. The anion resin is then transferred to the anion regeneration vessel. The resins are chemically regenerated using sulfuric acid for cation resin and a caustic solution for anion resin. The regenerated resins are transferred to the resin mix and storage vessel for mixing and sluicing back to the demineralizer vessel. Spent resin can be disposed of by sluicing from the resin mix and storage vessel to the radioactive waste disposal system of Unit 2. [See Figure 10.4-5 (3 of 5).] 460.14

The design pressure of the condensate side of the condensate demineralizer system is 700 psig.

10.4.6.3 Safety Evaluation

This system is not safety related. Failure of any portion or component of this system will not damage any safety related component or system. The following design evaluation is provided to demonstrate the system capability to perform its intended function.

<u>Massive Condenser Tube Failure</u>

The system provides some degree of protection even during massive leaks, such as a complete tube failure, thus affording "reaction" time to take corrective action or initiate a unit shutdown.

In order to prevent resin breakthrough into the steam generators, the condensate polishing demineralizer (CND) system design has the following features:

1. Each condensate polishing demineralizer, 3CND-DEMN1A through H, is provided with an underdrain system consisting of a rubber lined steel header and stainless steel Johnson well-screen laterals wrapped with two layers of Saran fabric capable of retaining resin beads greater than 50 mesh. 281.10

2. Resin strainers (traps), 3CND-STR1A through H, are used on the outlet of each demineralizer prior to manifolding at the common effluent header. The strainer baskets are designed to retain all particles larger than 50 mesh if demineralizer underdrain failure occurs.

The screen analysis of cation resin DOWEX HCR-W2 and anion resin DOWEX SBR-P used in the CND system shows that the resin beads are larger than 50 mesh (Table 10.4-7).

Each strainer is provided with a pressure differential indicating switch, 3CND-PDIS 35A through H, which actuates an alarm and an indicating light when the strainer has become fouled by resin and should be cleaned by backflushing.

280.10

3. The broken, disintegrated, and worn beads and resin fines (finer than 50 mesh) formed by attrition are removed from the resin along with accumulated crud in the ultrasonic resin cleaner (3CND-URC1) and during air/water scrub and backwash steps in the cation regeneration tank (3CND-TK1).

10.4.6.4 Tests and Inspection

The polisher vessels are hydrostatically tested in the assembly shop. The control system is also tested in the shop to ensure proper valve and pump operation sequences. Following installation, the system is hydrostatically tested as a complete unit.

The condensate polishing system is in continuous operation whenever the condensate system is operating. If no condenser in-leakage occurs, each demineralizer is regenerated at least once every 60 days; therefore, operability of the demineralizers and the regeneration system is demonstrated on a regular basis.

System equipment is tested for leakage and proper automatic operation prior to initial startup of the unit.

10.4.6.5 Instrumentation Requirements

The conductivity of the influent condensate to the condensate demineralizers, the effluent from each demineralizer, and the condensate returning to the condensate header is measured and recorded continuously, and an alarm signal is provided on the condensate polishing panel to indicate high conductivity.

A differential pressure transmitter is provided to monitor the differential pressure across the condensate demineralizer system. An alarm signal is provided to the condensate polishing panel and to the main control board common alarm, to indicate high differential pressure. A manual pushbutton is provided to open the normally closed bypass valve to bypass the demineralizer system.

Flow transmitters, recorders, and flow indicating totalizers are provided to monitor the throughput of each condensate demineralizer.

Continuous measurement of the waste stream for any radioactivity both before and during disposal is provided by an element installed on a sample line which discharges back to the sump. Presence of radioactivity automatically prevents disposal to the circulating

water outfall line. The validity of measurement by further tests and any possible need for diversion to the radwaste system will be assessed by station operating personnel.

10.4.7 Condensate and Feedwater Systems

The condensate and feedwater systems (Figures 10.4-1 and 10.4-6) supply feedwater at the required temperature, pressure, and flow rate to the secondary sides of the steam generators.

10.4.7.1 Design Basis

The condensate and feedwater systems are designed to provide approximately 15.12 x 10^6 pounds per hour of feedwater at 436°F to the steam generators during steady-state operation at maximum guaranteed turbine load (i.e., at NSSS warranted power). The portion of the feedwater system from the steam generators up to the first restraint beyond the isolation valve outside containment is Safety Class 2 and is designed to Quality Group B standards as defined in Regulatory Guide 1.26. The portion of the feedwater system upstream of the first restraint beyond the isolation valve to the turbine building wall is Safety Class 3 and is designed to Quality Group C standards. The remainder of the feedwater system and the entire condensate system are nonsafety related and are designed to Quality Group D standards. The safety related portions of the condensate and feedwater system are designed in accordance with the seismic design criteria discussed in Section 3.7.3. The remainder of the feedwater system and the entire condensate system are nonseismic.

and are transported to, and lowered into, the disposible shipping container for immobilization at the processing area and subsequent shielding and shipment offsite.

