original with figs.

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 666-6911

March 31, 1983

Docket No. 50-423
AEC-MP3-308
B10745

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) D. G. Eisenhut letter to W. G. Counsil, Acceptance Review of the Application for an Operating License for Millstone 3, dated, January 31, 1983.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3:
Response to the Requests for Additional Information
that Resulted from the Acceptance Review

Attached are our responses to your requests for additional information. Attachment 1 contains our responses to Enclosures 2 and 3. The actual references requested in Enclosure 3, questions E 290.1, E 291.11, E 291.13, E 291.14, E 291.15, E 291.16 and E 291.17, are being forwarded under a separate transmittal letter as of the same date as this transmittal. Attachment 2 contains our responses to Enclosures 4-11. The responses that have been provided are as they will appear in the Amendment 2 to our OL application. On or before April 15, 1983, the required 60 copies of Amendment 2 will be forwarded to you for insertion into your FSAR/ER sets.

Since Enclosures 4-11 have not been assigned a question number, these "requests for additional information/responses" will not be incorporated into Amendment 2 as part of the FSAR or ER volumes dedicated to questions/responses. FSAR or ER text changes that are a result of Enclosures 4-11 will appear in Amendment 2 or subsequent amendments. In cases where we are unable to provide a complete response at this time we have provided a commitment date.

B001 s ,/40
ADER. Dist
SEND DRWGS to: PM

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PDR ADOCK 05000423
A PDR

As discussed in a telephone conversation betweeen our Mr. J. M. Powers and your Ms. E. L. Doolittle on March 30, 1983, because of printing lead times required to correct a typographical error, only one (1) set of marked-up Figures 12.3-1 through 12.3-4 and 12.3-6 through 12.3-9 are being forwarded. (They are attached to one (1) of the three (3) signed originals.) These Figures constitute the FSAR change made as a result of Question 471.2. The thirty-nine (39) additional required copies will be forwarded on April 5, 1983.

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

C. Frederick Sears

By: C. F. Sears
Vice President
Nuclear and Environmental Engineering

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

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

Lisa E Dolce

Notary Public

My Commission Expires March 31, 1988

Attachment 1

Responses to Enclosures 2 and 3

FSAR QUESTIONS

MILLSTONE NUCLEAR POWER STATION - UNIT 3
DOCKET NO. 50-423

NRC Question	FSAR Section	Keywords
100.1	1.9	Standard Review Plan
210.1	3.9	Tables 3.9B-2, -14 & -17
220.1	3.5.1.4	Missile protection structures and/or barriers
220.2	3.7.4	Inservice surveillance
220.3	3.8.1.1.4	Figure 3.8-20
220.4	3.8.3.4	Computer programs and load transfer and assumptions
220.5	3.8.4.1	Passageway figure
220.6	3.8.4.4	Design and analysis areas of SRP 3.8.4
220.7	3.8.4.8	Masonry walls
220.8	3.8.5.1	Waterproofing membranes
230.1	2.5.2.4	Earthquake ground motion
230.2	2.5.2.7	Probability of exceeding OBE
250.1	5.2.4.8	PSI/ISI
252.1	5.3.3	Reactor vessel designer and manufacturer
270.1	3.11B.3, 3.11N.3	Qualification program results
280.1	9.5.1	Fire protection program
281.1	6.1.1.2	Time-history of pH
281.2	10.4.6, 5.4.2.1	Secondary water chemistry program
311.1	2.2	Chemicals stored or transported near site
311.2	2.2	Missiles from hazardous materials

FSAR QUESTIONS (Cont)

NRC Question	FSAR Section	Keywords
311.3	2.2	Demographic Data in miles
311.4	3.5.1.6	Aircraft hazards
410.1	3.5.1.4	Tornado missile protection
410.2	3.6.1.3.3	Pipe break/crack analysis
410.3	5.2.5.4	Limiting leakage conditions
410.4	9.2.6	Effects of system leakage or storage tank failure
410.5	9.3.3	Prevention of transfer of containment fluids
410.6	10.4.9.4	AFWS pump tests
430.1	8.2.2	Critical transmission line
430.2	8.3.1.2	Equipment qualification tests
440.1	5.1.1	Notes to Figure 5.1-2
440.2	5.2.2.2	WCAP-7769
440.3	5.4.7.2.1	Notes to Figure 5.4-6
440.4	5.4.7.2.2	NPSH for the RHR pumps
440.5	6.3.2.5	Specific equipment arrangement
440.6	6.3.3.3, 15.6.5.1	Small and large-break LOCAs
440.7	15.0	Accident analyses
440.8	15.4.6	Accident of SRP 15.4.6
451.1	2.3.2.1	Fog
451.2	2.3.4.2	Relative concentrations, χ/Q
460.1	11.2.1	Peak flow rate of Aux Bldg Sump
460.2	11.4.2	Operator error/single equipment malfunction
460.3	11.4.2	Solid waste containers

FSAR QUESTIONS (Cont)

NRC Question	FSAR Section	Keywords
460.4	11.5.2	Range of radioactivity concentrations
471.1	12.1.2	Radiation protection plan
471.2	12.3.1	Zone boundaries
471.3	12.3.2	Design basis radiation level in count room
471.4	12.3.4	Area radiation and airborne radioactivity monitoring instrumentation
471.5	12.3.4	Radiation instrumentation
471.6	12.3.4	R.G.s 8.2, 8.8 and 8.12 and ANSI N13.1-1969
471.7	12.4	Inhalation exposure
471.8	12.5.3	R.G. 8.13
471.9	11.3.2	Concentrations of airborne radioactive material
480.1	6.2.1.1.3	Inadvertent operation of containment heat removal system
480.2	6.2.1.1.3	Limiting containment conditions
480.3	6.2.2.3	Containment sump design
480.4	6.2.3.3	Secondary containment function
480.5	6.2.5.2	Hydrogen analyzers
630.1	13.5.1	Initial test program

Question No. Q100.1 (Section 1.9) NRC Letter: January 31, 1983

Section 1.9, Standard Review Plan Documentation of Differences, states:

"This section will be transmitted to the NRC tentatively upon docketing of the Millstone 3 Operating License Application. The actual submittal date will be finalized in a meeting to discuss the details of the review schedule between the applicant and the NRC to be held in October 1982."

The applicant and NRC have agreed to a submittal date of February 1983 for Section 1.9. This section should conform to the following:

1. Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982, shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982, or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

2. The evaluation shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of the Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.

3. The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.

Response:

Section 1.9, Standard Review Plan Documentation of Differences, was transmitted in Amendment 1, dated February 28, 1983.

NRC Letter: January 31, 1983

Question No. Q210.1 (Section 3.9)

Several tables, for example, Tables 3.9B-2, 3.9B-14, and 3.9B-17, indicate that additional information will be supplied later. Either submit this information or a schedule for its submittal.

Response:

Results of these analyses will be provided 11 months prior to fuel load.

NRC Letter: January 31, 1983

Question No. Q220.1 (Section 3.5.1.4)

As per Regulatory Guide 1.70, the structures and/or barriers used for missile protection should be tabulated. The table should contain the following information:

1. Systems or components that are protected by the structure/barrier.

2. Concrete thickness and strength for walls, roofs, and floors used for missile protection and the curing time on which the strength is based.

Response:

Refer to Section 3.5.1.4, Amendment 2.

3.5.1.3.5 Turbine Overspeed Protection

The turbine control system is an electrohydraulic control (EHC) system that includes both digital and analog circuitry, electronic servo hardware, and hydraulic valve actuators.

The EHC provides a normal overspeed protection system and an emergency overspeed protection system to limit turbine overspeed. These two systems are essentially separate and independent. The normal overspeed protection system is part of the turbine load and speed control system and is designed to limit turbine overspeed without a turbine trip under all load conditions. The emergency overspeed protection system is part of the emergency trip system and is designed to trip the turbine if the turbine speed exceeds 110 percent of rated speed (Section 10.2).

3.5.1.3.6 Turbine Valve Testing

The main turbine generator control, main stop, intercept, and reheat stop valves are routinely tested (Section 10.2.3.6).

Control and main stop valves are tested one at a time, and as each test is completed, the valve is returned to its original position before the next valve is tested. Intercept and reheat stop valves are interlocked so that a pair of these valves in one crossover pipe is tested together. For this test, one pair of pipe is tested and the valves returned to the open position before the next pair is tested.

3.5.1.4 Missiles Generated by Natural Phenomena

Q220.1 The only credible missiles generated by natural phenomena are those generated by a design basis tornado. Those Category I structures designed to withstand the effects of tornado missiles and the systems and components thus protected are identified in Tables 3.5-1 and 3.5-2.

Q220.1 A minimum of 2 feet thickness of reinforced concrete having a minimum strength of 3,000 psi (28 day compressive strength) was used for walls, roofs, and floors designated as missile protection. The minimum reinforcing steel each way in each face of any square foot of wall or slab providing missile protection is 1.85 square inches.

Postulated missiles generated by the design basis tornado (Section 3.3.2) are listed in Table 3.5-13, which includes all parameters necessary to determine missile penetration. These missiles are considered capable of striking in any orientation.

Q410.1 Ventilation openings in the various facility buildings housing essential shutdown equipment are protected by reinforced concrete labyrinths.

There are no other design basis missiles resulting from flood or any other natural phenomena described in Section 2.2.3.

Question No. Q220.2 (Section 3.7.4) NRC Letter: January 31, 1983

Provide an additional section or reference that discusses inservice surveillance as per SRP 3.7.4 in NUREG-0800.

Response:

The inservice surveillance of seismic instrumentation will be discussed in the Technical Specifications, Chapter 16, which will be provided six months prior to fuel load.

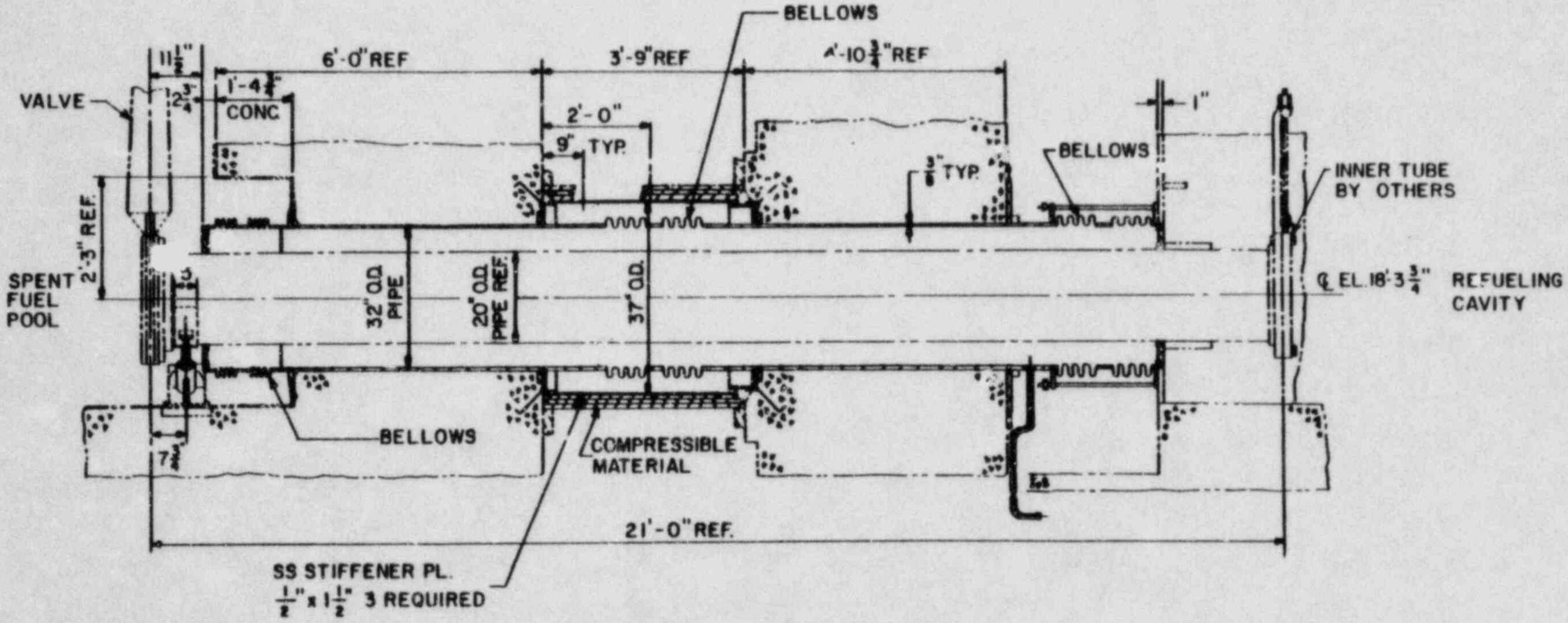
NRC Letter: January 31, 1983

Question No. Q220.3 (Section 3.8.1.1.4)

This section references Figure 3.8-20 that indicates it will be submitted later. Either provide this figure or state when it will be submitted.

Response:

Refer to Figure 3.8-20, Amendment 2.



ELEVATION LOOKING EAST

FIGURE 3.8-20
FUEL TRANSFER TUBE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: January 31, 1983

Question No. Q220.4 (Section 3.8.3.4)

Any computer programs that are utilized should be referenced and the manner of load transfer and assumptions should be discussed as per Section 3.8.3.4 of Regulatory Guide 1.70, Revision 3, and SRP 3.8.3 in NUREG-0800.

Response:

1. Computer programs are identified in Appendix 3A.

2. Regarding assumptions and load transfer, refer to FSAR Section 3.8.3.4: "Structural analyses are based on elastic behavior using commonly accepted principles of engineering mechanics appropriate to the geometry of the structure."

The amount of reinforcing steel required is determined in accordance with the procedures outlined in ACI 318 and the principal reinforcement patterns are located in the direction of tensile stresses. Bond and anchorage requirements of ACI 318 are complied with, and where biaxial tensile fields exist, the development lengths required by Section 12.5 of ACI 318 are increased by a minimum of 25 percent.

Structural steel is designed in accordance with the procedures outlined in the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, the Structural Welding Code, AWS D1.1, and the loading combinations given in Section 3.8.3.3.

The heat generated in the primary concrete shield wall due to gamma rays and neutrons emanating from the reactor core and due to gamma rays arising from neutron interactions in the primary shield wall is negligible and has no effect on the strength of the structure.

Computer programs that have been used in the design and analysis of structural elements can be found in Appendix 3A. Q220.4

3.8.3.5 Structural Acceptance Criteria

Design of interior concrete structures follows ACI 318, using the required strength section based on the strength design method. The basic criterion for concrete strength design is expressed as:

Required Strength ≤ Calculated Strength

All members and all sections of members are proportioned to meet this criterion. The required strength is expressed in terms of design loads, or their related internal moments and forces. Design loads are defined as loads which are multiplied by their appropriate load factors. Calculated strength is computed according to the provisions of ACI 318, including the appropriate capacity reduction factors. Capacity reduction factors are the same as those given in Section 9.2 of ACI 318.

Design of interior steel structures is based either on elastic working stress design methods using normal working stress levels given in Part 1 of AISC Specification or on the plastic design methods of Part 2 of AISC.

Section 3.7.2.8 describes the variations incorporated into the seismic analysis structural model to account for a cracked and an uncracked containment structure shell, for variations and uncertainties in subgrade shear modulus and spring constants, for virtual mass embedment, and for contact pressure distribution. Design of the internal structures is based upon the most conservative values resulting from these variations in assumptions and design parameters. Section 3.7.3.6 discusses differential seismic movement relating to interconnected components, systems, and equipment.

Question No. Q220.5 (Section 3.8.4.1) NRC Letter: January 31, 1983

This section states the drawing for the passageway will be supplied later. Either provide the figure or a date for its submittal.

Response:

The passageway has been eliminated and replaced with an open walkway. No figure is required.

Arrangement drawings for major Non-Seismic Category I structures are:

Title	Figure	
Service Building	3.8-72	(EM-5)
Turbine Building	3.8-73	(EM-3)
Waste Disposal Building	3.8-74	(EM-9)
Warehouse 5 and Millstone 2 Condensate Polishing Facility	3.8-75	(EM-50)
Auxiliary Boiler and Condensate Polishing Building	3.8-76	(EM-38)
Miscellaneous Yard Tankage	1.2-2 Plot Plan	

Q220.5

Plant grade is approximately el 24 feet-0 inch.

Containment Enclosure Building (Seismic Category I)

The containment enclosure building, a cylindrical steel framed structure with uninsulated metal siding and builtup roofing over insulated metal roof deck, envelops the containment structure above grade. It has a diameter of 156 feet, a height above grade of approximately 166 feet, and is supported on the containment structure.

The containment structure below grade is surrounded by ribbed fiberglass sheets, which extend from the top of the containment mat to above grade. This arrangement provides vertical ventilation channels which vent directly into the containment enclosure building or into the abutting buildings. The corrugated fiberglass sheet is covered by a 2-inch-thick layer of compressible material a 40-mil thick waterproofing sheet membrane, and then one more layer of 2-inch thick compressible material. The waterproofing membrane encloses the containment substructure and extends beneath the mat and above plant grade. The enclosure building design incorporates horizontal and vertical sliding joints to ensure that the integrity of the containment enclosure building will be maintained during maximum possible pressure transients of the containment structure under DBA conditions. It assures the proper performance of the supplementary leak collection and release system, as described in Section 6.5.1.

To provide the required degree of air tightness and to maintain a partial vacuum within the containment enclosure building, the metal siding, metal deck side joints, and end laps will have two continuous lines of caulking at all joints. Neoprene gaskets and sheets will be used to provide a flexible seal between the containment enclosure and the other buildings. The base of the containment enclosure building wall will be sealed in a similar manner to the subgrade waterproofing membrane.

The structural framing, but not the siding or roofing, is designed to remain intact under tornado loading. In addition, the design of the building is based on the structural loads being transferred through

NRC Letter: January 31, 1983

Question No. Q220.6 (Section 3.8.4.4)

This section should address the design and analysis areas identified in SRP 3.8.4 in NUREG-0800.

Response:

See Amendment 2 to FSAR Section 3.8.4 for a discussion of the design and analysis areas that are identified in SRP 3.8.4 of NUREG-0800.

Concerning the design report format, as discussed in FSAR Table 1.9-2, SRP 3.8.4 (Item 2), the Millstone 3 design information is not in the design report format of Appendix C to SRP 3.8.4. However, the material described in Appendix C can be found in the design criteria and design calculations which are contained in auditable files located at the Millstone 3 site.

Loads from the spent fuel racks will be provided in FSAR Section 9.1.2 in an amendment in June 1983 to fully address Appendix D of SRP 3.8.4 in NUREG-0800.

3.8.4.2 Applicable Codes, Standards, and Specifications

Codes, standards, specifications, and NRC regulatory guides used in establishing design methods and material properties for Seismic Category I concrete and steel structures other than the containment are given in Section 3.8.1.2.

3.8.4.3 Loads and Loading Combinations

All Seismic Category I structures other than the containment structure mat, shell, and dome are designed for the loads and load combinations in Table 3.8-3. Section 3.8.4.5 describes allowable stress levels.

Q220.6 For the spent fuel pool, the effects of loads imparted to the structure by the spent fuel racks as well as the effects of hydrostatic and seismically induced hydrodynamic loads are considered in the design. The spent fuel pool walls and mat are also designed for thermal effects based on temperatures indicated in Figures 3.8-79 and 3.8-80.

3.8.4.4 Design and Analysis Procedures

In general, design and analysis procedures conform to the requirements of the ACI 318-71 Code and the AISC specification, 7th Edition, for the Design, Fabrication and Erection of Structural Steel for Buildings except as noted in Section 3.8.4.3. Structural

Q220.6 analyses are based on elastic behavior using commonly accepted principles of engineering mechanics appropriate to the geometry of the structure.

The boundary conditions assumed for structural elements manual under design are based on the stiffness of the elements into which these elements are framed.

Material quality control procedures, as referred to in Section 3.8.4.6, ensure that minimum strength material requirements are achieved.

Seismic accelerations on the structures are determined by dynamic analysis as described in Section 3.7.3.4. Forces are determined and then applied statically in the design of the structures. The analytical techniques used to determine the forces are given in Section 3.7.2.

A rattle space is provided between all independent Seismic Category I and nonseismic structures so as to prevent their interaction during an earthquake.