11.4.2.2.4 Incompressible Waste Handling

Contaminated metallic materials and solid objects are placed in disposable shipping containers.

11.4.2.2.5 Waste Compaction Operation

Contaminated compressible materials are temporarily stored in suitably labeled waste hampers or polyethylene bags in different plant locations. These materials are transported to the waste compactor for compression to high density in containers meeting applicable DOT specifications. Additional compressible material is added and the container contents recompacted until it is filled. These sealed containers are temporarily stored until shipped offsite (Section 11.4.2.5).

11.4.2.3 Expected Volumes

Table 11.4-1 presents a listing of the expected volumes of spent resins from various sources entering the radioactive solid waste system. (Note that resin from the condensate polishing 460.14
demineralizers indicated on Table 11.4-1 is processed by Millstone Unit 2 - See Figure 10.4-5.) Figure 11.4-2 presents gross activities and weights or volumes for radioactive solid waste sources. Containers of evaporator bottoms, resins, and miscellaneous 430.75
incompressible waste, are considered "wet" and are either dewatered and shipped offsite in high integrity containers or solidified for 460.14
shipment offsite. Compacted or compressible waste is "dry" and contained in 55-gallon drums.

11.4.2.4 Packaging

Based on the gross activities supplied on Figure 11.4-2 and DOT Low Specific Activity Limits, container activity will vary between negligible for most compressible or compacted wastes to less than 300 Ci/cc for reactor water purification demineralizer resins. The specific radionuclide content of the solid wastes is not available, as discussed in Section 11.4.2.1.

The filling of containers and the storage of radioactive solid wastes conform with 10CFR20 and 10CFR50 requirements. Packages meet shipping regulations of 49CFR171-178 and 10CFR71 as applicable.

Complete solidification and absence of free liquid is ensured by the implementation of a process control program and preoperational testing.

11.4.2.5 Temporary Onsite Storage Facilities

The processing area is located on the ground floor of the solid waste building. Wastes are packaged to allow for immediate shipment after processing. Therefore, no decay time is assumed in the given estimate of package contents and activity levels. Shipping containers with high radiation levels are moved into the solidification area until ready for shipment and placed in shields, if necessary, prior to shipment.

Storage capacity and storage time are based on anticipated operational factors. Based on routine storage of 2 weeks and minimum expected holdup and process times of 2 weeks, average decay time prior to shipping is 4 weeks. There is onsite storage capacity for up to 4 years, if required.

The waste storage area has been designed using conservative engineering estimates and calculations, based on available operating data, to handle the following unusual operational occurrences:

1. Approximately 2-1/2 times average processing rates for periods such as refueling

2. Startup conditions that require unusually frequent regeneration of deep bed condensate polishing system demineralizer beds

3. Continuous operation of the condensate polishing system for a main condenser tube leak of 0.3 gpm and the resultant processing of approximately 2.3 times the average amount of regenerant chemical wastes.

Under any of the above conditions, the storage area of the waste storage building can accommodate up to 4 years solid waste production. As noted above, storage delay time for radioactive decay is not considered a factor in sizing the storage area. Thus, the storage area is designed to handle waste production with offsite shipping delays from 2 to 10 weeks.

Components of low activity, such as contaminated tools, can be packaged in available or specially designed shipping containers and temporarily stored, if necessary, in the waste storage building. However, it is expected that these items will be shipped immediately rather than being stored.

11.4.2.6 Shipment

The shipment of radioactive solid waste conforms with 10CFR20 and 10CFR50 requirements and 10CFR71 and 49CFR171 through 178. Solid waste is transferred either directly to a licensed disposal contractor or to a common carrier for delivery to a licensed burial site, as appropriate.

Table 11.4-4 summarizes the annual number of shipments and shipped containers for the expected and design cases.

11.4.2.7 Protection Against Uncontrolled Releases

Protection against uncontrolled releases of radioactive material from the radioactive solid waste system is achieved through the use of alarms, interlocks, and a retaining structure.