Tornado loads (described in Section 3.3.2) include wind force loads and the loads from tornado generated missiles. Section 3.5.3 gives

Q220.6 tornado missile impact effects.

Computer programs which have been used in the design and analysis of structural elements are identified in Appendix 3A. Q220.6

3.8.4.5 Structural Acceptance Criteria

Seismic Category I structures, as identified in Table 3.2-1, are designed to withstand the loading combinations given in Table 3.8-3.

Concrete structures are designed by the strength design method of ACI 318-71.

The basic criterion for strength design is expressed as:

Required Strength ≤ Calculated Strength

All concrete members and all sections of concrete members are proportioned to meet this criterion. The required strength is

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:

Evaluation of Category I masonry walls is currently being finalized and will be addressed in an amendment in June 1983 (See FSAR Section 1.9, SRP 3.8.4).

NRC Letter: January 31, 1983

Question No. Q220.8 (Section 3.8.5.1)

This section should reference or discuss the effect of the waterproofing membranes on the capability of the foundation to transfer shears.

Response:

Refer to Section 3.8.5.1, Amendment 2.

Technical justification: Sliding stability analysis considered friction factors (μ) as noted below:

Bldg	(Concrete to μ Founding Soil or Rock)	Ref. Calc.	
Auxiliary	0.65	Book 5 CAB	Pg. C35.305
Engr. safety features building	0.65	5 CAM	C32.39
Control	0.4	5CBX	C10.4
Service	0.7	5CH	C7.75

Between concrete and butyl rubber membrane ≥ 0.7[1,2]
Sliding stability is unaffected by membrane

References

1. Higdon and Stiles, Engineering Mechanics 3rd Edit. Prentice-Hall, Inc. 1968.

2. Beer and Johnson, Mechanics for Engineers 2nd Edit. McGraw-Hill Book Co. 196.

The containment enclosure building is supported entirely on the containment structure and has no foundations. Figure 3.8-61 shows typical details of the interface between the enclosure and ground/adjacent structures.

Significant amounts of groundwater are not expected. Figure 2.5.4-27 shows the design levels for groundwater. No Seismic Category I dewatering system is required. However, the following features have been incorporated to prevent seepage of groundwater into portions of structures below the piezometric surface: 1

1. All structures, except the containment, have waterstops installed at construction joints below grade.

2. The containment substructure is encased with a waterproof membrane to el 25 feet-0 inches. In the unlikely event the membrane should leak, a drainage system is provided within that membrane and connected to sumps located in the engineered safety features building.

3. The service, control, auxiliary, and engineered safety features building substructures are encased with a waterproof membrane to el 25 feet-0 inches and have drainage systems located under the mat of each building. These run into sumps for collection and then discharge. The coefficient of friction between the membrane and the concrete is equal to or greater than that between the concrete below the membrane and the soil or rock. Sliding stability is therefore not affected by the presence of the membrane. Q220.8

4. The fuel, waste disposal, and office buildings are provided with perimeter and substructure drains.

3.8.5.2 Applicable Codes, Standards, and Specifications

Section 3.8.3.2 contains the codes, standards, specifications, and NRC regulatory guides used in establishing design methods and material properties for foundations and concrete supports.

3.8.5.3 Loads and Loading Combinations

Foundation design is based upon appropriate loading combinations. The loads and loading combinations given in Section 3.8.1.3 are used for the containment foundation design. The loads and loading combinations given in Table 3.8-3 are used for the design of all other Seismic Category I foundations.

In addition to the above loads and load combinations, the following are used to check against sliding and overturning due to earthquakes, winds, tornadoes, and the design basis flood:

Question No. Q230.1 (Section 2.5.2.4) NRC Letter: January 31, 1983

Per Regulatory Guide 1.70, for the maximum potential earthquake ground motion at the site should be determined assuming seismic energy transmission effects are constant over the region. The set of conditions describing the occurrence of the potential earthquake that would produce the largest vibratory ground motion at the site should be defined.

Response:

The maximum potential earthquake ground motion at the site has been determined by assuming that the seismic energy transmission effects are constant over the region. Earthquakes occuring in the region are assumed to be associated in the tectonic province in which they occur. The largest historical earthquake in the tectonic province is then assumed to occur in that province at that point nearest to the site.

The site is located in the southeast New England/Maritime Tectonic Province. The largest earthquake in this province was Intensity VI (MM). The New England Tectonic Province and the Coastal Plain Province are close to the site. The largest earthquake in both provinces was Intensity VII (MM). As the smallest distance from the site to these provinces is about 10-20 km, it has been assumed that the maximum earthquake potential at the site is Intensity VII (MM). The assumption of constant seismic energy transmission effects has been utilized by Howell and Schultz (1975) to develop attenuation relations for the eastern United States. The application of such relationships shows that larger earthquakes in distant seismic sources will lead to Intensities IV-VI (MM) in the vicinity of the site. Therefore the maximum earthquake potential at the site is assumed to be Intensity VII (MM). (See FSAR Section 2.5.2.4)

<u>References</u>

Howell, B.F. and Schultz, T.R. 1975. Attenuation of Modified Mercalli Intensity with Distance from the Epicenter.

Bulletin of the Seismological Society of America, Vol. 65, No. 3, p 651-666.

Question No. Q230.2 (Section 2.5.2.7) NRC Letter: January 31, 1983

Per Regulatory Guide 1.70, the probability of exceeding the OBE during the operating life of the plant should be determined.

Response:

The Operating Basis Earthquake (OBE) for the Millstone 3 unit was computed as one half of Safe Shutdown Earthquake (SSE) (0.17g) and has been specified as 0.09g. The NRC also allows the OBE to be specified on the basis of probability studies. In such a case the, NRC requires a maximum 30 percent probability of exceeding OBE during the 40 year life of the plant. This corresponds to a return period of 110 years for the OBE. Algermissen et. al., (1982) have examined earthquake hazard in the U.S. An analysis of data provided by Algermissen et. al., yields an expected ground motion at Millstone of 0.06g for a return period of 110 years. The OBE at Millstone 3 is therefore higher than would be computed by probability studies in accordance with NRC criteria. The analysis also shows that the return period for the specified OBE of 0.09g is 250 years. This means that during the 40 years operating life of the plant, there is a 15 percent probability of exceeding 0.09g.

It should be noted that Algermissen et. al., (1982) assume much higher values of maximum earthquakes in each seismic source than was assumed in the FSAR. If maximum earthquakes assumed in the FSAR are used in the computations then the probability of exceeding 0.09g in a 40 year period would be significantly less than 15 percent.

Reference

1. Algermissen, S.T., Perkins D.M., Thenhaus P.C., Hanson S.L., and Bender B.L., "Probabilistic Estimates of Maximum Acceleration and Velocity in Rock in the Contiguous United States," U.S. Geological Survey Open-File Report 8-1033, 1982.

NRC Letter: January 31, 1983

Question No. Q250.1 (Section 5.2.4.8)

Provide a schedule for submittal of the final PSI/ISI program and relief requests.

Response:

The PSI (as-built) program and the first 10 year ISI plan and relief requests will be submitted to the NRC 6 months prior to commercial operation.

The PSI program plan description, along with the preliminary details for Class I components, will be forwarded to the NRC in June 1983.

NRC Letter: January 31, 1983

Question No. Q252.1 (Section 5.3.3)

As per Regulatory Guide 1.70, the introductory material should identify the reactor vessel designer and manufacturer and should describe their experience.

Response:

Information regarding the reactor vessel designer and manufacturer is included in FSAR Section 5.3.3.3.

The reactor vessel materials surveillance program is adequate to accommodate the annealing of the reactor vessel. Sufficient specimens are available to evaluate the effects of the annealing treatment.

Cyclic loads are introduced by normal power changes, reactor trip, startup, and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Vessel analyses result in a usage factor that is less than one.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates imposed by plant operating limits are 50°F per hour for normal operations and 100°F per hour under abnormal or emergency conditions. These rates are reflected in the vessel design specifications.

5.3.3.2 Materials of Construction

The materials in the fabrication of the reactor coolant are discussed in Section 5.2.3.

5.3.3.3 Fabrication Methods

Q252,1 The Millstone Unit 3 reactor vessel manufacturer is Combustion Engineering Incorporated. Combustion Engineering Incorporated is the largest reactor vessel fabricator in the United States and their experience is demonstrated by the fact that they have fabricated over 40 reactor vessels for Westinghouse designed NSSS's as well as additional vessels for other reactor vendors.

The fabrication methods used in the construction of the reactor vessel are discussed in Section 5.3.1.2.

5.3.3.4 Inspection Requirements

The nondestructive examinations performed on the reactor vessel are described in Section 5.3.1.3.

5.3.3.5 Shipment and Installation

The reactor vessel is shipped in a horizontal position on a shipping sled with a vessel lifting truss assembly. All vessel opening are sealed to prevent the entrance of moisture and an adequate quantity of desiccant bags are placed inside the vessel. These are usually placed in a wire mesh basket attached to the vessel cover. All carbon steel surfaces are painted with a heat resistant paint before shipment except for the vessel support surfaces and the top surface of the external seal ring.

NRC Letter: January 31, 1983

Question No. Q270.1 (Sections 3.11B.3 and 3.11N.3)

Provide a schedule for reporting the qualification program results.

Response:

BOP - The Environmental Qualification Document (EQD) will first be submitted 12 months prior to fuel load with periodic updates to be provided thereafter.

NSSS - The attached Westinghouse letter NS-EPR-2710 dated January 28, 1983 supplies the scheduled documentation submittal dates to the NRC for the Westinghouse Equipment Qualification Data Packages which are currently included as part of the Westinghouse Equipment Qualification program.

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

NS-EPR-2710

January 28, 1983

Darrell G. Eisenhut
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, MD 20014

Dear Mr. Eisenhut:

The attached Table 1 (Revision 8, Sheets 1-8) is being provided as an update for the scheduled documentation submittal dates to the NRC for the Equipment Qualification Data Packages (EQDP's) which are presently included as part of the Westinghouse Equipment Qualification Program which complies with IEEE Standards 323-74 and 344-75 and is defined in WCAP 8587. The "bar" located along the right hand column of the horizontal page indicates information which has changed since the last revision of this document. This schedule is updated periodically to reflect changes/additions in either the scheduled testing, documentation submittals or new equipment to be qualified as part of the Westinghouse program.

If you have any questions concerning this information, please contact Alex Ball (412-374-5792) or Clarence Draughon (412-374-5761).

Sincerely yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

AB/keg

Attchment(s)

TABLE 1(Sheet 1 of 8) Revision 8 1/28/83

Equipment	W-EQDP Reference	Manufacturer	System	Test Location		Scheduled Testing: Thermal Aging	Radiation	Seismic	HELB	Scheduled Documentation Submittal to the NRC
Safety Related Valve Electric Motor Operators	HE-1	Limitorque	CVCS SIS CCS	Limitorque Isomedix Acton	Start Complete	11-03-82 11-23-82	12-06-82 12-17-82	1-03-83 1-07-83	1-14-83 1-24-83	6-30-83
	HE-4	Limitorque	RHRS		Start Complete	---------------	Completed	---------------	---------------	7-24-81 NS-EPR-2478 5-19-82 NS-EPR-2600
Safety Related Solenoid Valves	HE-2/HE-5	ASCO	CVCS SIS RCS WPS SS SGBP RHR	ASCO Isomedix Acton	Start Complete	---------------	Completed	---------------	---------------	7-24-81 NS-EPR-2477 5-19-82 NS-EPR-2600
Safety Related Externally Mounted Limit Switches	HE-3/HE-6	NAMCO	CVCS SIS CSS RHR RCS WPS SS SGBP	NAMCO ACDC Georgia Tech	Start Complete	---------------	Completed	---------------	---------------	3-31-81 NS-TMA-2414 8-3-81 NS-EPR-2490
Reed Switches	HE-7	CROSBY	PAMS	Wyle Labs	Start Complete	---------------	Completed	---------------	---------------	3-30-83
Electrical Connectors for Solenoid Valves & Limit Switches	HE-8	CONAX LITTON	CVCS SIS CSS RHR RCS WPS SS SGBP	ACME Cleveland Development Co.	Start Complete	---------------	Completed	---------------	---------------	2-28-83
Garrett PORV	HE-9	GARRETT	RCS	W Forest Hills	Start Complete	---------------	Completed	---------------	---------------	2-28-83
Head Vent System	HE-10(1)	TARGET ROCK	HVS	East-West Labs	Start Complete	---------------	Completed	---------------	---------------	4-15-83
Pressure Transmitters Group A	ESE-1	Barton Veritrak	RPS PAM	Barton W-Forest Hills W-AESD ISOMEDIX	Start Complete	---------------	Completed	---------------	---------------	9-30-81(B) NS-EPR-2514 5-19-82(B) NS-EPR-2600 10-13-82(V) NS-EPR-2668

Equipment	W-EQDP Reference	Manufacturer	System	Test Location		Scheduled Testing: Thermal Aging	Radiation	Seismic	HELB	Scheduled Documentation Submittal to the NRC
Pressure Transmitters Group B	ESE-2	Barton Veritrak	RPS PAM	Barton W-Forest Hills W-AESD TSOMEDIX	Start Complete	----------------	Completed	----------	----------	6-16-80(B) NS-I%A-2258 3-31-81(V) NS-!MA-2418 8-3 6 ;V)(B) NS-EPR-2489
Differential Pressure Group A	ESE-3	Barton Veritrak	RPS PAM	W-Forest Hills W-AESD TSOMEDIX	Start Complete	----------------	Completed	----------	----------	5'-30-81(B) NS-EPR-2513 5-19-82(B) NS-EPR-2600 10-13-82(V) NS-EPR-2668
Differential Pressure Transmitter Group B	ESE-4	Barton Veritrak	PAM RPS	W-Forest Hills W-AESD TSOMEDIX	Start Complete	----------------	Completed	----------	----------	6-16-80(B) NS-TMA-2259 3-31-81(V) NS-TMA-2419 7-31-81(V) NS-EPR-2471 5-19-82(B)(V) NS-EPR-2600
Resistence Temperature Detector (RC-Bypass)	ESE-5	RdF	RPS	W-Forst Hills W-AESD TSOMEDIX SURGICUT Univ. of Buffalo	Start Complete	----------------	Completed	----------	----------	3-31-81 NS-TMA-2424 7-08-82 NS-EPR-2662
Resistence Temperature Detector (Well-mounted)	ESE-6	RdF	PAMS	W-Forest Hills W-AESD TSOMEDIX SURGICUT Univ. of Buffalo	Start Complete	----------------	Completed	----------	----------	3-31-81 NS-TMA-2425 7-13-81 NS-EPR-2470 5-19-82 NS-EPR-2600
Resistence Temperature Detectors (Fast response)	ESE-7	RdF	RPS	W-Forest Hills W-AESD TSOMEDIX SURGICUT Univ. of Buffalo	Start Complete	----------------	Completed	----------	----------	3-31-80 NS-TMA-2420 7-13-81 NS-EPR-2473 5-19-82 NS-EPR-2600
2 Section Excore Neutron Dectector (Power Range)	ESE-8	W-IGTD	RPS	W-IGTD W-AESD W-Forest Hills	Start Complete	----------------	Completed	----------	----------	2-28-83

Equipment	W-EQDP Reference	Manufacturer	System	Test Location		Scheduled Testing: Thermal Aging	Radiation	Seismic	HELB	Scheduled Documentation Submittal to the NRC
Excore Neutron Detectors (Source and Intermediate)	ESE-9	W-IGTD	RPS	W-Forest Hills W-AESD W-IGTD	Start Complete	----------------	See ESE-47	----------------	----------------	
Nuclear Instrumentation System (NIS)	ESE-10	W-ID	RPS	W-AESD W-ID	Start Complete	----------------	Completed	----------------	----------------	6-16-80 NS-TMA-2257 8-3-81 NS-EPR-2485 2-25-82 NS-EPR-2567
Source Range Preamplifier (Outside Containment)	ESE-11	W-ID	RPS	W-Forest Hills W-AESD W-ID	Start Complete	----------------	See ESE-47	----------------	----------------	
Main Control Board Switch Modules	ESE-12	W-ISD(OIM) Electroswitch	RPS ESF	W-AESD	Start Complete	----------------	Completed	----------------	----------------	2-28-83 (OIM)
					Start Complete	----------	Electroswitch Not Scheduled	----------	----------	
Process Protection System	ESE-13	W-ISD	RPS	W-AESD	Start Complete	----------------	Completed	----------------	----------------	1-28-81 NS-TMA-2380 8-3-81 NS-EPR-2482
Indicators	ESE-14	W-RID International	PAM	W-AESD	Start Complete	----------------	Completed	----------------	----------------	1-26-81(W) NS-TMA-2373 8-3-81(W) NS-EPR-2488
Recorders	ESE-15	W-CID	PAM	W-AESD	Start Complete	----------------	Completed	----------------	----------------	6-16-80 NS-TMA-2260 8-3-81 NS-EPR-2487
Solid State Protection System & Safeguard Test Cabinet (2 Train)	ESE-16	W-ID	RPS ESF	W-AESD W-ID	Start Complete	----------------	Completed	----------------	----------------	1-16-81 NS-TMA-2368 8-3-81 NS-EPR-2483 5-19-82 NS-EPR-2600
Solid State Protection System & Safeguard Test Cabinet (3 Trains)	ESE-17	W-ID	RPS ESF	W-AESD W-ID	Start Complete	----------------	Completed	----------------	----------------	7-8-82 NS-EPR-2622

Equipment	W-EQDP Reference	Manufacturer	System	Test Location		Scheduled Testing: Thermal Aging	Radiation	Seismic	HELB	Scheduled Documentation Submittal to the NRC
Instrument Bus Power Supply (Static Inverter) 7.5 KVA	ESE-18	W-PED	Electrical Power Supply	W-AESD	Start Complete	---------------	Completed	---------------		7-31-80 NS-TMA 2280 8-3-81 NS-EPR-2486 2-26-82 NS-EPR-2567
Instrument Bus Distribution Panel 7.5 KVA	ESE-19	W-PED	Electrical Power Supply	W-AESD	Start Complete	---------------	Completed	---------------		1-26-81 NS-TMA-2372 2-26-82 NS-EPR-2567
Reactor Trip Switch gear	ESE-20	W-LVSD	RPS	W-AESD	Start Complete	---------------	Completed	---------------		9-30-80 NS-TMA-2309 8-3-81 NS-EPR-2484 2-26-82 NS-EPR-2567
Pressure Sensor	ESE-21	Barton	RPS PAMS	Barton W-Forest Hills W-AESD ISOMEDIX	Start Complete	---------------	Completed	---------------		7-24-81 NS-EPR-2480 5-19-82 NS-EPR-2600
4 Section Excore Neutron Detector	ESE-22	W-IGTD	RPS	W-IGTD W-AESD W-Forest Hills	Start Complete	---------------	Completed	---------------		7-24-81 NS-EPR-2481 5-19-82 NS-EPR-2600
Loop Stop Valve Cabinet	ESE-23	W-ID	RPS	Analysis Only	Start Complete	-------------------	N/A	-------------------		7-31-83
RCP Speed Sensor	ESE-24	Electro Corp.	RPS	W-Forest Hills W-AESD ISOMEDIX	Start Complete	---------------	Not Scheduled	---------------		
Advanced Control Room	ESE-25(5)	W-CID	RPS ESF PAM	W-AESD	Start Complete	---------------	Not Scheduled	---------------		
Reactor Trip Switch gear	ESE-26(3)	W-LVSD	RPS	W-AESD	Start Complete	---------------	Not Scheduled	---------------		
Nitrogen-16 Detector	ESE-27	W-IGTD	RPS	W-IGTD W-AESD	Start Complete	---------------	Completed	---------------		2-28-83
Rod Position Detector	ESE-28(3)	Productos Electronicos, Inc. Puerto Rico	RPS	W-AESD W-R&D	Start Complete	---------------	Not Scheduled	---------------		

Equipment	W-EQDP Reference	Manufacturer	System	Test Location		Thermal Aging	Radiation	Seismic	HELB	Scheduled Documentation Submittal to the NRC
						Scheduled Testing				
Rod Position Data Cabinet	ESE-29(3)	W-ID	RPS	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Integrated Protection Cabinet	ESE-30(3)	W-ISD	RPS	W-AESD W-ISD	Start Complete	---------------	Not Scheduled	-------------		
Integrated Logic Cabinet	ESE-31(3)	W-ISD	RPS	W-AESD W-ISD	Start Complete	---------------	Not Scheduled	-------------		
Field Termination Cabinet	ESE-32(3)	W-ISD	RPS	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Instrument Bus Distribution Panel	ESE-33(3)	W-AESD	Electrical Power Supply	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Instrument Bus Distribution Panel	ESE-34(3)	W-AESD	Electrical Power Supply	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Instrument Power Supply	ESE-35(3)	W-AESD	Electrical Power Supply	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Source Range Preamplifier (In-Containment)	ESE-36(3)	W-ID	RPS	W-Forest Hills W-AESD	Start Complete	---------------	Not Scheduled	-------------		
PAMS Demultiplexer	ESE-37(3)	W-ISD	PAM	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Control Board Multiplexer	ESE-38(3)	W-ISD	RPS ESF	W-AESD	Start Complete	---------------	Not Scheduled	-------------		
Fiber Optic Cable	ESE-39(3)	W-ISD Siemens Amphenol	RPS	W-ISD	Start Complete	---------------	Not Scheduled	-------------		
Differential Pressure Indicating Switch Group B	ESE-40	Barton	RHR Mini-Flow	W-AESD	Start Complete	---------------	Completed	-------------		3-31-83
Microprocessor	ESE-41(4)	W-ID	PAMS	W-AESD	Start Complete	N/A	N/A	1-07-83 1-21-83	N/A	6-30-83
Resistance Temperature Detector (Strap-on)	ESE-42	MINCO	PAMS	W-Forest Hills W-AESD TSOMEDIX	Start Complete	-------------	Completed	----------------		2-15-83

Equipment	W-EQDP Reference	Manufacturer	System	Test Location		Scheduled Testing: Thermal Aging	Scheduled Testing: Radiation	Scheduled Testing: Seismic	Scheduled Testing: HELB	Scheduled Documentation Submittal to the NRC
Incore Thermocouples, Connectors and Adapters	ESE-43(4)	CKB Industries	PAMS	W-Forest Hills W-AESD	Start Complete	11-01-82 11-15-82	1-01-83 1-08-83	1-10-83 2-11-83	2-16-83 3-09-83	3-30-83
Incore Thermocouple Reference Junction Box	ESE-44(4)	W-IGTD	PAMS	W-IGTD W-Forest Hills W-AESD	Start Complete	1-15-82 2-28-82	1-15-82 2-28-82	10-25-82 11-04-82	11-08-82 11-24-82	3-30-83
Acoustic Leak Monitor Sensor	ESE-45(4)	W-ID	PAMS	W-AESD	Start Complete	--------------	Not Scheduled	--------------		
TMI Instrumentation: Remote Digital Display & Digital Recorder	ESE-46(4)	W-ID	PAMS	W-AESD	Start Complete	N/A	N/A	4-01-83 4-15-83	N/A	6-30-83
Boron Dilution Fix	ESE-47(2)	W-ID	RPS	W-AESD	Start Complete	N/A	N/A	3-08-83 3-22-83	N/A	6-30-83
High Volume Pressure Sensor	ESE-48([illegible])	Barton	PAMS	Barton W-Forest Hills W-AESD TSOMEDIX	Start Complete	--------------	Completed	--------------		2-15-83
Differential Pressure Indicating Switch Group A (Hydraulic Isolator)	ESE-49(8)	Barton	PAMS	Barton W-Forest Hills W-AESD	Start Complete	--------------	Completed	--------------		2-15-83
RVLIS (Reactor Vessel Level Instrumentation System)	ESE-50(6)	Barton W-ID RINCO	PAMS	Barton W-Forest Hills W-AESD TSOMEDIX	Start Complete	1-15-82 8-15-82	4-30-82 5-30-82	1-07-83 4-15-83	10-25-82 11-22-82	6-30-83
Core Cooling Monitor System	ESE-51(7)	W-ID	PAMS	W-AESD	Start Complete	1-15-82 6-30-82	1-15-82 7-30-82	1-10-83 4-15-83	11-08-82 3-09-83	6-30-83
R.G. 1.97 Detector System	ESE-52	W-IGTD	PAMS	W-IGTD W-AESD	Start Complete	7-18-83 8-15-83	8-18-83 9-01-83	9-07-83 10-07-83	10-07-83 11-07-83	3-30-84
Plant Safety Monitoring System (PSMS)	ESE-53	W-ITTC	PAMS	W-AESD	Start Complete	--------------	Completed	--------------		3-15-83

TABLE 1 (Sheet 7 of 8) Revision 8 1/28/83

Equipment	W-EQBP Reference	Manufacturer	System	Test Location		Thermal Aging (Scheduled Testing)	Radiation (Scheduled Testing)	Seismic (Scheduled Testing)	HELB (Scheduled Testing)	Scheduled Documentation Submittal to the NRC
Metal Impact Monitoring System (MIMS)	ESE-54	W-ITTC	MIMS	W-AESD	Start Complete	---------------	Completed	---------------		2-31-83
Auxiliary Safeguards Cabinet	ESE-55	W-ESD	ESF	W-AESD	Start Complete	---------------	Completed	---------------		7-31-83
Safety Related Pump Motors	AE-1(Medium)	W-MM&G		W-AESD ISOMEDIX W-R&D	Start Complete	---------------	Completed	---------------		7-15-81 NS-EPR-2479 5-19-82 NS-EPR-2600
	AE-2(Large)	W-LMD Buffalo		W-AESD ISOMEDIX	Start Complete	---------------	Completed	---------------		3-21-81 NS-TMA-2412 8-3-81 NS-EPR-2491 5-19-82 NS-EPR-2600
	AE-3(Canned)	Chempump	CVCS EBS SIS RHRS CCWS CSS	W-AESD Lockheed Permagrain	Start Complete	---------------	Completed	---------------		3-31-81 NS-TMA-2413 7-13-81 NS-EPR-2472 5-19-82 NS-EPR-2600
	AE-4(EBS)	WEIR		---	Start Complete	---------------	Not Scheduled	---------------		
Hydrogen Recombiner	SP-1	W-Sturtevant		W-AESD W-Canada Inc.	Start Complete	---------------	Completed	---------------		Accepted NRC LTR 6-22-78 WCAP-7709L Supplements 1-7
Component Aging Program - Phase I Results - Appendix A1										NS-TMA-2421 3-31-81 5-19-82 NS-EPR-2600
Materials Aging Program - Appendix A2										NS-TMA-2422 3-31-81 5-19-82 NS-EPR-2600
WCAP-8587 "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment" (Revision 5)										5-19-82 NS-EPR-2600

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References:

(1) This report includes information on the isolation valve, modulating valve and controller.
(2) This report includes information on the source and intermediate range neutron detectors, pre-amps and drawers.
(3) This equipment is applicable to 414 type plants.
(4) Equipment for Post TMI Instrumentation. Microprocessor used in Containment Monitor, Core Cooling Monitor and RVLIS Systems.
(5) Control Board qualification is on specific licensing applications. Westinghouse will not submit documentation to the NRC under the Generic '74 Program. This documentation will be submitted by the applicant.
(6) This report includes information on the microprocessor, strap-on RTDs, remote digital displays in control room, high volume pressure sensor, and the hydraulic isolator.
(7) This report includes information on the thermocouples, connectors and adaptors, reference junction box, microprocessor, remote digital display in control room and data links.
(8) Used as hydraulic isolator for the RVLIS system

Legend:

CVCS = Chemical Volume Control System
SIS = Safety Injection System
WPS = Waste Processing System
SGBP = Steam Generator Blowdown (Waste) Processing (System)
RHR = Residual Heat Removal (System)
PAM = Post Accident Monitoring
CCWS = Component Cooling Water System
MIMS = Metal Impact Monitoring System
RPS = Reactor Protection System
ESF = Engineered Safeguard Feature
RCS = Reactor Coolant System
SS = Sampling System
CSS = Containment Spray System
EBS = Emergency Boration System
HVS = Head Vent System

NRC Letter: January 31, 1983

Question No. Q280.1 (Section 9.5.1)

The fire protection program will be reviewed to the guidelines of BTP CMEB 9.5-1 (NUREG-0800), July 1981. Provide a comparison that shows 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:

Millstone Unit 3 conformance to BTP CMEB 9.5-1 is provided in the appendices to the separate report entitled "Fire Protection Evaluation Report". Documentation of deviations is also provided in FSAR Section 1.9, Standard Review Plan Documentation of Differences.

NRC Letter: January 31, 1983

Question No. Q281.1 (Section 6.1.1.2)

For all postulated design basis accidents involving release of water into the containment building, estimate the time-history of the pH of the aqueous phase.

Response:

A time-history of the pH of the water on the containment floor following a DBA is shown on the attached Figure 1. The time-history is representative of all LOCAs which actuate the containment depressurization system. In considering each LOCA case, the short-term mass and energy release differences have no effect on the ultimate sump pH, and an insignificant effect on the short-term pH.

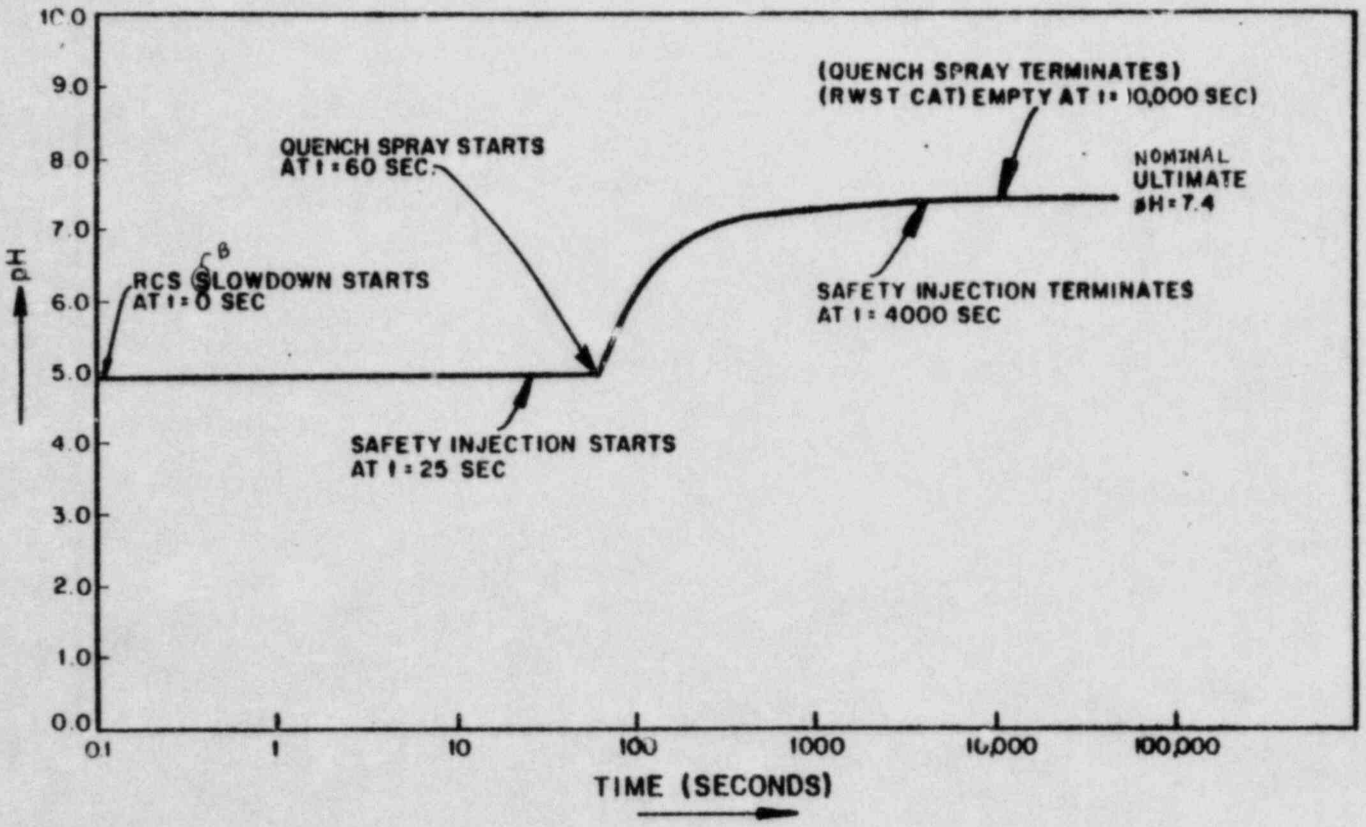

FIGURE Q281.1-1
CONTAINMENT SUMP pH TIME-HISTORY
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: January 31, 1983

Question No. Q281.2 (Sections 10.4.6 and 5.4.2.1)

Provide a discussion of how the secondary water chemistry program conforms with SRP 5.4.2.1, Branch Technical Position MTEB 5-3 in NUREG-0800.

Response:

The Millstone Unit 3 Chemistry Manual is the document containing the chemistry monitoring and control requirements. This document is scheduled to be completed by 12/31/83. A FSAR change to Section 5.4.2.1 will be submitted by 3/1/84. This submittal will incorporate the above information and bring the section into conformance with SRP 5.4.2.1 or exceptions will be noted.

NRC Letter: January 31, 1983

Question No. Q311.1 (Section 2.2)

Per Regulatory Guide 1.78, please provide a complete list of chemicals stored, used in manufacturing processes, or transported within a five mile radius of the plant. This should include materials transported on the nearby railroad and interstate highway. Please provide an analysis of the potential hazard associated with these chemicals.

Response:

Chemicals stored or currently used on the Millstone site are listed in Table 311.1-1. Chemicals transported on the railroad are identified in Table 311.1-2. Letters have been sent to the State of Connecticut Department of Transportation and the Federal Highway Administration requesting information on types and quantities of hazardous materials transported over state and federal roads within five miles of the plant. This information will be provided upon receipt from these agencies.

Data from local industries identifying specific chemicals used and/or stored for the manufacture of their products is being obtained. Chemicals stored or used at Millstone 3 will be included in a revision to Table 311.1-1.

The analysis of the potential hazard associated with these chemicals will be provided in September 1983.

TABLE Q311.1-1

ON-SITE STORAGE OF CHLORINE AND OTHER HAZARDOUS CHEMICALS -- TOTAL AMOUNT AND LOCATION

	CHEMICAL	QUANTITY	MILLSTONE LOCATION
1.	Sulfuric Acid	MP1: 7000 gals	Sulfuric acid storage tank just north of the gas turbine building.
		MP2: 8000 gals	Elevation 4'6" condensate polishing facility.
2.	Sodium Hydroxide 50% by Weight	MP1: 7500 gals	Caustic sol. storage tank just north of the gas turbine building.
		MP2: 7500 gals	Condensate polishing facility.
3.	Hydrazine 35% by Weight (non-flammable)	MP1: One 55 gal drum	Reactor building; elevation 41'6" near RBCCW pumps.
		MP2: Two 55 gal drums	Elevation 14'6" northwest corner of turbine building.
		Site: Ten 55 gal drums	Warehouse No. 3
4.	Penetone Products	Penetone 724; Six 55 gal drums	Warehouse No. 3
5.	Ammonia 30% by Weight	MP2: 1500 gals	Northwest corner; elevation 14'6" turbine building.
6.	Acetone	Four 55 gal drums	Warehouse No. 3
7.	Nitrogen (Liquid)	MP1: 10,000 gal	Just west of condensate storage tank.
		Seven 160 liter cylinders	Adjacent to radwaste and control buildings.
8.	Hydrogen	108,333 scf/ month	South end of site, adjacent to chlorine tank cars.

TABLE Q311.1-1 (Cont)

	CHEMICAL	QUANTITY	MILLSTONE LOCATION
9.	Chlorine	Two 55 ton tank cars	South end of site adjacent to gas turbine building.
10.	Helium	Four 20 ft^3 bottles	Gas bottle storage shed.
		Eight 213 ft^3 bottles	
11.	Methanol	15 gals	Chemistry lab.
12.	Phosphoric Acid 60% by Weight	5700 gals	MP1: Radwaste; used for solidfication. Scheduled to be replaced by Dow Process.
13.	Ureaformaldehyde	5700 gals	MP1: Radwaste
14.	Halon	MP1: Four 100# bottles	MP1: Computer room.
		MP2: One 22# bottle	MP2: Computer room.
		approximately 60#	Proposed for new MP2 computer room.
		MP2: Six 298# bottles	New records storage vault CPF building.
		1-196# Cylinder	Millstone Unit 1 Fire Pump House
15.	Carbon dioxide-CO_2	MP1: Six 100# bottles	Generator exciter.
		Six 75# bottles	Gas turbine building.
		One 750# tank	Turbine deck.
		MP2: Seven 75# bottles	Generator exciter.
		17-75# bottles	Old records storage area.
		One 750# tank	Turbine deck.

TABLE Q311.1-1 (Cont)

CHEMICAL	QUANTITY	MILLSTONE LOCATION
	Site: Three 75# bottles	Flammable liquid storage building.

TABLE Q311.1-2

LIST OF HAZARDOUS MATERIALS TRANSPORTED
BY RAIL PAST THE MILLSTONE SITE

Hazardous Material	No. of Carloads
Chlorine	20
Anhydrous Ammonia	8
Carbon Dioxide - Liquid	9
Propane	66
Ethyl Alcohol	22
Rosin	1
Ammonium Nitrate	9
Hydrochloric (Muriatic) Acid	5

NOTE:

(1) Based on ConRail link report from January 1978 through June 1979

NRC Letter: January 31, 1983

Question No. Q311.2 (Section 2.2)

Table 2.2-2 provides a list of hazardous materials [carried on Conrail/Amtrak track] potentially capable of producing significant missiles. Please provide the basis of this list, i.e., years from which the data was compiled.

Response:

The data was compiled from January, 1978 through June, 1979.

Please refer to reference entitled, "Hazardous Materials Node Report Produced for Stone & Webster by ConRail. Node Activity for New Haven, Connecticut, by Type from January 1978 through June 1979" in Section 2.2.4.

NRC Letter: January 31, 1983

Question No. Q311.3 (Section 2.2)

Throughout section 2.2 the demographic data is presented in kilometers. Please provide the requisite data in miles with rings at 1, 2, 3, 4, 5, 10, 20, (25 optional) 30, 40, and 50 miles. Please include all forms of transient and seasonal populations (see Regulatory Guide 1.70 Section 2.1.3.3) within 5 miles and significant concentrations of same beyond 5 miles.

Response:

Population per sector, based on English system distances, is listed in Tables Q311.3-1 through Q311.3-14. Distances do not conform exactly to those given in Regulatory Guide 1.70 but instead are provided in distances suitable for the Probabalistic Safety Study. Transient population within 5 miles are listed by distance and direction in Tables Q311.3-15 through Q311.3-17.