The spent resin dewatering tank, evaporator bottoms tank, spent resin transfer pump, and spent resin recycle pump are located in curbed cubicles where any leakage will be retained. The walls and floors are suitably finished to facilitate decontamination. The spent resin dewatering pump, waste forwarding pump, disposal waste shipping container, resin fill and dewater head are located on the 24 feet-6 inches elevation of the waste solidification building. In the event of spillage of radioactive liquid in this area, the liquid is collected by a network of floor drains. The floor is pitched toward the floor drains. The drains are piped to the waste disposal building sumps (FSAR Figure 3.8-74) which are collected by the aerated drains system (FSAR Section 9.3.3) and forwarded to the radioactive liquid waste system (FSAR Section 11.2) for processing.

The spent resin dewatering tank and the evaporator bottoms tank are provided with low level alarms at the solid waste panel. For an alarm condition, a visual inspection by the operator will be conducted. 460.2

A radiation detection element is provided at the fill station to monitor the radiation levels adjacent to the container during the fill process. The setpoint of this detector will be set below the safe maximum limit established by D.O.T. (49CFR171-178). The high radiation interlock will automatically shut down and flush the process module if this predetermined level is exceeded.

If the shipping container is filled above the maximum safe levels, a level switch will automatically stop the process. If the pressure in the waste shipping container exceeds atmospheric pressure, a pressure switch in the container vent line will stop the process. The "overfill" or "overpressure" and "abnormal stop" lights will be energized on the solid waste panel for these conditions.

If the electrical circuitry from the control module to the shipping container becomes open due to a poor connection or any other reason, the respective system will stop. On loss of power, valves will fail shut and remain shut until the operator resets them.

The dewatering pump will automatically shut off in the event of very low pressure on the suction side. The system will automatically shut down after a line break of the waste fill line.

The single worst operator error or equipment failure would result in the spillage of spent resin and/or evaporator bottoms tank contents.

The drainage system in the waste solidification building is designed to handle such an event.

NRC Letter: May 3, 1983

Question No. Q460.18 (Sections 11.3)

Describe the hydrogen and oxygen monitor shown on Figure 11.3-2 (Sheet 2) at K-5. Provide the instrument and readout location, range, setpoint, sensor checks, and calibration. Will the instruments be nonsparking and capable of withstanding a hydrogen explosion as required by Acceptance Criterion 6 of SRP 11.3?

Response:

The hydrogen analyzer is a Comsip Delphi model B1C-1B9E. The theory of operation is based on the differences of thermal conductivity. An indicator is provided at the analyzer at elevation 48 feet-2 inches in the auxiliary building and a remote readout is provided at the local gaseous waste panel at elevation 43 feet-6 inches in the auxiliary building.

The range of the instrument is 80-100 percent H_2 with a setpoint at approximately 96 percent decreasing H_2 concentration. Channel checks will be performed daily with quarterly calibrations.

The oxygen analyzer is a Comsip Delphi model J1C-1B6T. The theory of operation is based on the paramagnetic property of oxygen acting on the instrument magnetic field and measured by photocells. Readout and location is the same as the hydrogen analyzer.

The range of the instrument of 0-4 percent O_2 with a setpoint at approximately 0.2 percent O_2 concentration. Channel checks will be performed daily with quarterly calibrations.

Both analyzers are explosion-proof and purchased under Class 1, Division 2, Group B case requirements. The oxygen analyzer is nonsparking. The remote possibility of a hydrogen explosion would be contained by the hydrogen analyzer.

NRC Letter: May 3, 1983

Question No. Q460.19 (Sections 11.5)

Section 11.5.2.5 refers to the use of glass sampling bulbs. Experience at many nuclear power plants indicates that it is not prudent to use glass containers for collecting and transferring radioactive samples due to breakage of the container. Justify your position for using glass sampling bulbs.

Response:

Refer to revised FSAR Section 11.5.2.5. Glass sampling bulbs are not used for collecting or transferring radioactive samples.

calibrations will be conducted using isotopic sources, electronic signals, or a combination of both.

11.5.2.5 Sampling

Section 9.3.2 discusses the various process and effluent samples taken periodically for chemical and radiochemical analysis.

Table 11.5-3 lists liquid process and effluent samples to be taken periodically and monitored for radioactivity. Those not covered in Section 9.3.2 are included in the individual system designs. Sampling of these fluid systems is via local sampling connections, e.g., the fuel pool cooling and purification system (Section 9.1.3). The Technical Specifications (Chapter 16) and Section 9.3.2 describe various periodic liquid samples analyzed for gross beta-gamma activities, including the sampling frequencies. Section 11.2 gives the expected and design concentrations of radionuclides expected in the various periodic process and effluent liquid samples.

Prior to collecting a sample, liquid sample lines are purged of stagnant water and undissolved solids for a sufficient time to ensure that a representative sample is obtained.