TABLE Q311.3-1

1980 POPULATION DISTRIBUTION
0-50 (MILES)

	Distance (Miles)										
Direction	0-10	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6,209	3,585	3,801	4,215	2,394	3,819	22,992	15,034	16,350	9,419	9,416	97,234
NNE	13,314	3,449	7,323	12,988	13,256	16,095	7,918	10,604	14,725	14,430	15,579	129,681
NE	33,198	3,521	4,375	2,492	1,747	4,426	3,753	9,235	16,357	49,825	108,963	237,892
ENE	24,075	8,129	3,167	4,347	6,244	8,012	6,631	11,430	21,117	44,766	74,863	212,781
E	7,806	613	1,898	2,910	4,632	7,414	1,652	4,908	6,921	1,135	1,592	41,481
ESE	875	125	0	0	0	0	0	615	0	0	0	1,615
SE	0	0	0	0	154	889	0	0	0	0	0	1,043
SSE	0	0	119	125	395	1,676	0	0	0	0	0	2,315
S	304	0	292	226	2,128	6,674	262	0	0	0	0	9,886
SSW	118	721	138	581	1,826	6,602	8,465	8,756	518	0	0	27,725
SW	161	0	472	3,149	1,681	5,897	8,206	13,479	22,557	56,346	82,581	194,529
WSW	5,472	335	0	0	0	0	0	0	310	10,562	45,270	61,949
W	4,992	6,324	5,059	3,739	8,566	10,729	14,997	32,277	120,757	142,931	109,444	459,815
WNW	3,510	2,782	3,485	4,311	2,175	6,066	18,007	39,549	94,929	65,705	138,140	378,659
NW	4,008	673	979	1,287	1,810	6,905	21,332	39,104	118,751	280,731	98,897	574,477
NNW	5,395	1,104	1,089	1,389	2,881	6,688	9,153	16,221	82,445	56,488	46,438	229,291
Total	109,437	31,361	32,197	41,759	49,889	91,892	123,368	201,212	515,737	732,338	731,183	2,660,373

TABLE Q311.3-2

1985 POPULATION DISTRIBUTION
(0-50 Miles)

	Distance (Miles)											
DIRECTION	0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6,384	3,709	3,912	4,284	2,455	3,957	23,609	15,179	16,554	10,076	9,761	99,880
NNE	13,626	3,588	7,467	13,148	13,437	16,421	8,508	11,390	15,482	15,051	15,949	134,067
NE	33,411	3,719	4,624	2,596	1,782	4,584	4,114	9,967	17,574	51,922	113,306	247,599
ENE	24,297	8,299	3,323	4,506	6,464	8,543	7,069	12,579	24,369	50,548	84,619	234,616
E	7,872	650	1,960	3,029	4,847	7,741	1,695	5,414	7,094	1,389	1,947	43,638
ESE	960	141	0	0	0	0	0	633	0	0	0	1,734
SE	0	0	0	0	173	998	0	0	0	0	0	1,171
SSE	0	0	134	141	444	1,881	0	0	0	0	0	2,600
S	341	0	328	254	2,388	7,492	294	0	0	0	0	11,097
SSW	133	809	155	652	2,051	7,409	9,502	9,828	582	0	0	31,121
SW	169	0	530	3,534	1,886	6,619	9,210	15,131	25,316	63,246	92,690	218,331
WSW	5,827	343	0	0	0	0	0	0	347	11,855	50,813	69,185
W	5,247	6,469	5,248	3,892	8,923	11,397	15,836	32,973	120,630	143,186	111,192	464,993
WNW	3,689	2,858	3,573	4,490	2,340	6,618	19,025	40,663	96,648	68,167	139,867	397,938
NW	4,206	722	1,034	1,350	1,911	7,399	22,266	41,515	122,277	280,976	101,434	585,090
MNW	5,593	1,191	1,188	1,500	3,103	7,290	10,122	17,299	84,245	58,860	48,278	238,669
TOTAL	111,755	32,498	33,476	43,376	52,204	98,349	131,250	212,571	531,118	755,276	769,856	2,771,729

TABLE Q311.3-3

1990 POPULATION DISTRIBUTION
(0-50 Miles)

	Distance (Miles)											
DIRECTION	0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6,546	3,845	4,038	4,382	2,508	4,038	24,188	15,349	16,732	10,618	10,086	102,330
NNE	13,994	3,756	7,704	13,492	13,794	16,796	9,080	12,173	16,285	15,654	16,533	139,261
NE	33,958	3,995	4,972	2,701	1,783	4,660	4,432	10,726	18,817	53,874	115,930	255,048
ENE	24,508	8,470	3,404	4,611	6,662	8,984	7,708	13,903	25,912	52,328	85,862	242,352
E	7,961	678	2,036	3,125	5,030	8,086	1,877	5,978	7,758	1,486	2,084	46,099
ESE	1,043	157	0	0	0	0	0	694	0	0	0	1,894
SE	0	0	0	0	193	1,111	0	0	0	0	0	1,304
SSE	0	0	149	157	494	2,093	0	0	0	0	0	2,893
S	380	0	364	283	2,659	8,339	327	0	0	0	0	12,352
SSW	148	901	172	726	2,283	3,245	10,576	10,938	647	0	0	34,636
SW	173	0	590	3,934	2,100	7,367	10,251	16,841	28,181	70,395	103,171	243,003
WSW	6,194	348	0	0	0	0	0	0	387	13,194	56,556	76,679
W	5,473	6,556	5,355	3,992	9,230	11,849	16,307	33,430	121,678	144,531	113,678	472,079
WNW	3,854	2,933	3,649	4,691	2,480	7,057	19,843	41,837	98,320	70,148	141,846	396,653
NW	4,354	767	1,079	1,401	1,995	7,804	23,217	43,848	126,477	285,359	104,083	600,384
NNW	5,741	1,260	1,266	1,583	3,301	7,820	10,959	18,379	86,327	61,444	50,254	248,334
TOTAL	114,327	33,666	34,778	45,078	54,512	104,249	138,765	224,096	547,521	779,031	800,083	2,876,106

TABLE Q311.3-4

2000 - POPULATION DISTRIBUTION
(0-50 Miles)

	Distance (miles)											
	0-10	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0-	20.0-25.0 -	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
DIRECTION												
N	6,723	4,184	4,312	4,476	2,556	4,177	25.449	16,407	18,056	11,685	10,729	108,754
NNE	14,105	4,075	8,098	14,028	14,355	17,458	9,619	13,057	17,607	16,59[illegible]	17,446	146,439
NE	33,961	4,355	5,424	2,857	1,807	4,890	4,812	11,529	20,117	56,342	119,918	266,012
ENE	24,780	8,7[illegible]	3,614	4,849	6,955	9,539	8,541	14,918	28,267	56,030	89,983	256,272
E	7965	728	2,153	3,301	5,282	8,534	2,072	6,297	8,710	1,556	2,181	48,779
ESE	1140	176	0	0	0	0	0	694	0	0	0	2,010
SE	0	0	0	0	217	1,251	0	0	0	0	0	1,460
SSE	0	0	168	176	557	2,358	0	0	0	0	0	3,259
S	427	0	411	320	2,995	9,394	369	0	0	0	0	13,916
SSW	167	1,015	194	819	2,571	9,292	11,912	12,324	729	0	0	39,023
SW	178	0	665	4,431	2,366	8,300	11,550	18,971	31,743	79,298	116,215	273,717
WSW	6,785	352	0	0	0	0	0	0	436	14,864	63,711	86,148
W	5,805	6,644	5,606	4,254	9,787	12,951	17,866	34,599	125,287	1[illegible],472	117,327	488,598
WNW	4,074	3,030	3,761	4,971	2,728	7,948	21,572	43,721	101,040	73,366	145,793	412,004
NW	4,555	849	1,175	1,516	2,168	8,606	25,062	48,039	132,619	289,048	107,362	620,999
NNW	5,896	1,441	1,479	1,759	3,516	8,422	12,290	20,245	90,375	66,280	52,843	264,546
Total	116,561	34,645	37,060	47,757	57,860	113,120	151,114	240,801	574,986	813,532	843,508	3,031,944

TABLE Q311.3-5

2010 POPULATION DISTRIBUTION
0-50 Miles

	Distance (mi)											
Direction	0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6902	4557	4665	4774	2577	4003	26468	17255	18778	12147	11256	113382
NNE	14831	4564	9030	15654	15978	18456	10363	14292	19329	17342	17925	157764
NE	36323	5239	6538	3043	1643	4635	5007	12045	20753	57555	122483	275264
ENE	25086	9143	3391	4770	7216	9792	9069	15515	29458	57961	92581	263982
E	8187	745	2370	3364	5488	8882	2201	6493	9270	1602	2247	50849
ESE	1276	205	0	0	0	0	0	694	0	0	0	2175
SE	0	0	0	0	253	1456	0	0	0	0	0	1709
SSE	0	0	196	205	648	2744	0	0	0	0	0	3793
S	497	0	477	372	3485	10932	429	0	0	0	0	16192
SSW	194	1181	226	951	2992	10813	13863	14341	848	0	0	45409
SW	159	0	773	5156	2753	9657	13441	22077	36939	92276	135240	318471
WSW	7537	341	0	0	0	0	0	0	507	17296	74138	99819
W	6041	6458	5395	4190	10097	12769	17085	34344	134595	157780	125673	514427
WNW	4279	3151	3822	5442	2848	8225	22179	46163	103799	74321	151134	425363
NW	4542	905	1218	1553	2230	8938	27032	52038	143630	318041	112252	672379
NNW	5343	1493	1551	1767	3700	8857	13049	22278	96017	72458	56864	283877
Total	121697	37982	39652	51241	61908	120159	160186	257535	613923	878779	901793	3244855

TABLE Q311.3-6

2020 POPULATION DISTRIBUTION
0-50 Miles

	Distance (mi)										
Direction	0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	7014	4983	5080	5168	2555	3607	27369	18277	19522	12194	11703	117472
NNE	15699	5159	10292	17946	18246	19640	10980	15459	21209	18000	18318	170948
NE	39792	6392	7991	3223	1359	4088	5102	12505	21223	58461	124853	284989
ENE	25344	9495	2900	4492	7465	9994	9680	16036	30448	59698	94562	270114
E	8478	737	2641	3362	5693	9258	2388	6665	9774	1645	2307	52948
ESE	1414	235	0	0	0	0	0	694	0	0	0	2343
SE	0	0	0	0	289	1667	0	0	0	0	0	1956
SSE	0	0	224	235	741	3142	0	0	0	0	0	4342
S	570	1	547	425	3989	12513	490	0	0	0	0	18535
SSW	222	1352	259	1089	3426	12379	15870	16418	971	0	0	51906
SW	123	0	885	5903	3153	11057	15386	25274	42288	105642	154826	364537
WSW	8333	317	0	0	0	0	0	0	580	19804	84877	113911
W	6162	6048	4891	3931	10210	11812	15000	33235	148397	171245	136597	547528
WNW	4426	3262	3828	5987	2870	8095	22071	48774	106318	73421	157403	436455
NW	4343	940	1228	1546	2223	8948	29062	55697	156933	361845	117347	730112
NNW	5593	1478	1551	1665	3783	8978	13273	24299	102680	79395	61252	303947
Total	127513	40399	42317	54972	66002	125178	166671	273333	660343	961350	964045	3482123

TABLE Q311.3-7

2030 POPULATION DISTRIBUTION
0-50 Miles

	Distance (mi)											
Direction	0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	7070	5457	5560	5673	2494	2977	28129	19404	20179	11791	12064	120798
NNE	16790	5867	11916	20976	21229	21035	11529	16632	23285	18545	18513	186317
NE	44593	7855	9835	3405	946	3215	5069	12652	21064	58371	125139	292144
ENE	25569	9857	2112	3993	7608	9742	9676	16009	30476	59578	94651	269271
E	8871	702	2972	3260	5758	9358	2450	6690	9831	1649	2312	53853
ESE	1553	267	0	0	0	0	0	694	0	0	0	2514
SE	0	0	0	0	329	1894	0	0	0	0	0	2223
SSE	0	0	254	267	842	3569	0	0	0	0	0	4932
S	647	1	621	483	4532	14216	557	0	0	0	0	21057
SSW	252	1536	294	1238	3890	14061	18029	18652	1102	0	0	59054
SW	69	0	1006	6706	3581	12560	17480	28713	48039	120011	175888	414053
WSW	9193	281	0	0	0	0	0	0	659	22495	96421	129049
W	6174	5405	4066	3456	10126	9995	11436	31179	166893	189069	150460	580259
WNW	4523	3373	3782	6626	2789	7522	21182	51621	108640	70611	164681	445350
NW	3952	954	1201	1487	2138	8612	31162	59042	172953	422210	122882	826593
NNW	5137	1385	1469	1446	3776	8314	12963	26333	110463	87186	66181	325153
Total	134393	42940	45088	59016	70038	127570	169662	287621	713584	1061516	1029192	3740620

TABLE Q311.3-8

1980 POPULATION DISTRIBUTION
0-10 Miles

Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
	Distance (mi)														
N	0	0	116	495	119	357	773	91	17	45	317	359	1697	1823	6209
NNE	0	0	31	325	475	806	614	241	288	1904	1850	1295	1862	3623	13314
NE	23	153	57	439	410	191	1036	2595	5649	6537	5717	4020	3728	2643	33198
ENE	6	68	160	210	111	108	514	4127	1182	140	7223	1364	4475	4387	24075
E	0	16	528	165	212	250	844	1871	209	76	751	0	621	2263	7806
ESE	0	0	73	89	68	11	0	0	0	0	0	0	219	415	875
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	147	157	304
SSW	0	0	0	0	0	0	0	0	0	0	0	0	6	112	118
SW	0	0	0	0	29	132	0	0	0	0	0	0	0	0	161
WSW	0	0	0	0	1302	179	204	112	0	1009	1103	1510	35	18	5472
W	0	0	0	257	1019	383	409	694	23	295	435	187	525	765	4992
WNW	0	0	0	516	723	504	524	23	40	32	157	52	518	421	3510
NW	0	0	37	580	364	147	645	297	89	319	573	52	513	392	4008
NNW	0	122	458	198	438	272	246	232	400	155	455	75	1076	1268	5395
Total	29	359	1460	3274	5270	3340	5809	10283	7897	10512	18581	8914	15422	18287	109437

TABLE Q311.3-9

1985 POPULATION DISTRIBUTION
0-10 Miles

	Distance (mi)														
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	119	507	122	365	791	93	17	46	324	368	1746	1886	6384
NNE	0	0	32	333	487	824	628	246	293	1916	1860	1325	1907	3775	13626
NE	23	156	58	449	420	196	1060	2588	5621	6505	5728	4064	3790	2753	33411
ENE	6	70	164	215	114	110	513	4107	1176	142	7336	1389	4500	4455	24297
E	0	17	541	169	216	256	854	1862	208	77	762	0	628	2282	7872
ESE	0	0	75	92	69	12	0	0	0	0	0	0	246	466	960
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	165	176	341
SSW	0	0	0	0	0	0	0	0	0	0	0	0	7	126	133
SW	0	0	0.	0	30	139	0	0	0	0	0	0	0	0	169
WSW	0	0	0	0	1367	188	215	118	0	1082	1182	1619	38	18	5827
W	0	0	0	264	1062	402	429	729	24	316	466	201	563	791	5247
WNW	0	0	0	531	748	529	550	24	42	33	166	56	557	453	3689
NW	0	0	38	597	375	155	677	312	94	342	602	54	542	418	4206
NNW	0	125	469	203	450	279	253	237	416	165	478	79	1119	1320	5593
Total	29	368	1496	3360	5460	3455	5970	10316	7891	10624	18904	9155	15808	18919	111755

TABLE Q311.3-10

1990 POPULATION DISTRIBUTION
0-10 Miles

	Distance (miles)													
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	121	515	124	371	804	94	18	47	330	374	1792	1956	6546
NNE	0	0	32	338	494	837	637	249	298	1948	1892	1346	1943	3980	13994
NE	24	159	59	456	427	199	1075	2630	5712	6610	5799	4093	3835	2880	33958
ENE	6	71	166	218	116	112	521	4173	1195	142	7368	1390	4558	4472	24508
E	0	17	549	172	220	260	869	1892	211	77	765	0	632	2297	7961
ESE	0	0	76	93	70	12	0	0	0	0	0	0	274	518	1043
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	184	196	380
SSW	0	0	0	0	0	0	0	0	0	0	0	0	8	140	148
SW	0	0	0	0	31	142	0	0	0	0	0	0	0	0	173
WSW	0	0	0	0	1395	191	219	120	0	1173	1282	1755	41	18	6194
W	0	0	0	275	1092	411	438	744	25	341	505	217	611	814	5473
WNW	0	0	0	553	775	540	562	25	43	34	170	61	603	488	3854
NW	0	0	40	621	390	158	691	318	96	371	614	56	560	439	4354
NNW	0	127	477	209	468	286	257	241	424	175	488	84	1146	1358	5741
Total	30	374	1520	3450	5602	3519	6073	10486	8022	10919	19213	9376	16187	19556	114327

TABLE Q311.3-11

2000 POPULATION DISTRIBUTION
(0-10 miles)

	Distance (Miles)													
DIRECTION	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	TOTAL
N	0	0	118	505	121	364	788	92	17	46	323	367	1,857	2,125	6,723
NNE	0	0	32	332	485	821	625	245	292	1,909	1,854	1,320	1,928	4,262	14,105
NE	23	156	58	448	419	195	1,055	2,576	5,593	6,473	5,766	4,183	3,948	3,068	33,961
ENE	6	69	163	214	113	110	510	4,086	1,170	146	7,536	1,422	4,661	4,574	24,780
E	0	17	539	169	216	255	852	1,853	207	79	783	0	646	2,349	7,965
ESE	0	0	75	91	69	12	0	0	0	0	0	0	309	584	1,140
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	207	220	427
SSW	0	0	0	0	0	0	0	0	0	0	0	0	9	158	167
SW	0	0	0	0	32	146	0	0	0	0	0	0	0	0	178
WSW	0	0	0	0	1,436	197	226	124	0	1,320	1,443	1,975	46	18	6,785
W	0	0	0	283	1,124	423	451	766	25	384	568	245	687	849	5,805
WNW	0	0	0	569	798	556	578	25	44	35	178	68	677	546	4,074
NW	0	0	41	640	402	163	712	327	99	417	632	57	589	476	4,555
NNW	0	124	468	208	482	287	253	237	427	193	502	91	1,187	1437	5,896
TOTAL	29	366	1,494	3,459	5,697	3,529	6,050	10,331	7,874	11,002	19,585	9,728	16,751	20,666	116,561

TABLE Q311.3-12

2010 POPULATION DISTRIBUTION
(0-10 Miles)

	Distance (Miles)														
DIRECTION	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	TOTAL
N	0	0	115	491	118	353	765	90	17	45	314	356	1,922	2,316	6,902
NNE	0	0	31	322	471	796	605	240	292	2,025	1,970	1,281	1,902	4,896	14,831
NE	22	151	56	435	407	189	1,021	2,858	6,282	7,270	6,136	4,163	3,954	3,379	36,323
ENE	6	67	158	208	110	107	567	4,589	1,314	139	7,200	1,308	4,954	4,359	25,086
E	0	16	523	163	210	247	876	2,081	232	76	748	0	649	2,366	8,187
ESE	0	0	72	88	67	11	0	0	0	0	0	0	359	679	1,276
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	241	256	497
SSW	0	0	0	0	0	0	0	0	0	0	0	0	11	183	194
SW	0	0	0	0	28	131	0	0	0	0	0	0	0	0	159
WSW	0	0	0	0	1,285	176	202	111	0	1,586	1,733	2,372	55	17	7,537
W	0	0	0	319	1,099	378	404	685	23	456	683	294	825	875	6,041
WNW	0	0	0	641	842	497	517	23	40	31	166	82	809	631	4,279
NW	0	0	46	721	446	149	637	293	88	501	566	51	559	485	4,542
NNW	0	121	454	216	543	294	244	230	395	224	449	103	1,141	1,429	5,843
TOTAL	28	355	1,455	3,604	5,626	3,328	5,838	11,200	8,683	12,353	19,965	10,010	17,381	21,871	121,697

TABLE Q311.3-13

2020 POPULATION DISTRIBUTION
(0-10 Miles)

Direction	Distance (miles) 0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	108	461	111	332	719	84	16	42	295	335	1,976	2,535	7,014
NNE	0	0	29	303	443	745	566	229	286	2,204	2,148	1,203	1,830	5,713	15,699
NE	21	142	53	408	362	177	955	3,307	7,382	8,543	6,702	4,078	3,896	3,746	39,792
ENE	5	63	148	195	104	100	657	5,393	1,545	127	6,568	1,108	5,370	3,961	25,344
E	0	15	490	153	196	232	910	2,445	273	69	682	0	646	2,367	8,478
ESE	0	0	68	83	63	11	0	0	0	0	0	0	411	778	1,414
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	276	294	570
SSW	0	0	0	0	0	0	0	0	0	0	0	0	12	210	222
SW	0	0	0	0	22	101	0	0	0	0	0	0	0	0	123
WSW	0	0	0	0	997	137	157	86	0	1,916	2,094	2,866	67	13	8,333
W	0	0	0	367	1,024	293	313	531	18	547	825	355	997	892	6,162
WNW	0	0	0	738	887	386	401	18	31	24	143	99	971	728	4,426
NW	0	0	53	830	504	122	494	227	68	606	439	40	488	472	4,343
NNW	0	113	427	224	625	300	226	216	334	262	349	118	1,034	1,365	5,593
Total	26	333	1,376	3,762	5,358	2,936	5,398	12,536	9,953	14,340	20,245	10,202	17,974	23,074	127,513

TABLE Q311.3-14

2030 POPULATION DISTRIBUTION
(0-10 Miles)

	Distance (miles)														
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	98	418	100	301	652	76	14	38	268	304	2,020	2,781	7,070
NNE	0	0	26	275	401	674	509	214	280	2,458	2,403	1,090	1,722	6,738	16,790
NE	19	129	48	370	347	160	860	3,952	8,961	10,369	7,500	3,926	3,772	4,180	44,593
ENE	5	57	134	176	94	90	785	6,547	1,875	108	5,609	812	5,920	3,357	25,569
E	0	14	443	139	178	211	957	2,968	332	59	583	0	635	2,352	8,871
ESE	0	0	61	75	57	10	0	0	0	0	0	0	467	883	1,553
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	313	334	647
SSW	0	0	0	0	0	0	0	0	0	0	0	0	14	238	252
SW	0	0	0	0	12	57	0	0	0	0	0	0	0	0	69
WSW	0	0	0	0	563	77	88	49	0	2,320	2,535	3,471	81	9	9,193
W	0	0	0	431	898	166	177	300	10	656	999	430	1,208	899	6,174
WNW	0	0	0	866	937	218	227	10	17	14	104	120	1,168	842	4,523
NW	0	0	62	974	579	80	279	128	39	733	248	22	371	437	3,952
NNW	0	103	387	233	731	307	201	196	242	307	197	136	859	1,238	5,137
Total	24	303	1,259	3,957	4,897	2,351	4,735	14,440	11,770	17,062	20,446	10,311	18,550	24,288	134,393

TABLE Q311.3-15

TRANSIENT POPULATION - PARK VISITATION

Park Name	Sector Location (miles)	Annual Attendance	Average Daily Attendance
Ocean Beach	3.4 E	500,000	1,370[1]
Fort Griswold	5.5 ENE	47,220	129[2]
Harkness Memorial	2.8 E	146,745	868[2]
Rocky Neck	3.8 W	533,312	1,630[3]
Submarine Memorial	5.6 NE	68,000	186[1]
Waterford Beach	2.8 ESE	53,000	663[4]

NOTES:

[1]Year-round use - average daily attendance based on 365 days/year.

[2]Seasonal attendance between April 15 to September 30.

[3]Seasonal camping with other year-round use - 53,275 campers from April 15 to September 30. 480,037 other visitors over 365 days.

[4]Based on attendance from mid-June through Labor Day.

Sources: State of Connecticut, Department of Environmental Protection, Parks and Recreation Unit, State Park Attendance, 1978.

American Automobile Association, Campbook, Northeastern Region, 1981 Edition.

Telecon: Ellis, C.S. (SWEC) 1981p to Bugbee, R. (Parks and Recreation Dept.), Waterford, CT., December 21, 1981.

Ellis, C.S. (SWEC) 1981j to Submarine Memorial Assoc., Groton, CT., Nov. 20, 1981.

Ellis, C.S. (SWEC) 1981h to Butler, Mrs., Ocean Beach Park, New London, CT., Nov. 19, 1981.

TABLE Q311.3-16

1977 TRANSIENT POPULATION - EMPLOYMENT*

	Distance (miles)					
DIRECTION	0-1	1-2	2-3	3-4	4-5	Total
N	-	-	-	-	-	-
NNE	-	-	-	-	-	-
NE	-	-	315	-	655	970
ENE	-	-	-	-	2,767	2,767
E	-	-	-	-	-	-
ESE	-	-	-	-	-	-
SE	-	-	-	-	-	-
SSE	-	-	-	-	-	-
S	-	-	-	-	-	-
SSW	-	-	-	-	-	-
SW	-	-	-	-	-	-
WSW	-	-	-	-	-	-
W	-	-	-	-	-	-
WNW	-	-	-	-	-	-
NW	-	200	-	115	-	315
NNW	-	-	-	-	-	-
Total	-	200	315	115	3,422	4,052

NOTE:

*Firms with 50 or more emplpoyees

Source: 1977 Facts Book. Southeastern Connecticut Chamber of Commerce, New London, CT.

TABLE Q311.3-17

TRANSIENT POPULATION
1981-82 SCHOOL ENROLLMENTS

	0-1	1-2	2-3	3-4	4-5	Total
N	0	384	0	271	0	655
NNE	0	0	0	0	369	369
NE	0	0	1,073	755	2,790	4,613
ENE	0	293	0	800	0	1,093
E	0	0	0	0	0	0
ESE	0	0	0	0	0	0
SE	0	0	0	0	0	0
SSE	0	0	0	0	0	0
S	0	0	0	0	0	0
SSW	0	0	0	0	0	0
SW	0	0	0	0	0	0
WSW	0	0	0	0	0	0
W	0	0	0	0	0	0
WNW	0	0	381	0	0	381
NW	0	0	785	372	0	1,157
NNW	0	0	0	0	1,716	1,716
Total	0	677	2,239	2,198	4,875	9,989

Sources: Connecticut Education Directory, 1981-82, Connecticut State Board of Education, Hartford, CT.

NRC Letter: January 31, 1983

Question No. Q311.4 (Section 3.5.1.6)

This section does not include or reference sufficient information for the staff to evaluate aircraft hazards at Millstone 3. Provide or reference appropriate information that is identified in Section 3.5.1.6 of Regulatory Guide 1.70, Revision 3, and that addresses the acceptance criteria in SRP 3.5.1.6 in NUREG-0800.

Response:

The results of the aircraft hazards evaluation presented in FSAR Section 3.5.1.6 were obtained from the analysis performed for Millstone Unit 1, Docket 50-245, and submitted to the NRC in response to SEP Topic III-4.D. Based upon new data received from the airports in the vicinity of Millstone Station, a new analysis will be performed and the results submitted in August 1983.

Question No. Q410.1 (Section 3.5.1.4) NRC Letter: January 31, 1983

The tornado missile protection analysis should take into account the effect of missiles on ventilation openings in the various facility buildings housing essential shutdown equipment. Reference or provide a discussion addressing this subject.

Response:

Refer to Section 3.5.1.4, Amendment 2.

3.5.1.3.5 Turbine Overspeed Protection

The turbine control system is an electrohydraulic control (EHC) system that includes both digital and analog circuitry, electronic servo hardware, and hydraulic valve actuators.

The EHC provides a normal overspeed protection system and an emergency overspeed protection system to limit turbine overspeed. These two systems are essentially separate and independent. The normal overspeed protection system is part of the turbine load and speed control system and is designed to limit turbine overspeed without a turbine trip under all load conditions. The emergency overspeed protection system is part of the emergency trip system and is designed to trip the turbine if the turbine speed exceeds 110 percent of rated speed (Section 10.2).

3.5.1.3.6 Turbine Valve Testing

The main turbine generator control, main stop, intercept, and reheat stop valves are routinely tested (Section 10.2.3.6).

Control and main stop valves are tested one at a time, and as each test is completed, the valve is returned to its original position before the next valve is tested. Intercept and reheat stop valves are interlocked so that a pair of these valves in one crossover pipe is tested together. For this test, one pair of pipe is tested and the valves returned to the open position before the next pair is tested.

3.5.1.4 Missiles Generated by Natural Phenomena

Q220.1 The only credible missiles generated by natural phenomena are those generated by a design basis tornado. Those Category I structures designed to withstand the effects of tornado missiles and the systems and components thus protected are identified in Tables 3.5-1 and 3.5-2.

Q220.1 A minimum of 2 feet thickness of reinforced concrete having a minimum strength of 3,000 psi (28 day compressive strength) was used for walls, roofs, and floors designated as missile protection. The minimum reinforcing steel each way in each face of any square foot of wall or slab providing missile protection is 1.85 square inches.

Postulated missiles generated by the design basis tornado (Section 3.3.2) are listed in Table 3.5-13, which includes all parameters necessary to determine missile penetration. These missiles are considered capable of striking in any orientation.

Q410.1 Ventilation openings in the various facility buildings housing essential shutdown equipment are protected by reinforced concrete labyrinths.

There are no other design basis missiles resulting from flood or any other natural phenomena described in Section 2.2.3.

NRC Letter: January 31, 1983

Question No. Q410.2 (Section 3.6.1.3.3)

Section 3.6.1.3.3 and associated tables indicate that the pipe break/crack analysis is in progress for some of the systems and that results of the analysis will be provided in an amendment to the FSAR. Either provide this information or a schedule for its submittal.

Response:

The results of the pipe break/crack analysis will be provided 15 months prior to fuel load.

NRC Letter: January 31, 1983

Question No. Q410.3 (Section 5.2.5.4)

As per Regulatory Guide 1.70, the limiting leakage conditions that will be included in the technical specifications should be provided.

Response:

The design of the plant is based on the capability of detecting 1 gpm unidentified leakage. Section 5.2.5.2.4 demonstrates that this level of leakage can be detected. The limiting leakage conditions for Millstone 3 will be stated in FSAR Chapter 16, Technical Specifications.

NRC Letter: January 31, 1983

Question No. Q410.4 (Section 9.2.6)

Per Regulatory Guide 1.70, provide a discussion of the provisions for mitigating environmental effects of system leakage or storage tank failure.

Response:

The condensate storage tanks have been designed to prevent ingress of contaiminated water. The condensate storage and surge tanks normally receive clean makeup water from the condensate liquid waste system regenerant demineralizer. This makeup water is supplied only when radiation is not present in the LWC system. Valve 3LWC-AOV49 (Figure 9.2-6), which supplies water to the condensate surge and storage tanks, can only be opened when radiation is not present in the LWC system. (Section 11.2.4 states that this valve closes on radiation detection).

Water is also received from the water treating system (for tank fill) which is demineralized water (See Section 9.2.3.1).

Furthermore, backflow into the condensate storage tank from the condensate surge tank is prevented by a locked closed valve.

The limiting tank failure for the plant is analyzed in Section 2.4.13.3.

NRC Letter: January 31, 1983

Question No. Q410.5 (Section 9.3.3)

Per Regulatory Guide 1.70, provide a discussion of the system design that includes prevention of transfer of contaminated fluids to non-contaminated drainage systems.

Response:

The Reactor Plant Aerated Drains System described in Section 9.3.3.2.4 collects potentially contaminated drainage. The system also includes three uncontaminated underdrain sumps. There is no connection between the uncontaminated underdrain sumps and the contaminated section of the aerated drains systems.

Inadvertent transfer of potentially contaminated fluids to non-contaminated drainage systems is prevented by the following design features:

1. All gaseous drains described in Section 9.3.3.2.3 are processed through the radioactive gaseous waste and boron recovery systems

2. All aerated drains described in Section 9.3.3.2.4, except for the turbine building floor sump, are processed through the liquid waste system (LWS). The radiation monitor in the LWS discharge line and analysis of the low level waste drain tanks further prevents discharge of radioactive liquid to uncontaminated areas (Sections 11.2 and 11.5).

3. The turbine building floor sump is monitored for radioactivity and is redirected from the yard drainage system to the liquid waste system upon receipt of a high radiation level alarm (Sections 11.2 and 11.5).

4. Except for the Refueling Water Storage Tank (RWST), all potentially contaminated tanks in the yard area have their overflow routed to radioactive liquid waste sumps. The RWST is designed to accommodate maximum injection without overflow for at least two hours after receipt of a high water level alarm.

5. Pumps handling radioactive liquids are fitted with mechanical seals and outboard restriction bushings to minimize leakage. Leakage is directed to a radioactive sump through a drain connection.

6. Discharge from the condensate regenerant demineralizer to the environment is diverted to the regenerant evaporator feed tanks upon a high alarm (Section 11.5.1).

7. Effluent from the auxiliary condensate flash tank is diverted from the auxiliary condensate feed tank to the contaminated

auxiliary building sump following a high radiation alarm (Section 11.5.1).

8. The waste neutralization sump monitor measures the radioactivity in the equipment drains from the condensate polishing system and ensures that no radioactive leakage reaches the environment.

9. Liquid waste monitors, floor drains monitor, and the regenerative evaporator monitor automatically terminate discharge to the environment from their respective streams if the radioactivity exceeds a predetermined level.

NRC Letter: January 31, 1983

Question No. Q410.6 (Section 10.4.9.4)

Provide a discussion of the monthly AFWS pump tests being performed on a staggered test basis to reduce the likelihood of leaving more than one pump in a test mode following the tests and the associated technical specification for these tests.

Response:

Refer to Section 10.4.9.4, Amendment 2 for the response to this question. This requirement for the monthly AFWS pump tests being performed on a staggered test basis will be part of the Millstone 3 Technical Specifications (See Chapter 16).

The total system unavailability was determined from hardware failures, common cause failures, independent human errors, and test and maintenance errors. Figure 10.4-8 shows the simplified unavailability model for the Millstone 3 auxiliary feedwater system. Hardware unavailability was determined from the fault tree logic. The calculated system unavailability was 4.6×10^{-6}. The unavailability for independent human errors, and test and maintenance errors were based on the Zion Probability Safety Study, Volume 3 (Auxiliary Feedwater System, Section 1.5.2.3.9). It was assumed that the unavailability for these failures would be similar to the Zion Study since the Millstone 3 and Zion auxiliary feedwater systems are similar in design. The common cause unavailability (1×10^{-6}) shown on Figure 10.4-8 was an assumed value. There were no significant common cause failures found in the Zion Study.

10.4.9.4 Inspection and Testing Requirements

All piping in the auxiliary feedwater system will be hydrostatically tested during construction, and all active system components, such as pumps, valves, and controls, will be functionally tested periodically.

The steam generator auxiliary feedwater pumps, their drives, and the pump discharge valves are tested once a month on a staggered test basis. Each of the three pumps will be tested individually. The two motor driven pumps will be tested approximately 15 days apart to minimize the likelihood of leaving both in the test mode. An independent check of the system valve lineup will be performed after each test prior to restoring the system to operational status. An incorrect valve lineup will be alarmed on the ESF control panel. Steam is admitted to the turbine drive, and the motor drives are energized during these tests. Flow is established by recirculating auxiliary feedwater to the DWST. Following the completion of the test, the auxiliary feedwater pumps are shut off, and the motor-operated shutoff valves leading to the main feedwater lines are opened. Pump discharge pressure is indicated on the main control board for each auxiliary feedwater pump. Q410.6

All system components are tested and inspected in accordance with the applicable codes.

Periodic sampling of the water in the DWST is provided in the tank heating cycle. The samples are tested for oxygen content and pH.

Inservice inspection of Safety Class 2 and 3 portions of the auxiliary feedwater system will be performed as required by the ASME Code, Section XI (Section 6.6).

10.4.9.5 Instrumentation Requirements

The DWST is provided with redundant level indication on the main control board and on the auxiliary shutdown panel. High, low, and low-low (DWST) level alarms are provided on the main control board.

NRC Letter: January 31, 1983

Question No. Q430.1 (Section 8.2.2)

From the information presented in Section 8.2.2, it is not evident that the most critical transmission line has been determined. If analysis shows that loss of any single transmission line (as defined in SRP paragraph 8.2.2) will have a negligible effect on Millstone, Unit 3, Section 8.2.2 should so state; if not, the results of such a loss should be described.

Response:

Refer to Section 8.2.2, Amendment 2.

controlled by varying the reactive power generation on the Millstone units. The Millstone 1 and 2 operators control the unit excitation as specified by CONVEX Operation Instruction No. 6913. When Millstone 3 is placed in service, it will be included in the voltage regulation scheme. The unit operators are required to balance the reactive power output of the units.

The CONVEX system operator supervises the system reactive power dispatch. He directs the loading of all of the reactive power sources in CONVEX to balance the reactive supply. He keeps the Millstone reactive power generation in balance with the Eastern Area requirements so that the effect on the system of voltage variations is minimized when a unit is lost.

One objective of the reactive power dispatch is to prevent the voltage at Millstone from going below the minimum required by Millstone 1 reserve station service transformer. The maximum allowable voltage at Millstone is 362 kV based on equipment ratings.

If abnormal system conditions result in voltages approaching minimum levels, the "Guidelines for Dispatch of Reactive Power on the Northeast Utilities System" directs the CONVEX operator to take specific corrective actions to restore voltage. Many of these actions will also be taken when the Millstone reactive power output reaches 400 MVAR (with two units in service).

Actual experience and system simulations show that the CONVEX operator is able to control the system voltages within the desired limits.

The Millstone plant is connected to the transmission system by four 345-kV circuits (described in Section 8.2.1). For a short distance, these lines are on double circuit steel poles. Stability studies show that the plant remains in synchronism with the rest of the system, even if two 345-kV circuits are lost simultaneously as a result of a 3-phase fault.

Q430.1 The loss of any single 345 kV circuit will have a negligible effect on the Millstone units. Hence, no single circuit is critical to the operation of the units, the switchyard, the transmission system and the supply of offsite power. Furthermore, the loss of a second 345 kV circuit has been analyzed (i.e., after one has already been switched out of service) and results show that the loss will not result in instability of the Millstone units and will not cause facilities remaining in service to be loaded beyond acceptable limits.

The loss of a Millstone unit will not affect system stability.

The Millstone units are connected to the large interconnected transmission system in the eastern half of the United States. The interconnected system frequency is a 60± 0.03 Hz. Loss of large amounts of generation result in frequencies of as low as 59.94 Hz which recover to 60 Hz within a few minutes.

NRC Letter: January 31, 1983

Question No. Q430.2 (Section 8.3.1.2)

No reference is made to any equipment qualification tests; either state (and justify) that no such tests are necessary or identify the equipment requiring qualification and provide a schedule for such qualification.

Response:

Refer to Section 8.3.1.2.7 and the response to NRC question 270.1 for a schedule for submittal of the Environmental Qualification Document which will contain a list of equipment to be qualified by system and by equipment mark number.

NRC Letter: January 31, 1983

Question No. Q440.1 (Section 5.1.1)

Section 5.1.1 implies that typical values for principal parameters of the RCS are on Figure 5.1-2 that indicates these values at each numbered point are in process flow diagram tables. Provide or reference these tables.

Response:

These typical values are presented as notes to Figure 5.1-2, in Amendment 1.

NRC Letter: January 31, 1983

Question No. Q440.2 (Section 5.2.2.2)

Show that the results presented in WCAP-7769 are applicable to Millstone 3.

Response:

As discussed in Section 5.2.2.2, WCAP-7769 provides details of the methods used for relief device sizing based on a typical plant worst condition. The results, therefore, are applicable only to the typical plant presented in the WCAP. What is applicable, however, is the methodology utilized for safety valve sizing (Section 2 of WCAP-7769). By using this methodology, the maximum reactor coolant surge rate into the pressurizer is determined; actual valve capacity can then be chosen. For Millstone 3, the safety valve capacity exceeds the maximum surge rate calculated for the turbine trip event with no subsequent reactor trip.

Details of safety valve sizing and resultant overpressure analyses are provided in the Millstone 3 Overpressure Protection Report per Article NB-7300 of the ASME Code, Section III. The turbine trip analyses are also discussed within the FSAR in Section 15.2.3.

NRC Letter: January 31, 1983

Question No. Q440.3 (Section 5.4.7.2.1)

Section 5.4.7.2.1 refers to Figure 5.4-6. The notes for this figure appear to be incomplete in that the data for the process flow conditions are not presented. Provide the process flow conditions.

Response:

Revised notes to Figure 5.4-6 are provided in Amendment 2.