Sample taps suitable for connection to a sampling chamber are provided at all off-line process monitors to obtain a sample for laboratory gamma spectrum analysis. This analysis provides the ratio of nuclides needed to calculate the specific radionuclide releases required by Regulatory Guide 1.21.

Gas samples are collected in a stainless steel sample vessel with valves on each end. After adequate purging of the sample vessel, the gas sample is collected by closing valves at both ends of the sample vessel. 460.19

11.5.3 Effluent Monitoring and Sampling

In accordance with the requirements of General Design Criterion 64 (Section 3.1.2.64), each effluent discharge path is continuously monitored for radioactive effluents resulting from normal operations, including anticipated operational occurrences and from postulated accidents (Figure 11.5-1). Each effluent monitor alarms if the radionuclide concentration exceeds a predetermined level.

The ventilation vent high range monitor is capable of monitoring all postulated primary system accident releases through normal gaseous effluent paths.

11.5.4 Process Monitoring and Sampling

The following monitors automatically terminate discharge to the environment from their respective streams if the radionuclide concentration exceeds a predetermined level:

1. Fuel drop monitors

NRC Letter: May 3, 1983

Question No. Q471.23

Based on the criteria in C.1.e of R.G. 8.10, the RPM should have sufficient authority to enforce safe plant operation. It is the Staff's position that to ensure the RPM's ability to communicate promptly with an appropriate level of management about halting an operation he deems unsafe, he should report directly to the Station Superintendent. Your organization description in the FSAR should be revised to reflect this position.

Response:

Millstone Administrative Control Procedure ACP 6.02, Maintenance of Occupational Radiation Exposures ALARA, Section 4.7 states: "Stop Work Authority. The authority to be exercised by Senior Station Management, on-the-job supervision, job/task department heads, job/task leaders, or health physics staff when radiological conditions/job practices indicate that to proceed will or may result in conditions that violate NRC regulations, station procedures, or ALARA controls specified for the job."

Hence, by procedure, not only the RPM, but the entire health physics staff has stop work authority. Based on 13 years of operational experience at Millstone, it is apparent that the RPM does have sufficient authority to enforce safe plant operation. The stop work authority of the Health Physics Department is well established and is accepted by station and contractor personnel.

Therefore, there is no need to revise the organization as presented in Section 13.1 of the FSAR. This organization has operated effectively at Millstone Units 1 and 2.

NRC Letter: May 3, 1983

Question No. Q471.24

Indicate (and provide the resumes for) the individuals named as RPM and his backup. In accordance with 12.5.II.A of the SRP, the RPM should meet the qualifications specifications of R.G. 1.8. Also it is the Staff's position that the individual who will act as RPM in the RPM's absence (e.g., while on vacation), should have at least a B.S. degree in science or engineering 2 years experience in radiation protection, 1 year of which should be nuclear power plant experience, 6 months of which should be onsite. In addition, your FSAR should be changed to address the qualification for health physics technicians as specified in ANSI N18.1.

Response:

The RPM for Millstone Unit 3 is John P. Kangley, who became Radiological Services Supervisor in November 1982. His backup is Benito L. Granados, Health Physics Supervisor. The resumes of both are provided in Section 13.1 of the FSAR. Both meet or exceed the qualifications specified in Regulatory Guide 1.8. The requirements for personnel filling these two positions are specified in Section 12.5.1 of the FSAR.

Refer to revised FSAR Section 12.5.1 for the qualifications for Health Physics Technicians.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 Organization

The health physics program is established to provide an effective means of radiation protection for permanent and temporary employees and for visitors at the station. To provide an effective means of radiation protection, the health physics program incorporates a philosophy from management (Section 12.1.1); employs qualified personnel to supervise and implement the program (Section 13.2); provides appropriate equipment and facilities; and utilizes written procedures designed to provide protection of station personnel against exposure to radiation and radioactive materials in a manner consistent with Federal and State regulations (Section 13.5). The health physics program is developed and will be implemented through the applicable guidance of Regulatory Guides 8.2, Revision 0; 8.8, Revision 3; and 8.10, Revision 1.

The health physics department will implement and enforce the health physics program. The health physics program at Millstone 3 is developed and will be implemented as established for Millstone 1 and 2. Common to Millstone 1, 2, and 3, the radiological services supervisor, who reports directly to the station services superintendent, directs the health physics supervisor and each unit-specific ALARA Coordinator.

Reporting directly to the health physics supervisor is a health physicist. Also reporting directly to the health physics supervisor is a radiation protection supervisor who is assigned to each unit and the Health Physics Services group. An assistant radiation protection supervisor, reporting directly to the unit radiation protection supervisor, will direct the health physics technicians. The ultimate responsibility of the health physics program lies with the station superintendent.

The radiological services supervisor shall meet or exceed the qualification for radiation protection manager, as specified in Regulatory Guide 1.8, Revision 1. The health physics supervisor shall meet or exceed the qualifications for radiation protection manager in Regulatory Guide 1.8, Revision 1.

Additional information on the qualifications and experience of the health physics personnel can be found in Section 13.1.1.

Health physics technicians shall meet or exceed the qualifications specified in ANSI N18.1 471.24

The chemistry department is responsible for measuring the radioactive content of all gaseous and liquid effluents from the site in accordance with the requirements of the Environmental Technical Specifications and 10CFR20.

The chemistry supervisor reports directly to the radiological services supervisor.

The health physics department coordinates with all station, corporate, and contractor organizations to provide health physics coverage for all activities that involve radiation or radioactive material. The health physics department is organized to provide the following services:

1. Preparation and implementation of health physics procedures for routine and nonroutine activities associated with the operation, maintenance, inspection, and testing at the station

2. Compliance with regulatory limits for maximum permissible dose limits and contamination levels

3. Maintenance of a personnel radiation dosimetry program and dosimetry records

4. The surveying of station areas, maintenance of survey records, and the posting of survey results for daily activities within the station

5. Assistance in the station training program by providing specialized radiation protection training

6. Procurement, maintenance, and calibration of radiation detection instruments and equipment for assessment of the radiation areas

7. Procurement, maintenance, and issuance of protective clothing and equipment

8. Assistance in the shipping, storage, and receiving of all radioactive material to assure compliance with regulatory requirements

9. Assistance in the decontamination of personnel, equipment, and facilities

10. Preparation, maintenance, and issuance of the required regulatory, station, and personnel reports that are associated with radiation or radiation exposure

11. Preparation, maintenance, and implementation of the respiratory protection program

It is a policy of the Northeast Nuclear Energy Company (NNECo.) to keep personnel radiation exposure within the applicable regulations, and beyond that, to keep it as low as reasonably achievable.

12.5.2 Equipment, Instrumentation, Facilities

The criteria for purchasing the various types of portable and laboratory equipment used in the health physics and chemistry departments is based on several factors. Portable survey and

laboratory radiation detection equipment is selected to provide the appropriate detection capabilities, ranges, accuracy and durability.

NRC Letter: May 3, 1983

Question No. Q630.2 (Chapter 13)

Discuss the program which will provide the training to Reactor Operators and Senior Reactor Operators in the following areas:

a. Recognition of emergency conditions

b. Classification of observed emergency conditions in accordance with the Emergency Classification System

c. Notification of emergency to offsite authorities

d. Recommendation of protective actions to offsite authorities

e. Direction of station staff to take protective actions.

(Ref. NUREG-0800, Sections 13.2.1.I.B.1 and 13.2.1.II.1.b)

Response:

The Chemistry and Health Physics course includes the topic "Emergency Plan" where the Reactor Operators and Senior Reactor Operators are trained on emergency plans and specifically the Millstone Site Emergency Plan. The Procedures course provides training in the use and implementation of Millstone Unit 3 procedures. Training in the following areas is provided by the indicated procedures training topic(s):

a. Recognition of emergency conditions - Abnormal Condition Procedures, Emergency Operating Procedures and specifically Incident Assessment and Classification.

b. Classification of observed emergency conditions in accordance with the Emergency Classification System - Incident Assessment and Classification.

c. Notification of emergency to offsite authorities - Emergency Plan Implementing Procedures.

d. Recommendation of protective actions to offsite Authorities - Emergency Plan Implementing Procedures.

e. Direction of station staff to take protective actions - Emergency Plan Implementing Procedures.

NRC Letter: May 3, 1983

Question No. Q630.3 (Chapter 13)

Provide the outlines of the courses, Fundamentals of Nuclear Training and Nuclear Plant Training (Ref. NUREG-0800, Section 13.2.1.I.B.1).

Response:

Training in Fundamental of Nuclear Training and Nuclear Plant Training are provided by the courses included in the Training Program Description, FSAR Section 13.2.1D. This training is provided to the personnel (position titles) indicated in FSAR Table 12.2-2 Training Category column D. The subject matter, duration and the organization teaching each course in this training are provided in the attachments to this response.