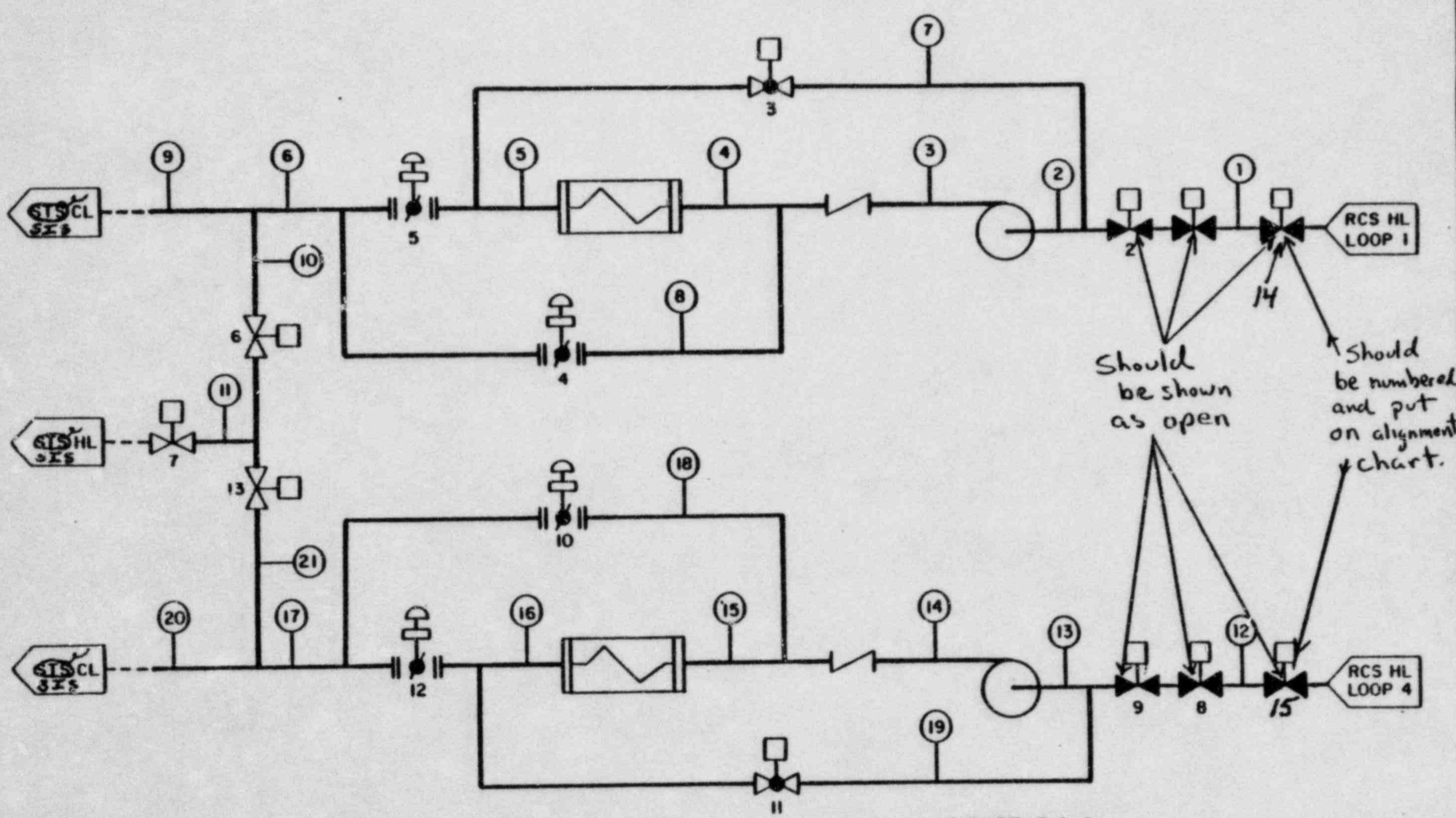

FIGURE 5.4-6
RESIDUAL HEAT REMOVAL SYSTEM PROCESS FLOW DIAGRAM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NOTES TO FIGURE 5.4-6

MODE A INITIATION OF RESIDUAL HEAT REMOVAL SYSTEM OPERATION

This mode presents the process flow conditions for the initiation of RHS operation. This begins the second phase of plant cooldown, when the reactor coolant temperature and pressure have been reduced to 350°F and 400 psig by use of the steam generators, transferring heat to the secondary side. 1

Both residual heat removal subsystems take suction from their respective RCS hot legs, discharging through heat exchangers with the return flow routed to the four RCS cold legs via cross-connected discharge headers. Q440.3 During the initial phases of RHS operation, reactor coolant flow through the heat exchangers is manually limited to control the rate of heat removal. The total flow is automatically requlated by flow control valves in the heat exchanger bypass lines to maintain a constant total return flow. The heat removal rate is limited to both control the RCS cooldown rate to 50°F/hr, based on equipment stress considerations, and to limit component cooling water temperature to a maximum of 120°F.

During this initial phase of RHS operation, one reactor coolant pump is maintained in operation. This results in a slight RHS return flow imbalance between the four RCS cold legs due to their different Q440.3 operating pressures. In the data presented, reactor coolant pump Number 2 is assumed operating.

MODE B END CONDITIONS OF NORMAL COOLDOWN

This mode presents the process flow conditions for the completion of RHS operation, typically 20 hours after RHS initiation. 1

The flow distribution of this mode, with all reactor coolant flowing through the heat exchangers with the bypass isolated, characterizes the majority of RHS operations.

Reactor coolant pump operation has also been terminated at this time, with all RCS cold legs in equilibrium.

NOTES TO FIGURE 5.4-6 (Cont)

MODE A - INITIATION OF RESIDUAL HEAT REMOVAL SYSTEM OPERATION

	Location	Fluid	Pressure (psig)	Temperature (°F)	Flow (gpm)	Flow (lb/hr)
	1	RC	400	350	4145(1)	1.85×10^6
	2	RC	384	350	4145(1)	1.85×10^6
Q440.3	3	RC	521	350	4145(1)	1.85×10^6
	4	RC	521	350	1232(1)	5.50×10^5
	5	RC	520	149	1232(1)	5.50×10^5
Q440.3	6	RC	447	291	4000	1.85×10^6
	7	RC	384	350	Note 1	
	8	RC	521	350	2913(1)	1.30×10^6
Q440.3	9	RC	447	291	4000	1.85×10^6
	10	RC	447	291	0(2)	
	11	RC	-	-	0	
	12	RC	400	350	4145(1)	1.85×10^6
	13	RC	384	350	4145(1)	1.85×10^6
	14	RC	521	350	4145(1)	1.85×10^6
	15	RC	521	350	1232(1)	5.50×10^5
Q440.3	16	RC	520	149	1232(1)	5.50×10^5
	17	RC	447	291	4000(2)	1.85×10^6
	18	RC	521	350	2913(1)	1.30×10^6
	19	RC	384	350	Note 1	
	20	RC	447	291	4000	1.85×10^6
Q440.3	21	RC	447	291	0(2)	

(1) At reference conditions 350°F and 400 psig
(2) At reference conditions 295°F and 500 psig

NOTES:

Q440.3 1. Miniflow continues until flow at locations 7 and 19 is greater than 1000 gpm. The miniflow is then isolated.

2. The RCS cold leg distribution is a result of operating reactor coolant pump number 2 during this phase of RHR operation.

NOTES TO FIGURE 5.4-6 (Cont)

MODE B - END CONDITIONS OF NORMAL COOLDOWN

Location	Fluid	Pressure (psig)	Temperature (°F)	Flow (gpm)	Flow (lb/hr)	
1	RC	0/Note 1	120	4000	1.98×10^6	
2	RC	11	120	4000	1.98×10^6	
3	RC	163	120	4000	1.98×10^6	
4	RC	163	120	4000	1.98×10^6	
5	RC	162	103	4000	1.98×10^6	
6	RC	47	103	4000	1.98×10^6	Q440.3
7	RC	11	-	0	0	
8	RC	163	-	0	0	
9	RC	47	103	4000	1.98×10^6	
10	RC	47	-	0		Q440.3
11	RC	-	-	0	0	
12	RC	0/Note 1	120	4000	1.98×10^6	
13	RC	11	120	4000	1.98×10^6	
14	RC	163	120	4000	1.98×10^6	
15	RC	163	120	4000	1.98×10^6	
16	RC	162	103	4000	1.98×10^6	
17	RC	47	103	4000	1.98×10^6	
18	RC	163	-	0	0	
19	RC	11	-	0	0	
20	RC	47	103	4000	1.98×10^6	
21	RC	47	-	0		Q440.3

NOTES:

1. RCS is assumed depressurized with the water level drained to the centerline of reactor coolant piping.

2. All reference conditions, 104°F and 0 psig. Q440.3

NOTES TO FIGURE 5.4-6 (Cont)

RHRS VALVE ALIGNMENT CHART

	Valve Number	Operational Mode A	Operational Mode B
	1	O	O
	2	O	O
Q440.3	3	C	C
	4	P	C
	5	O	O
	6	C	C
Q440.3	7	C	C
	8	O	O
	9	O	O
Q440.3	10	P	C
	11	C	C
	12	O	O
Q440.3	13	C	C
	14	O	O
	15	O	O

NOTES:

O = Open
C = Closed
P = Partially open

NRC Letter: January 31, 1983

Question No. Q440.4 (Section 5.4.7.2.2)

Specify the available net positive suction head for the RHR pumps.

Response:

Refer to Section 6.3.2.2.3, Section 5.4.7.2.2, and Table 5.4-8, Amendment 2.

The (RHS) pumps are started automatically on receipt of an SIS signal. The pumps deliver water to the RCS from the RWST during the injection phase. Each pump is a single stage vertical position centrifugal pump.

A minimum flow bypass line is provided for the pumps to recirculate and return the pump discharge fluid to the pump suction downstream of the RHS heat exchangers should these pumps be started with their normal flow paths blocked. Once flow greater than 1,000 gpm is established to the RCS, the bypass line is automatically closed. This line prevents deadheading of the pumps and permits pump testing during normal operation.

The limiting NPSH available when pumping, occurs for pump runout flow.

Q440.4 The available NPSH values for this case are:

RHS Pump A: 36.0 ft
RHS Pump B: 40.5 ft

The required NPSH for normal and runout flow conditions are given in Table 5.4-8.

The RHS pumps are discussed further in Section 5.4.7. A pump performance curve is given on Figure 6.3-3.

The pumps have a self-contained mechanical seal which is normally cooled by the component cooling water system. However, after a LOCA, cooling water will not be supplied or required, because the pumps will be pumping water having a maximum temperature of 50°F. The RHS pumps are not utilized in the recirculation phase.

Centrifugal Charging Pumps

In the event of an accident, the charging pumps are started automatically on receipt of an SIS and are automatically aligned to take suction from the RWST during injection. During recirculation, suction is provided from the containment recirculation pump discharge.

The charging pumps deliver flow to the RCS at the prevailing RCS pressure. Each centrifugal charging pump is a multistage centrifugal diffuser design (barrel-type casing) with vertical suction and discharge nozzles. The pump lubricating oil coolers are cooled by the charging pumps seal cooling subsystem (Section 9.2.2.4).

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the pump suction after cooling via the seal water heat exchanger during normal plant operation. The minimum flow bypass line contains two valves in series which close on receipt of the SIS. This signal also closes the valves to isolate the normal charging line and volume control tank and opens the charging pump/refueling water storage tank suction valves to align the high

Q440.4

The RHS pumps are protected from overheating and loss of suction flow by mini-flow bypass lines that assure flow to the pump suction. Should the pump suction be isolated or the RCS pressure be above the shutoff head of the pump. A valve located in each mini-flow line is regulated by a signal from the flow transmitters located in each pump discharge header. The control valves open when the RHS pump discharge flow is less than 500 gpm and close when the flow exceeds 1,000 gpm.

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also actuated by the pressure sensor.

The two pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The RHS pumps also function as the low head safety injection pumps in the ECCS. (See Section 6.3 for further information and for the RHS pump performance curves.)

Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water existing 24 hours after reactor shutdown when the temperature difference between the two systems is small.

The installation of two heat exchangers in separate and independent residual heat removal trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

The residual heat exchangers also function as part of the ECCS (Section 6.3).

Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor-operated valves have backseats to facilitate repacking and to limit steam leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

TABLE 5.4-8

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump

Number	2	Q440.4
Design pressure (psig)	600	
Design temperature (°F)	400	
Design flow (gpm)	4,000	
Design head (ft)	350	
NPSH required at 3,800 gpm (ft)	18	
NPSH required at runout flow 5,500 (ft)	28	
Power (hp)	400	

Residual Heat Exchanger

Number	2	Q440.4
Design heat removal capacity (Btu/hr)	35.27×10^6	
Estimated UA (Btu/hr °F)	3.5×10^6	

	Tube Side	Shell Side
Design pressure (psig)	600	150
Design temperature (°F)	400	200
Design flow (lb/hr)	1.98×10^6	3.3×10^6
Inlet temperature (°F)	120	92.2
Outlet temperature (°F)	102.2	102.9
Material	Austenitic	Carbon Steel Stainless Steel
Fluid	Reactor Coolant Water	Component Cooling

NRC Letter: January 31, 1983

Question No. Q440.5 (Section 6.3.2.5)

Identify the specific equipment arrangement for the plant design and provide an evaluation to ensure that valve motor operators located within containment will not become submerged following a LOCA. Include all equipment in the ECCS or any other system that may be needed to limit boric acid precipitation in the reactor vessel during long-term-cooling or that may be required for containment isolation.

Response:

A design to assure that valve motor operators do not become submerged following a LOCA is part of the Millstone 3 environmental qualification program (Section 3.11). The maximum containment flood level (El (-) 10'-8") is identified in Appendix A to FSAR Appendix 3B. The operators for all valves which must operate in the flooded condition are located above the maximum flood level.

Methods used to limit boric acid precipitation in the reactor vessel during long term cooling are described in Section 6.3.2.5. Identification of equipment used in the ECCS is provided in Figure 6.0-1. Arrangement of the ECCS equipment is shown in building layout figures in Section 3.8.

NRC Letter: January 31, 1983

Question No. Q440.6 (Sections 6.3.3.3 and 15.6.5.1)

Section 6.3.3.3 of the FSAR defines a small-break LOCA as a break up to 0.5 ft^2 in area and a large-break LOCA as larger than 0.5 ft^2. Section 15.6.5.1 defines a large break as greater than 1.0 ft^2 and a small break as less than 1 ft^2. Resolve the discrepancy.

Response:

A small break LOCA is defined as a rupture of the reactor coolant pressure boundary with a total cross sectional area less than 1.0 ft^2. The area associated with a large break is equal to or greater than 1.0 ft^2. Refer to Sections 6.3.3.2 and 6.3.3.3, Amendment 2. The Chapter 15 references to break sizes are correct.

The analyses deal with breaks of up to 1.0 ft^2 in area, where the safety injection pumps play an important role in the initial core recovery because of the slower depressurization of the RCS. |Q440.6

Results and Conclusions

The analysis of these breaks has shown that the high head portion of the ECCS, together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperature below required limits of 10CFR50.46. Hence, adequate protection is afforded by the ECCS in the event of a small break LOCA.

6.3.3.3 Major Reactor Coolant System Pipe Ruptures (LOCA)

Discussion

A major LOCA is defined as a rupture 1.0 ft^2 or larger of the RCS piping including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system. The boundary considered for LOCAs as related to connecting piping, is defined in Section 3.6. |Q440.6

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The safety injection system is actuated when the appropriate pressurizer low pressure setpoint is reached. Reactor trip and safety injection system actuation may be provided by a high containment pressure signal depending on the actual break size. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection provide additional negative reactivity insertion to supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

When the pressure falls below approximately 600 psig, the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injection bypasses the core and goes out through the break until the explusion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. This conservatism is consistent with the acceptable features of ECCS Evaluation Models as defined by Appendix K, 10CFR50.

The pressure transient in the reactor containment during a LOCA affects ECCS performance in the following ways: The time at which end of blowdown occurs is determined by zero break flow which is a result of achieving pressure equlibrium between the RCS and the containment. In this way, the amount of accumulator water bypass is also affected by the containment pressure, since the amount of accumulator water discharged during blowdown is dependent on the length of the blowdown phase and RCS pressure at end of blowdown.

NRC Letter: January 31, 1983

Question No. Q440.7 (Section 15.0)

For each of the accidents analyzed in Chapter 15, Accident Analyses, show that the most limiting single active component failure or single operator error as defined by the corresponding Chapter 15 SRP in NUREG-0800 has been assumed in the analysis of each accident.

Response:

The response to this question will be provided in June 1983.

NRC Letter: January 31, 1983

Question No. Q440.8 (Section 15.4.6)

Provide the analysis as required per Regulatory Guide 1.70, Revision 3, and SRP 15.4.6 of NUREG-0800.

Response:

An alternate plant specific analysis will be performed in place of the analysis required by Regulatory Guide 1.70, Revision 3 and SRP 15.4.6 of NUREG-0800. This analysis will support the position stated in the FSAR by evaluating the probabilities and consequences associated with the occurrence of a boron dilution event. Our analysis will be forwarded to you for your review in August 1983.

NRC Letter: January 31, 1983

Question No. Q451.1 (Section 2.3.2.1)

Per Regulatory Guide 1.70, provide extreme values of frequency and duration for fog.

Response:

The extreme values of fog frequency and duration are presented in Table 2.3-24. This table presents the duration (persistence) in hours of poor visibility conditions (visibility less than 1 mile). The extreme value for duration of such a poor visibility condition based on a January 1974 through December 1981 meteorological data base is 20 hours.

Table 2.3-24 also gives the frequency of visibility less than 1 mile. The extreme value for frequency of poor visibility is 13 percent in May 1979.

NRC Letter: January 31, 1983

Question No. Q451.2 (Section 2.3.4.2)

Per Regulatory Guide 1.70, provide hourly cumulative frequency distributions of relative concentrations (χ/Q), using onsite data at appropriate distances from the effluent release point(s), such as the minimum site boundary distance (exclusion area). The (χ/Q) values from each of these distributions that are exceeded 5% and 50% (median value) of the time should be reported. For the outer boundary of the low population zone, provide cumulative frequency of χ/Q estimates for:

(1) the 8-hour time period from 0 to 8 hours;

(2) the 16-hour period from 8 to 24 hours;

(3) the 3-day period from 1 to 4 days; and

(4) the 26-day period from 4 to 30 days.

Report the worst condition and the 5% and 50% probability level conditions.

Response:

Section 2.3.4 of Regulatory Guide 1.70, Revision 3, issued in November 1978, specifies the requirement of short-term diffusion estimates for the FSAR. This section references Regulatory Guide 1.4 for guidance on appropriate diffusion models for estimating χ/Q values resulting from a loss of coolant accident for PWR's. Regulatory Guide 1.145, Revision 1, which was issued in November 1982, supersedes Regulatory Guide 1.4. Consequently, the specifications presented in Section 2.3.4 of Regulatory Guide 1.70, Revision 3, do not reflect the guidance given in Regulatory Guide 1.145. The short-term diffusion analysis for FSAR Tables 2.3-34 through 2.3-37, and the following discussion are based on the guidance presented in Regulatory Guide 1.145.

The maximum sector 8 percent equal risk χ/Q value is another name for the maximum sector χ/Q as defined in Regulatory Guide 1.145. The χ/Q value at the exclusion area boundary or low population zone should be the maximum sector χ/Q or the 5 percent overall site χ/Q, whichever is higher. Hourly 8 percent equal risk χ/Q's which were estimated at the low population zone for each sector are presented in FSAR Tables 2.3-35 and 2.3-37. The maximum sector 8 percent equal risk χ/Q for the 0-2 hr period is 6.26×10^{-5} sec/m^3, which is larger than the 5 percent overall site χ/Q of 4.77×10^{-5} sec/m^3. Hourly 8 percent equal risk χ/Q's which were estimated at the exclusion area boundary for each sector are presented in FSAR Tables 2.3-34 and 2.3-36. The maximum sector 8 percent equal risk χ/Q for the 0-2 hr period (containment building release) is 5.42×10^{4} sec/m^3, which is larger than the 5 percent overall site χ/Q at 4.48×10^{-4} sec/m^3. The maximum sector 8 percent equal risk χ/Q for the 0-2 hr period (containment ventilation vent release) is

4.30×10^{-4} sec/m^3, which is larger than the 5 percent overall site χ/Q at 3.50×10^{-4} sec/m^3.

As stated in Regulatory Guide 1.145, "sector χ/Q values for the outer low population zone boundary should be determined for various time periods throughout the course of the postulated accident." Regulatory Guide 1.145 further states, "for a given sector, the average $\bar{\chi}/Q$ values for the various time periods should be approximated by a logarithmic interpolation between the 2-hr sector χ/Q and the annual average (8,760 hr) χ/Q for the same sector." The exclusion area boundary and low population zone 50 percent equal risk sector χ/Q values, which were determined for various time periods throughout the course of the postulated accident are presented in attached Tables 451.2-1 through 451.2-4.