Course Description:	Reactor Theory
Duration:	2 Weeks
Taught By:	Millstone Station Training Department

Subject Matter:

Atomic Nature of Matter
Electron and Nuclear Structure
Nuclear Mass and Stability Concepts
Radiation and Radioactive Decay
Decay Modes and Decay Rates
Nuclear Reactions and Neutron Interactions
Nuclear Cross Sections
Macroscopic Cross Section, Neutron Flux and Neutron Reaction Rate
Binding Energy
Fission Process
Characteristics of Fission
Prompt and Delayed Neutrons
Neutron Thermalization and Moderation
Neutron Cycle and Neutron Balance
The Four and Six Factor Formulas
Characteristics of the Six Factors
K_{eff}
Reactivity
Reactivity Coeffecients
Moderator Temperature Coefficient
Doppler Effect
Doppler Coefficient
Void and Pressure Coefficients and Reactivity Defects
Isothermal Temperature Coefficient and Defect
Power Coefficient and Defect
Excess Reactivity and Chemical Shim
Control Rods
Factors Affecting Rod Worth
Differiential and Integral Rod Worths
Fission Product Poisons and Samarium
Xenon Production and Removal
Xenon Behavior Following Transients
Neutron Sources
Subscritical Multiplication, ICRR, and 1/M Plots
Power Rate Changes
Period Equation
Use of Period Equation
Time-In-Life Effects
PWR Response
Shutdown Margin
Estimated Critical Position

Course Description:	Heat Transfer, Thermodynamics and Fluids Flow
Duration:	2 Weeks
Taught By:	Millstone Station Training Department
Subject Matter:	Properties of Working Fluids Energy, Work and Heat Energy, Enthalpy and Power Phases of Matter First Law of Thermodynamics Applications of the General Energy Equation Second Law of Thermodynamics Entropy Steam Tables Superheated Steam Tables Liquid Heat Capacity Mollier Diagram Heat Transfer Modes Conduction Heat Transfer Convection Heat Transfer Combined Heat Transfer Radiation Heat Transfer Boiling Heat Transfer Carnot Cycle Rankine Cycle Power Plant Components Plant Efficiency Heat Exchangers S/G and Condenser Core Power Operation Power Operation in the Reactor Core Fuel Rod Temperature Profiles Reactor Heat Transfer Reactor Power Limits Safety Limits Core Thermal Considerations Properties of Fluids Gases Basic Fluid Flow Types of Flow Bernoulli's Equation Fluid Friction Head Loss Flow Measurement Pumps Pump Laws Pump Head Eductors Heat Balance System Design

Core Flow
Natural Circulation

Course Description:	Nuclear Steam Supply Systems (NSSS)
Duration:	8 Weeks
Taught By:	Westinghouse Training Department Millstone Station Training Department
Subject Matter:	Reactor Coolant System Pressurizer and Relief Tank Reactor Vessel Reactor Internals Nuclear Fuel Nuclear Coolant Pump Steam Generator Chemical and Volume Control System Reactor Makeup Systems Boron Recovery System Boron Thermal Regeneration Residual Heat Removal Fuel Handling Introduction to Engineered Safety Features Emergency Core Cooling System Containment Quench Spray Auxiliary Feedwater Component Cooling Water Introduction to I&C Rod Control System Excore NIS Rod Position Indicating System Pressurizer Level Control System Pressurizer Pressure Control System Temperature Indicating System Steam Generator Water Level Control Steam Dump System Reactor Protection Trips Reactor Protection System Protection System Logics Control Systems Review Transient Analysis Instrument Failure Analysis Incore Instrumentation Systems Technical Specifications Accident Analysis

Course Description: Balance-of-Plant (BOP)

Duration: 5 Weeks

Taught By:
Millstone Station Supervisory Staff
NUSCO Staff Engineers
Millstone Station Training Department

Subject Matter:
General Onsite AC Power
345 kV
Main Generator and Exciter
Generator Voltage Regulator and Exciter
6.9 kV
4160 V
480 V Load Center
480 V MCCs
120 V dc
120 V ac Vital and Nonvital
Diesel Generator and Support Systems
Diesel Generator Sequencer
Main Steam
Steam Generator Blowdown
Condensate
Condensate Makeup and Drawoff
Feedwater
Feedwater Chemical Feed
Feedwater Heater Drains and Vents
Condensate Demineralizers
Condensate Air Removal
Cathodic Protection
Circ Water/Traveling Water Screens/Condenser Tube Cleaning
Vacuum Priming
Service Water
Chlorination
Auxiliary Boiler
Auxiliary Steam and Condensate
Hot Water Heating and Preheating
Turbine
Moisture Separator Reheater and Drains
Gland Steam and Exhaust
Turbine Lube Oil
Generator Seal Oil
Electro Hydraulic Control
Generator Stator Liquid Cooling
Generator Hydrogen Cooling
Generator CO_2 and Hydrogen
Plant Air Systems
Turbine Plant Component Cooling
Turbine Plant Sampling
Domestic Water
Water Treatment

Heat Tracing
Fire protection
Nitrogen
Waste Oil Disposal
Service Building HVAC
Control Building HVAC
Turbine Building Ventilation
Main Steam Valve Building Ventilation
Auxiliary Boiler Room Ventilation
Diesel Generator Building Ventilation
Reactor House Ventilation
Screen Plant Gaseous Drains/Boron
Recovery System
Reactor Plant Aerated Drains/Radioactive
Liquid
Waste Systems
Radioactive Gaseous Waste
Radioactive Solid Waste
Reactor Plant Chilled Water
Condensate Demineralizer Liquid Waste
Condensate Demineralizer Component Cooling
Charging Pump/Safety Injection Pump Cooling
Neutron Shield Tank Cooling
Reactor Plant Sampling
Post Accident Sampling
Containment Vacuum
Containment Ventilation
Hydrogen Recombiner Building Ventilation
Auxilialry Building HVAC
Waste Disposal Building HVAC
Spent Fuel Building and ESF Building
Ventilation
Enclosure Building Ventilation (SLCRS)

Course Description: Chemistry and Health Physics

Duration: 1 Week (Training Shift)

Taught By: Millstone Station Training Department

Subject Matter:

Chemistry
Chemistry Fundamentals
Corrosion
Radiation Chemistry
Radioactive Contaminants of the Reactor Coolant
Water Treatment
Secondary Chemistry
Hazardous Chemicals - Gases and Liquids
Hazardous Chemicals - Solids
Sampling Techniques
Analytical Techniques
Radiochemistry
Technical Specifications and Plant Chemistry Monitoring Program

Health Physics
Terminology and Decay Equations
Units of Exposure and Dose
10CFR20
Radiation Dose Limits
HP Procedures
Properties and Interaction of Radiation
Biological Effects of Radiation
Internal Exposure
Time, Distance and Shielding
Dose Rate and Shielding
Detector Principles
Dosimeters and Survey Instrumentation
Contamination Sources
Respirators
Emergency Plan

Course Description:	Transient and Accident Analysis
Duration:	1 Week (Training Shift)
Taught By:	Millstone Station Training Department
Subject Matter:	Moderator Temperature Coefficient Fuel Temperature Coefficient Void Coefficient Power Coefficient Interaction of Feedback Mechanisms Delayed Neutron Effects Step Load Changes Ramp Load Changes Entry into the Power Range Purpose of Accident Analysis Accident Analysis Methodology Trip Setpoints Relationship between Safety Analysis and Technical Specification Relationship of the Engineered Safety Features System to Safety Analysis 10CFR Limits Classification of Accidents Increase in Heat Removal by the Secondary System Decrease in Heat Removal by the Secondary System Decrease in Reactor Coolant System Flow Rate Reactivity and Power Distribution Anomalies Increase in Reactor Coolant Inventory Decrease in Reactor Coolant Inventory Radioactive Releases from a Subsystem or Component Anticipated Transient without Trip

Course Description:	Procedures
Duration:	2 Weeks (Taining Shift)
Taught By:	Millstone Station Supervisory Staff Millstone Station Training Department
Subject Matter:	General Operating Procedures Plant Heatup Reactor Startup Plant Startup Load Changes Three Loop Operation Plant Shutdown Reactor Shutdown Reactor Cooldown Reactivity Calculations Refueling Operation New Fuel Handling Spent Fuel Handling Plant Cooldown from Outside Control Room

System Operating Procedures

Abnormal Condition Procedures
Condenser Tube Leak
Dropped Control Rod
Steam Generator Tube Leak
Inadvertant ECCS Initiation
Loss of One Protective System Channel
Steam Generator Chemistry
Loss of Shutdown Cooling (RHR)
Failure of Reactor Coolant Pump Seal
Malfunction of Rod Drive Control System
Turbine Generator Trips
High Activity in the Reactor Coolant
Flooding
Loss of DC Bus

Emergency Operation Procedures
Incident Assessment and Classification
Emergency Plan Implementation
Reactor Trip or Safety Injection
Loss of Reactor Coolant
Loss of Secondary Coolant
Steam Generator Tube Rupture
Loss of Condenser Vacuum
Loss of Component Cooling (RPCCW)
Natural Occurences
Fire
Loss of Instrument Air
Control Room Evacuation
Emergency Boration
Earthquake
Loss of Service Water
Loss of One DC Bus (125 volt)
Loss of Containment Vacuum/Integrity
Feedline Rupture
Irradiated Fuel Damage While Refueling
Anticipated Transient Without SCRAM
Loss of All AC Power
Unisolable SGTR
Loss of RCS Depressurization Capability
for SGTR
Loss of Secondary Side Cooldown Capability
for SGTR
Improper SI Operation During SGTR
Loss of Cold Leg Recirculation
Multiple Steam Generator Blowdown
Secondary Side Rupture Without High Head SI
Maintenance of Subcriticality
Maintenance of Reactor Coolant System
Inventory
Maintenance of Core Cooling