TABLE Q451.2-1

MEDIAN (50-PERCENT EQUAL RISK) GROUND LEVEL X/Q VALUES ($x10^{-5}$ sec/m^3) AT THE EXCLUSION AREA BOUNDARY FOR THE 0 TO 720 HOUR PERIOD FOLLOWING AN ACCIDENT

(Containment Building)

Downwind Sectors	Distance (meters)	0-2 hrs	0-8 hrs	8-24 hrs	1-4 days	4-30 days
N	782	2.75	2.37	2.21	1.88	1.50
NNE	826	4.65	3.70	3.30	2.58	1.80
NE	548	12.30	9.87	8.85	6.99	4.98
ENE	524	13.10	10.60	9.56	7.59	5.46
E	524*	8.07	6.89	6.36	5.35	4.17
ESE	524*	8.60	7.39	6.85	5.81	4.58
SE	524*	8.60	7.34	6.79	5.72	4.47
SSE	524*	8.02	6.77	6.22	5.18	3.98
S	524*	8.23	7.16	6.68	5.74	4.62
SSW	524*	8.70	7.74	7.30	6.43	5.36
SW	524*	4.48	4.19	4.06	3.77	3.40
WSW	524*	1.65	1.65	1.65	1.65	1.65
W	524*	2.60	2.32	2.19	1.93	1.62
WNW	524*	4.32	3.74	3.48	2.97	2.38
NW	524	3.40	3.18	3.07	2.85	2.57
NNW	532	1.97	1.97	1.97	1.97	1.97

NOTE:

*Overwater Sector

TABLE Q451.2-2

MEDIAN (50-PERCENT EQUAL RISK) GROUND-LEVEL χ/Q VALUES ($\times 10^{-6}$ sec/m^3) AT THE LOW POPULATION ZONE FOR THE 0- TO 30-DAY PERIOD FOLLOWING AN ACCIDENT

(Containment Building)

Downwind Sector	Distance (meters)	0-2 hrs	0-8 hrs	8-24 hrs	1-4 days	4-30 days
N	3862	2.71	1.92	1.61	1.11	0.644
NNE	3862	6.50	4.06	3.21	1.93	0.926
NE	3862	13.20	7.72	5.90	3.30	1.43
ENE	3862	11.10	6.68	5.19	3.01	1.37
E	3862	5.12	3.41	2.78	1.79	0.52
ESE	3862	5.14	3.48	2.86	1.87	1.02
SE	3862*	4.89	3.31	2.72	1.78	0.970
SSE	3862*	4.42	2.97	2.44	1.58	0.853
S	3862*	4.78	3.29	2.74	1.83	1.02
SSW	3862*	5.01	3.54	2.98	2.04	1.19
SW	3862	1.93	1.51	1.33	1.01	0.686
WSW	3862	0.232	0.232	0.232	0.232	0.232
W	3862	1.04	0.781	0.676	0.494	0.315
WNW	3862	2.46	1.69	1.40	0.934	0.521
NW	3862	1.48	1.15	1.01	0.770	0.520
NNW	3862	0.289	0.289	0.289	0.289	0.289

NOTE:

*Overwater sector

TABLE Q451.2-3

MEDIAN (50-PERCENT EQUAL RISK) GROUND-LEVEL χ/Q VALUES
(x 10^{-5} sec/m^3) AT THE EXCLUSION AREA BOUNDARY FOR
THE 0 TO 720 HOUR PERIOD FOLLOWING AN ACCIDENT

(Containment Ventilation Vent)

Downwind Sector	Distance (meters)	0-2 hrs	0-8 hrs	8-24 hrs	1-4 days	4-30 days
N	722	3.41	2.91	2.69	2.26	1.77
NNE	1383	2.68	1.97	1.69	1.21	0.752
NE	706	10.70	8.18	7.15	5.33	3.50
ENE	600	10.80	8.67	7.77	6.12	4.34
E	600*	7.00	5.87	5.38	4.44	3.38
ESE	600*	7.00	5.98	5.52	4.65	3.64
SE	600*	6.92	5.88	5.42	4.55	3.53
SSE	600*	6.53	5.48	5.02	4.15	3.16
S	600*	6.77	5.84	5.43	4.63	3.68
SSW	600*	7.00	6.20	5.84	5.12	4.24
SW	600*	4.10	3.74	3.57	3.23	2.80
WSW	600*	1.29	1.29	1.29	1.29	1.29
W	600*	2.35	2.05	1.91	1.65	1.33
WNW	600*	4.15	3.47	3.18	2.62	1.98
NW	600	3.13	2.85	2.71	2.45	2.11
NNW	644	1.39	1.39	1.39	1.39	1.39

NOTE:

*Overwater sector

TABLE Q451.2-4

MEDIAN (50-PERCENT EQUAL RISK) ELEVATED χ/Q VALUES ($\times 10^{-6}$ sec/m^3) AT THE LOW POPULATION ZONE FOR THE 0 TO 720 HOUR PERIOD FOLLOWING AN ACCIDENT

(Containment Ventilation Vent)

Downwind Sector	Distance (meters)	0-2 hrs	0-8 hrs	8-24 hrs	1-4 days	4-30 days
N	3862	2.74	1.93	1.62	1.11	0.646
NNE	3862	6.59	4.11	3.24	1.94	0.936
NE	3862	13.60	7.92	6.04	3.35	1.44
ENE	3862	11.50	6.90	5.35	3.07	1.39
E	3862	5.14	3.42	2.79	1.80	0.953
ESE	3862	5.14	3.48	2.86	1.87	1.02
SE	3862*	4.91	3.32	2.73	1.79	0.971
SSE	3862*	4.42	2.97	2.44	1.58	0.853
S	3862*	4.78	3.29	2.74	1.83	1.02
SSW	3862*	5.11	3.60	3.02	2.06	1.20
SW	3862	1.95	1.52	1.34	1.02	0.688
WSW	3862	0.232	0.232	0.232	0.232	0.232
W	3862	1.05	0.781	0.677	0.494	0.315
WNW	3862	2.48	1.70	1.41	0.939	0.523
NW	3862	1.48	1.15	1.01	0.771	0.521
NNW	3862	0.289	0.289	0.289	0.289	0.289

NOTE:

*Overwater sector

NRC Letter: January 31, 1983

Question No. Q460.1 (Section 11.2.1)

Provide the peak flow rate of the Auxiliary Building Sump, or provide a schedule for the submittal of this information (Table 11.2-1).

Response:

Refer to Table 11.2-1, Amendment 2.

TABLE 11.2-1

LIQUID WASTE MANAGEMENT SYSTEM DAILY INPUT FLOWS

Source	Average Flow Rate (gal/day)	Peak Flow Rate (gal/day)	
High-Level Waste Drain Tanks			
Containment Building Sump	40	300	
Auxiliary Building Sump	200	1,500	Q460.1
Sample Fluids	35	200	
Laboratory Wastes	400	500	
Miscellaneous	660	800	
Low-Level Waste Drain Tanks			
Turbine Plant Leakage to Sumps	7,200		
Miscellaneous	40	650	
Regenerant Evaporator Feed Tanks	3,400		

NRC Letter: January 31, 1983

Question No. Q460.2 (Section 11.4.2)

Discuss the effectiveness of the physical and monitoring precautions taken to prevent (e.g., retention basis, curbing and level instrumentation) the potential for operator error or single equipment malfunction that may result in uncontrolled releases of radioactive material.

Response:

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'-6" 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 towards 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.

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. (49CFR 171-178). The high radiation interlock will automaticlaly shutdown 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 "refill" 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: January 31, 1983

Question No. Q460.3 (Section 11.4.2)

Discuss the provisions for sealing, decontaminating and moving the solid waste containers to storage or shipping areas, along with discussion on the potential for radioactive spills due to containers dropped from cranes, monorails, etc. Discuss the provisions for collecting and processing decontamination liquids and spillage. Provide layout drawings of the packaging, storage, and shipping areas of the solid radioactive waste system.

Response:

Before shipment or movement, containers will be sealed in accordance with 49CFR171-178 and 10CFR71, as applicable (see Section 11.4.2.4). Containers are prepared for shipment prior to leaving the process area in accordance with Millstone Station Procedures.

Refer to Section 11.4.2.7, Amendment 2, for a discussion of provisions for collecting and processing radioactive spills.

Refer to Figure 3.8-74 for the layout of the solid waste disposal building, as noted in 11.4.2, Amendment 2.

Regulatory Guide 8.8 guidelines in terms of "as low as is reasonably achievable" (ALARA) doses to plant personnel and the general public.

7. The temporary waste storage facilities in the waste storage building are shielded to provide protection of operating personnel in accordance with the radiation protection design considerations in Section 12.1.2.

8. Health physics personnel conducting periodic surveys using portable radiation detectors assure that radiation levels outside shielded areas are within design levels and establish access limitations within the shielded areas.

9. This system is not safety related and is classified as nonnuclear safety (NNS).

10. The portion of the solid waste building's foundation and adjacent walls up to a height sufficient to contain the liquid inventory in the building is designed to the seismic criteria of Section 5 and to the quality assurance criteria of Section 6 of Regulatory Guide 1.143.

11. Portions of the system that handle radioactive liquid waste will meet the applicable design bases of Section 11.2.1.

12. The system is designed to process, handle, and store the waste types, and quantities described in Section 11.4.2 below, which are generated as a result of normal operation.

13. The system is able to reduce the volume of input streams to minimize packaged quantities for transport and disposal. Flammable material such as waste solidification binder are subject to and handled in accordance with fire protection standards as discussed under SRP Section 16.0. Details of fire protection provisions are delineated in Section 9.5.1.

11.4.2 System Description

The radioactive solid waste system is shown on Figure 11.4-1.

Q460.3 The layout of the solid waste building, including packaging, storage, and shipping areas, is shown in Figure 3.8-74.

Radioactive solid waste system components are listed in Table 11.4-2, which summarizes design and operating conditions.

11.4.2.1 System Inputs

Materials handled as solid wastes include concentrated waste solutions from the waste evaporator and regenerant chemical evaporator, concentrated boric acid discarded from the boron evaporator in the boron recovery system (Section 9.3.5), spent resin

NRC Letter: January 31, 1983

Question No. Q460.4 (Section 11.5.2)

Provide the range of radioactivity concentrations to be monitored, and the bases for the range provided for each radiation monitor. Provide the setpoints for alarms and controls for each radiation monitor that uses them, and the bases for their selection.

Response:

Refer to Tables 11.5-1 and 11.5-2 for the ranges and sensitivity of process and effluent radiation monitors. A listing of control actions for these radiation monitors is provided in Section 11.5.1. The range of the detectors is selected to envelope the release concentrations expected for normal operations and for hypothetical design basis accidents and generally include state-of-the-art detection capability available for field installation.

For effluent monitors the high alarms and controls are based on activity concentrations specified in 10 CFR 20, Appendix B, Table II. Alert alarms are based on the ALARA design guidelines of 10CFR50, Appendix I.

For process monitors the high alarms and controls ar determined by system requirements for these monitors, and technical specifications where appropriate. Alert alarms are set at a fraction of the high alarm value.

NRC Letter: January 31, 1983

Question No. Q471.1 (Section 12.1.2)

Discuss the provisions of your radiation protection plan and how they are consistent with the provisions of NUREG-0761.

Response:

NUREG-0761 was issued in March 1981 in draft form for comment. Numerous comments were provided to the NRC by the industry. Until such comments are incorporated and the NUREG is issued in final form, the Applicant does not consider it appropriate to commit to the requirements of NUREG-0761. However, the Applicant does believe that current Millstone radiation protection procedures and practices comply with the intent of all of the draft NUREG-0761 recommendations. The NRC's special Health Physics Appraisal Program follows very closely the scope of NUREG-0761 in performance of their appraisals. Such an appraisal was performed in August 1980 and the Millstone Health Physics Program was found to be adequate for operation. Weaknesses identified during the appraisal have been subsequently corrected. A continuation of this same program, with appropriate updates and improvements will be implemented at Millstone Unit 3. The basic detail of this program are discussed in Section 12.5 of the FSAR.

NRC Letter: January 31, 1983

Question No. Q471.2 (Section 12.3.1)

Provide, in addition to Figure 12.3-1 through 12.3-4, any differentiation in zone boundaries between normal operation and refueling outage conditions, identify shield wall thicknesses, personnel and equipment decontamination areas, contamination control areas, traffic patterns, location of health physics facilities, location of airborne radioactivity and area radiation monitor, location of sampling ports, instrumentation and control panels for radwaste equipment and components, location on onsite laboratory for analysis of chemical and radioactive samples, and the location of the count room.

Response:

Radiation zone boundaries for refueling outage conditions are included as Figure 12.3-6 through 12.3-9, Amendment 2.

Refer to Section 12.3.1, Amendment 2, for the bases used to determine shield wall thicknesses.

Personnel and equipment decontamination areas are identified in Section 12.3.2.7, paragraph 2.

Traffic patterns are identified in Figures 12.2-1 through 12.2-3 and described in Section 12.3.2.1, paragraph 4.

Refer to Section 12.5.2 and Figure 12.5-1 for arrangement of health physics facilities, onsite laboratory for analysis of chemical and radioactive samples, and count room. The count room is located in the Service Building, El 24'-6" (Figure 3.8-72).

Refer to Table 11.5-1 for locations of airborne radioactivity monitors. Refer to Table 12.3-2 for locations of area radiation monitors. Specific locations of radiation monitors are shown in Figures 12.2-1 through 12.2-3.

Refer to Sections 11.5.2.2.1 through 11.5.2.2.9 and Figure 11.5-1 (Sheets 1 through 3) for airborne contamination sampling port locations. Refer to Tables 9.3-1 and 9.3-2 for liquid and gaseous waste system sampling points.

Refer to Figure 3.8-74 for locations of instrument and control panels for radwaste equipment and components.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Shielding

Radiation shielding is designed to ensure that radiation exposure to the general public and to personnel in-plant is kept to levels as low as is reasonably achievable (ALARA), consistent with the requirements set forth in 10CFR20 for normal operation and 10CFR100 for accident conditions and with the overall objectives set forth in USNRC Regulatory Guide 8.8. The design of this radiation shielding is based upon radiation zone criteria which were established in support of the expected access requirements and durations of occupancy during normal operations, during refueling outages, and during accident situations. Descriptions of the zone criteria are presented in Table 12.3-1 and the detailed radiation zone criteria for normal and shutdown operations are illustrated on Figures 12.3-1 through 12.3-4 and 12.3-6 through 12.3-9.

Radiation shielding is provided on the basis of maximum concentrations of radioactive materials within each shield region (e.g., 1 percent failed fuel) rather than the annual average values. For batch processes, as an example, the point of highest radionuclide concentration in the batch process will be assumed (e.g., just prior to draining of a tank). The shielding designs are, therefore, intentionally conservative in that the dose rates reflect maximum, rather than average, sources to be shielded. These maximum dose rates are based on anticipated occupancy requirements and are set such that the maximum exposure of plant personnel is within the limits set by 10CFR20. The average exposures are expected to be a small fraction of the limiting values because it is not expected that the plant will run at 1 percent failed fuel with all tanks full to capacity, all demineralizer beds at saturation, etc.

In computing the dose rates on which the confirmation of shielding thicknesses is based, a number of explicit and implicit conservative measures are included. Dose points are generally calculated along vertical shield surfaces opposite the most intense source in the vicinity. These calculations are based on the inherent assumption that plant personnel spend the required time in each zone in contact with the shield at this point. This is a demonstrably conservative approach, since the dose rate actually decreases dramatically as the dose points are moved along the surface of the shield due to the slant penetration involved. The additional reduction of intensity with distance is also ignored by this approach.

The shield wall thicknesses are derived from design basis fuel defect of one percent and dose rate limitation of adjacent zones and are expected to provide adequate protection for abnormal conditions which may occur during normal plant operations. Q471.2

Zone designations are based on the annual occupational exposure limits, access requirements and occupancy time for the specific location in the plant as described in Section 12.4.1 and Table 12.4-1.

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NRC Letter: January 31, 1983

Question No. Q471.3 (Section 12.3.2)

Specify the design basis radiation level in the count room during normal operation and anticipated operational occurrences.

Response:

There is at least 5 feet of concrete between the counting room and process equipment in the auxiliary building. Therefore, the calculated dose rate in the counting room due to design basis operation is only 0.002 mRem/hr. Due to conservative assumption used in design basis shielding calculations (eg. 1 percent failed fuel) this calculation bounds any anticipated operational occurrences. This dose rate is sufficiently low to ensure proper operation and accuracy of counting room equipment. In fact, the dose rate is less than that due to natural background (0.005-0.01 mRem/hr).

NRC Letter: January 31, 1983

Question No. Q471.4 (Section 12.3.4)

For the area radiation and airborne radioactivity monitoring instrumentation, provide information on the instrument sensitivity, accuracy, precision, calibration methods and frequency, and alarm setpoints.

Response:

Refer to Table 12.3-2 for the sensitivity of area radiation monitors and Table 11.5-1 for airborne monitor sensitivity.

Refer to Section 12.3.4.2, Amendment 2, for a description of calibration methods and frequency.

For area monitors, the alert alarms are set at the area radiation zone limit and the high alarms are set at the high radiation area maximum dose rate. For airborne monitors, the high alarms are based on the activity concentrations specified in 10CFR20, Appendix B, Table I, Column I, and the alert alarms are set at one-tenth the high alarm values.

In accordance with ANSI N13.10-1974, area monitor precision is within ±10 percent at the 95 percent confidence level for mid-scale readings. Instrument error (accuracy) of the area monitors does not exceed ±20 percent over the upper 80 percent of the dynamic range.

dedicated microprocessor and in designated cases, a rate meter. The microprocessor provides local display and control functions for the detector, computes and stores time-averaged detector outputs, stores all necessary operating parameters (e.g., high and alert trip values), and also handles all communication between the detector and the RMS computer system. The rate meter, where provided, is located adjacent to the detector and provides local analog display. A high activity level is indicated by both audible and visible alarms which may be acknowledged at the rate meter.

The RMS computer system provides centralized display, control, and recording functions for all radiation monitors. A dual minicomputer system, located in the control building, polls each microprocessor every several seconds to obtain the latest readings, as well as registers any alarms present. All alarms are immediately displayed on the RMS control room CRT console, and recorded at once on the RMS hard-copy printer located in the control room. The system operator also uses the console to either output data from individual monitors or input commands to these monitors. All commands sent are also outputted to the hard-copy printer, and all detector outputs are recorded on random access discs and periodically recorded on a permanent sequential tape record. A second CRT console is located in the service building laboratory.

All alarms are annunciated both in the control room and locally at the microprocessor, and can be acknowledged at either location.

In addition, those monitors designated Category IE are connected directly to one of two control room IE cabinets. The output of each monitor is digitally displayed and also recorded on a two-pen recorder. A remote indication and control module (RIC) is furnished in the cabinets for each IE monitor. The RIC handles all remote control functions for IE monitors. The IE cabinets are connected by electronic isolators to the minicomputers to allow data from the IE monitors to be displayed on the control room console and also to allow data from IE monitors to be written into the RMS data base.

The area RMS is calibrated both by a standard factory calibration and by onsite calibration. Factory calibration includes checks for linearity and energy response. Sources traceable to NBS standards are used. Onsite calibration includes detector response using sources of known energy and strength. Onsite calibration is performed either locally or in the calibration laboratory in the service building. The frequency of onsite calibration of safety related monitors will be provided in the technical specifications. Q471.4

Table 12.3-2 gives the mark numbers, names, locations, and ranges of the 56 area monitors in the RMS. The following paragraphs provide a brief description of the different types of area monitors.

NRC Letter: January 31, 1983

Question No. Q471.5 (Section 12.3.4)

Describe the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of Section 70.24 of 10CFR70 for the storage area for new fuel.

Response:

The NRC has determined that an installation that incorporates a storage space design which precludes criticality need not comply with Section 70.24 of 10CFR70 and has readily granted exemption to applicants for their new fuel storage facilities based on design and procedural controls. (for example: MNP-2 Docket #70-2950). The Northeast Nuclear Energy Company (NNECO) will submit a request and provide information in support of an exemption from the requirements of 10CFR70.24 as provided in 10CFR70.24(d) as a part of the Special Nuclear Material License for storage only of unirradiated reactor fuel at Millstone-3.

NRC Letter: January 31, 1983

Question No. Q471.6 (Section 12.3.4)

Indicate whether, and if so, how, the guidance of Regulatory Guides 8.2, 8.8, and 8.12 and ANSI N13.1-1969 has been followed; if not followed, describe the specific alternative methods used.