Maintenance of Reactor Coolant Inventory
Maintenance of a Heat Sink
Maintenance of Containment Integrity

Surveillance Procedures

Station Procedures

Course Description: Mitigating Core Damage

Duration: 1 Week (Training Shift)

Taught By: Millstone Station Training Department

Subject Matter:

Core Cooling Mechanics
- Natural Circulation
- Reactor Coolant Pumps
- Emergency Core Cooling System (ECCS)

Potentially Damaging Operating Conditions
- Kewaunee Nuclear Plant - Partial Blackout and Natural Circulation
- St. Lucie Unit 1 - Loss of Reactor Coolant Pump Cooling and Natural Circulation Cooldown
- David Besse and TMI-2 Transients
- Crystal River Unit 3 - Partial Loss of Non-Nuclear Instrument System Power Supply During Operation

Modes of Natural Circulation/Michelson's Concerns
- Modes of Natural Circulation
- Michelson's Concerns

Incore Instrumentation
- Incore Thermocouples
- Incore Flux Mapping System

Excore Instrumentation
- Excore Nuclear Instrumentation
- Excore Nuclear Instrumentation and Analysis of Reactor Vessel Water Level at TMI

Vital Instrumentation
- TMI - Lessons Learned on Vital Instrumentation
- Other Postulated Instrumentation Problems during an Accident Condition

Reactor Chemistry
- Sources of Radioactivity
- Sample Calculations of Activity Levels in Containment and Radiation Levels Outside Containment
- Radiochemical Consequences of TMI
- Corrosion Effects in Containment Following a LOCA
- Post Accident Sampling System

Radiation Monitoring

Behavior of G-M Detectors in High Radiation Fields

Behavior of Scintillation Detectors in High Radiation Fields

Expected Accuracy of Monitors at Different Locations

Radiation Monitor Failure Modes

Anticipated Radiation Hazards

Anticipated Paths for Radioactive Materials

Radiation Hazards to Operating Personnel

Gas Generation

Sources of Hydrogen

Hydrogen Combustion

Hydrogen Phenomena at TMI-2

TMI-2 Hydrogen Generation Calculation Example

Control of Hydrogen in the Primary Coolant and in Containment

Estimate of Core Damage

Source Description:	Facility License and Technical Specifications
Duration:	1 Week (Training Shift)
Taught By:	Millstone Station Supervisory Staff Millstone Station Training Department
Subject Matter:	Definitions: Safety Limits and Limiting Safety Systems Settling (LSSS) Bases for Safety Limits and LSSS Limiting Conditions for Operations (LCO) Surveillance Requirements Design Features Administrative Controls

NRC Letter: May 3, 1983

Question No. Q630.6 (Chapter 13)

Discuss the certifications completed pursuant to Sections 55.10 (a)(6) and 55.33 (a)(4) and (5) of 10CFR Part 55. Provide the title of the individual who will certify the eligibility of individuals for licensing or renewal of license (Ref. Enclosure 1 of H.R. Denton's March 20, 1980 letter, Section A.3).

Response:

The Training Supervisor evaluates all candidates for license. He reviews, recommends and forwards applications to the Station Superintendent for site approval. All applications for license are forwarded to the Senior Vice President, Nuclear Engineering and Operations for final corporate approval and certification in accordance with Sections 55.10 (a) (6) or 55.33 (a) (4) and (5) of 10CFR Part 55.

NRC Letter: May 3, 1983

Question No. Q630.7 (Chapter 13)

Provide a commitment to comply with the following TMI-related requirements as specified in Item I.A.2.1 of NUREG-0737:

a. As an operating license Applicant, Millstone 3 is not subjected to the one year experience requirements for cold license SRO candidates. However, after one year of station operation, we will require Millstone 3 to comply with the one year experience requirement for hot license SRO Applicants.

b. The requirement for three months on shift experience for control room operators and SRO candidates as an extra person on shift is not required for cold license candidates and, hence, is not applicable to Millstone 3. However, we will require Millstone 3 to comply with this requirement for hot license candidates after three months of station operation.

Response:

Refer to FSAR Sections 13.2.2.3.1 and 13.2.2.3.2.