Response:

Regulatory Guide 8.2 - Guide for Administrative Practices in Radiation Monitoring

The applicant complies fully with this Regulatory Guide and the referenced ANSI standard N13.2-1969 (same title). These guides are fairly limited in scope and are intended to provide some very general guidelines to companies just developing a radiation monitoring program. The radiation monitoring program at Millstone has been in effect for over 10 years and as indicated in response to 471.1 has been found acceptable when compared against guidelines that are much more detailed in scope than Regulatory Guide 8.2

Regulatory Guide 8.8 - Infomation Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA) - Revision 3 - 1978.

The applicant is in full compliance with the intent of each of the guidelines of Regulatory Guide 8.8 Sections 12.1 and 12.5 of the FSAR provide a description of the applicants ALARA program, radiation protection program and radiation protection facilities and hence describe the basic compliance with Regulatory Postions C.1, C.3 and C.4 of the Regulatory Guide. Because of a favorable evaluation of our ALARA program, Northeast Utilities was one of three utilities invited by INPO to help prepare INPO ALARA Good Practices. Hence, the ALARA procedures to be used at MIllstone 3 form the basis for many of the INPO Good Practices for an ALARA program.

In regard to Regulatory Position C.2, "Facility and Equipment Design Features", it should be noted that a significant percent of the station design was completed prior to the issuance of Regulatory Guide 8.8. Regardless, because of the significant experience of Westinghouse, Stone & Webster and Northeast Utilities in the design and operation of nuclear power plants and the growing concern in the 1970's for reducing occupational exposure, ALARA design principles were an important consideration in the design of Millstone Unit 3. To verify this, the applicant is currently performing a detailed review of the Millstone Unit 3 design against each of the individual ALARA design recommendations of Section C.2 of Regulatory Guide 8.8 If any potential ALARA concerns are identified,a cost-benefit evaluation is performed to determine if a plant design change is warranted. To date, the review is approximately 30% complete. Thus far, the review has confirmed that ALARA principles are inherent in the design of Unit 3.

Regulatory Guide 8.12 -Criticality Accident Alarm Systems Implementation of the guidelines contained in Regulatory Guide 8.12 has been described in the response to Question 471.5. ANSI N13.1-1969 -Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities The design basis of sampling airborne radioactive material implements the considerations described in ANSI N13.1-1969 regarding isokinetic probes, location of sample taps, and the concerns of deposition due to elbows and line losses.

The ANSI N13.1 recommendation of locating sampling points a minimum of five times the major dimension (diameter) downstream of a duct bend has been followed where possible. Where this is not possible, due to duct configuration, at least three diameters plus three feet has been provided downstream of a bend.

NRC Letter: January 31, 1983

Question No. Q471.7 (Section 12.4)

Provide the estimated inhalation exposure to personnel.

Response:

Appropriate radiological controls in hand with the plant ventilation design features will result in negligible inhalation exposure to personnel.

The design features of machinery access arrangement and ventilation intake and exhaust configurations are expected to preclude inhalation exposure during normal operation. The ventilation systems for the normally open buildings maintain clean air flow from the least potentially contaminated areas to the more potentially contaminated spaces. For isolated spaces such as the containment building, airborne contamination may be removed by installed recirculation cleanup systems and purge machinery. Areas which may have high airborne radioactivity concentrations are initially surveyed prior to personnel entry and protective clothing and breathing apparatus are prescribed as needed. Radiological protection procedures are contained in station procedures which describe the limitations to inhalation exposure and the measures established to prevent or control access to areas having the potential for airborne contamination.

NRC Letter: January 31, 1983

Question No. Q471.8 (Section 12.5.3)

Indicate whether, and if so, how, the guidance of Regulatory Guide 8.13 is being followed; if it will not be followed, describe the specific alternative approaches to be used.

Response:

The Applicant is in full compliance with Regulatory Guide 8.13 - Instructions Concerning Prenatal Radiation Exposure. The Regulatory Guide is presented during the General Radiation Worker Training Course and the annex to the regulatory guide is provided to the female employees. They are given the opportunity to ask any questions on the annex and then the option on whether or not to accept the normal radiation worker limits.

NRC Letter: January 31, 1983

Question No. Q471.9 (Section 11.3.2)

Provide a tabulation showing the concentrations of airborne radioactive material, by radionuclide, expected during normal and anticipated operational occurrences for equipment cubicles, corridors, and areas normally occupied by operating personnel.

Response:

Concentrations of airborne activity for the expected and design conditions in the containment structure turbine building, and fuel building are tabulated in FSAR Table 12.2-11. The concentrations based upon design conditions are expected to envelope anticipated operational occurrences. The airborne concentrations are averages based on assumed total leak rates described in NUREG-0017 and the ventilation rates for the respective buildings. Corridor and areas normally occupied by operating personnel are expected to have negligible airborne activity concentrations since clean air ventilation flow is normally directed from areas with less potential for contamination (manned areas) to areas with greater potential for contamination.

Equipment cubicles are the most likely areas for airborne concentrations, but are not normally occupied or accessible without prior survey and control. For purpose of quantification, the worst airborne concentration could conceivably exist in cubicles for which the combination of relatively high system volatile radionuclide concentrations and low cubicle ventilation rate would simultaneously exist for a given leak rate. A cubicle such as the letdown heat exchanger cubicle in the auxiliary building could develop airborne concentrations of approximately 7×10^{-5} μci/cc assuming all the design basis leak rate takes place in that cubicle for expected coolant radioactivity concentrations.

Based on the above assumption, it is expected that other cubicles will have airborne concentrations of less than 7×10^{-5} μci/cc.

NRC Letter: January 31, 1983

Question No. Q480.1 (Section 6.2.1.1.3)

Reference or provide a discussion of the administrative controls and/or electrical interlocks that would prevent the inadvertent operation of the containment heat removal system or other systems that could result in pressures lower than the external design pressure of the containment structure. Identify the worst single failure that could result in the inadvertent operation of the containment heat removal system.

Response:

As demonstrated in Section 6.2.1.1.3.5, inadvertant operation of the containment heat removal systems cannot reduce the containment pressure below the external design pressure of the containment structure. The evaluation conservatively assumed the quench spray system is actuated and reduces the containment temperature to 40°F, which is the minimum RWST temperature. The resultant containment pressure is higher than the minimum design pressure.

The only other system capable of lowering the containment pressure is the containment vacuum system. As noted in Section 9.5.10.3, excessive depressurization of the containment structure is not considered credible, since the containment vacuum pumps have a relatively small capacity when compared to the containment structure free volume. Furthermore, continued operation of the containment vacuum ejector after establishment of containment operating pressure is not considered credible because of control room indication and administrative controls.

No single active component failure nor single operator action can actuate the containment heat removal systems.

opposite side, and some are modeled as being insulated (ending boundary).

Resistance to heat transfer at the liner-concrete interface is considered in the containment analysis by use of a conservatively low value of thermal contact conductance of 100 Btu/hr/ft²/°F (Gido 1978). Since the steel liner is used as a form for pouring of the concrete, and since the concrete mix is very wet, the liner, in effect, becomes "glued" to the concrete.

The model considers transient heat conduction to the containment structure through the composite thermal resistance made up of the paint film on the steel liner, the liner itself, the liner-concrete interface, and the concrete. Section 6.2.1.1.3.2 discusses the mesh sizing for the passive heat sinks.

6.2.1.1.3.5 External Pressure

Inadvertent operation of the containment heat removal systems will cause a decrease in the pressure inside the containment, thereby increasing the normal external pressure differential on the containment structure.

The analysis of maximum external differential pressure assumes inadvertent actuation of the quench spray system caused by a single spurious containment depressurization actuation (CDA) signal. Note that two active component failures are necessary to generate a
Q480.1 spurious CDA signal.

The maximum external pressure differential is calculated by determining the minimum attainable pressure inside the containment and subtracting this value from the barometric pressure. The minimum
1 expected pressure is calculated to be 8.07 psia, based on the following assumptions:

1. Minimum initial air partial pressure — 8.9 psia

2. Normal maximum initial containment temperature — 100°F

3. Final containment temperature which equals the minimum RWST temperature — 40°F

See Table 6.2-3 for containment design evaluation parameters.

6.2.1.1.3.6 Loss-of-Coolant Accident Results

The loss-of-coolant accident (LOCA) containment transient analysis was performed using the LOCTIC computer code (Section 6.2.1.1.3.2) for a spectrum of pipe break locations and sizes.

The results of the peak containment pressure analysis are tabulated in Table 6.2-4. The mass and energy release data used for this analysis are given in Section 6.2.1.3. The initial containment

NRC Letter: January 31, 1983

Question No. Q480.2 (Section 6.2.1.1.3)

Specify the limiting containment conditions of temperature, pressure, and humidity for normal plant operation. Discuss the action that will be taken if these conditions are exceeded in the containment or locally, within a subcompartment.

Response:

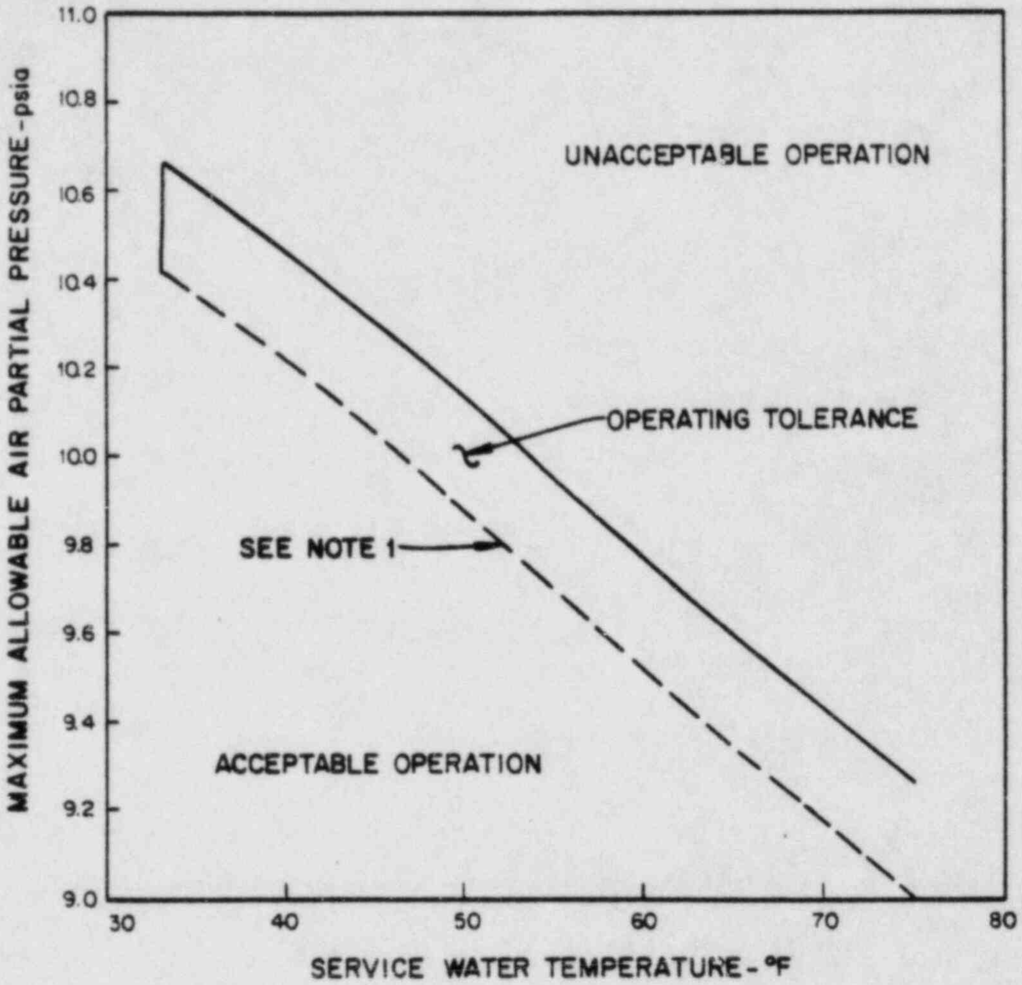
The limiting temperature and pressure conditions and action to be taken should these conditions be exceeded will be discussed in Section 16, Technical Specification, for normal plant operation. Limiting conditions will be specified only for bulk containment pressure and temperature. Local temperatures are not expected to differ significantly from the containment bulk temperature due to the operation of the Containment Air Recirculation System (see Section 9.4.7). There are no limits on the humidity inside the containment during normal plant operation.

The primary containment air partial pressure during normal operation is maintained between 9.0 and 10.4 psia. The setpoint value of this partial pressure depends primarily on the temperature of the service water which is used to cool the containment recirculation coolers. The allowable air partial pressure is given in Figure 480.2-1 as a function of service water temperature. The acceptable operating range is below and to the left of the curve, with the indicated tolerances.

Figure 480.2-1 has been established to provide information concerning where the air partial pressure must be maintained as a function of service water temperature to ensure depressurization within 60 minutes following a loss-of-coolant accident (LOCA). In the event of a LOCA the containment atmosphere is restored to subatmospheric conditions by cooling with the quench sprays and recirculation sprays. Since the recirculation spray is cooled by service water, the rate at which the containment atmosphere is cooled becomes a function of the service water temperature. The pressure to which the containment returns for a given service water temperature following a LOCA depends on the initial air partial pressure. Thus, for a given service water temperature, the containment air partial pressure must be regulated below a certain value to assure that the containment will become subatmospheric within 60 minutes following a LOCA.

The lowest setpoint value of air partial pressure in the containment is 9.0 psia. This corresponds to a total pressure of approximately 9.5 psia at an air temperature of 80°F and 80°F dew point.

The limiting conditions for operation and the actions to be taken if the limits are exceeded will be provided in the Technical Specifications at the time of their submittal.



NOTE:

1. SET POINT PRESSURE VALUE IN CONTAINMENT VACUUM SYSTEM INSTRUMENTATION
2. MINIMUM ALLOWABLE AIR PARTIAL PRESSURE IS 8.9 psia
3. AN OPERATING TOLERANCE OF + 0.25 IS INCLUDED AS SHOWN. OPERATION WITH THE CONTAINMENT AIR PARTIAL PRESSURE UP TO 0.25 psia ABOVE THE SET POINT VALUE IS ACCEPTABLE.
4. THE OPERATING CURVE REQUIRES THAT THE AVERAGE CONTAINMENT TEMPERATURE IS NOT LESS THAN 80° NOR GREATER THAN 120° F

FIGURE Q480.2-1
MAXIMUM ALLOWABLE AIR PARTIAL PRESSURE FOR SUBATMOSPHERIC CONTAINMENT
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

NRC Letter: January 31, 1983

Question No. Q480.3 (Section 6.2.2.3)

Several concerns have been identified relative to containment sump designs and their effect on long term cooling following a loss of coolant accident (LOCA). The staff is engaged in a generic program, designated as Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance", to resolve these concerns.

Draft NUREG-0897, which is currently under staff review, summarizes key technical findings related to USI A-43, provides recommendations for resolution of attendant safety issues, and provides guidance for the design and performance evaluation of the containment emergency sump. The proposed technical resolution includes recommended changes to Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems" and NRC's Standard Review Plan (NUREG-0800), Section 6.2.2, "Containment Heat Removal System" and 6.3, "Emergency Core Cooling System".

Pending the completion of Unresolved Safety Issue A-43, more immediate actions are required to assure greater reliability of safety system operation. We, therefore, require the following actions to provide additional assurance that long term cooling of the containment and reactor core can be achieved and maintained following a postulated LOCA.

1. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials, and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

   This inspection shall be performed at the end of each shutdown before containment isolation.

2. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

   Describe any changes you plan to make to reduce vertical flow in the neighborhood of the sump Ideally, flow should approach uniformly from all directions.

3. Evaluate the extent to which the containment sumps satisfy each of the positions of Regulatory Guide 1.82. The following additional guidance is provided for this evaluation:

Question No. Q480.3 (Cont)

a. Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump, the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

b. Estimate the exent to which debris could block the trash rack or screens. If a blockage problem is identified, describe the corrective actions you plan to take.

c. For each type of thermal insulation used in the containment, provide the following information:

   i. Type of material including composition and density,

   ii. Manufacturer and brand name,

   iii. Method of attachment,

   iv. Location and quantity in containment of each type,

   v. An estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

d. Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

Response:

1. The Applicant will have a surveillance procedure to ensure an inspection of the containment to identify debris which is capable of blocking the containment sump. The inspection will be performed prior to plant heatups and after major maintenance activities in the containment have been completed.

2. Examination of the containment geometry shows that all pipe break effluent is blocked from direct impingement on the containment water surface in the area of the sump by the internal containment structure. (Refer to Figures 3.8-59 and 3.8-60 for the internal containment design.) Liquid pipe break effluent is channeled toward the center of the containment and will not fall to the containment floor in the area of the sump, which is located adjacent to the containment wall. Steam from the pipe break will condense on the containment heat sinks and flow down to the containment floor. The

Question No. Q480.3 (Cont)

very small amount of water reaching the sump area in this manner has no effect on sump performance.

Most of the spray falls on the operating floor inside the crane wall and into the refueling cavity. A detailed review of the drainage paths indicates that the flow is generally channeled toward the center of the containment and does not fall on the containment floor in the area of the sump. A small portion of the spray falls outside the crane wall, through two layers of grating, to the containment floor. This will not affect the sump performance because of the small amount of water involved, and the dispersive nature of the flow.

The containment floor is designed to allow water to flow uniformly to the sump. A scale model test of the sump, including a detailed mock-up of the containment floor, has been performed. (Refer to Section 6.2.2.4.2.) The test led to the addition of grating above the pump suction inlets to serve as a vortex suppressor (see Figure 6.2-38). The final configuration was tested for all system operating conditions, and acceptable performance was demonstrated.

3. Refer to Section 6.2.2.2, and Section 6.2.2.3, Amendment 2, for the response to this question. Information is provided under the subheadings: Containment Recirculation System and Insulation.

Insulation

Reflective insulation is used on most piping within containment. Reflective insulation consists of multiple layers of Type 304 austenitic stainless steel sheets encased by inner and outer jacketing of type 304 austenitic stainless steel sheets. The minimum thickness of the inner and outer jacketing is 22 gage. Design details permit tight interlocking of adjacent sections of the assembled insulation. Where removal of insulation is required, quick release mechanical fasteners are provided.

In the unlikely event of a postulated pipe break, some insulation in the immediate vicinity of the break could possibly be removed by direct jet impingement. Since the insulation is fabricated and installed in overlapping sections, only sections in the immediate vicinity of the break would be affected.

Insulation that breaks away from the piping will fall to the floor below the piping and remain there. This is due to the weight of the reflective insulation and the low coolant velocity.

If insulation does become entrained in the coolant flow, it is extremely unlikely that the insulation will be transported to the containment sump, due to the torturous path that the insulation will be required to follow.

In the unlikely event that insulation should reach the containment sump, the design of the sump allows 50 percent blockage of the fine screening without loss of function.

Q480.3 The allowance for 50 percent plugging or blockage of the sump is conservative. Lighter particles will float on the water surface which will be above the screen assembly. Heavier particles will sink to the containment floor and will not be drawn into the screens due to the low inlet velocities which are provided.

Piping, 1 inch in diameter and under, requiring insulation, is insulated with calcium silicate and covered with stainless steel lagging. The lagging serves to minimize dislodging of insulation from the effects of a high energy pipe rupture, thereby, minimizing the potential for containment sump screen clogging. Some piping and equipment are insulated with encapsulated fiberglass blankets enclosed in stainless steel lagging. This is the "Nukon" system manufactured by Owens-Corning Fiberglas Corporation. Sump clogging tests have been conducted by Owens-Corning, which [illegible] dislodged "Nukon" blankets packed on the sump screen, with the [illegible] that they do not interfere with the recirculation cooling system ([illegible]eer 1978).

Small amounts of insulation, such as "Min-k" and "Foamglas", are utilized in areas where the installation of reflective insulation is impractical.

Q480.3 Concerning the effect of insulation particles on the recirculation pumps, based upon testing of similar pumps under plant conditions, with identical bearing configuration the presence of small particles in the

pumped fluid has no long-term effect on the operability and performance of the recirculation system pumps.

The recirculation system pumps are designed to accommodate the anticipated debris present in the containment sump. Parts of the pump running with close tolerances are provided with surface hardness and finish to withstand particulate matter.

Specifically the bearings of the pump are carbon/graphite type with flushing grooves. The pump's shaft sleeves are 304L stainless steel with hardfacing. The combination of the two provides a highly reliable bearing surface in the presence of particulates.

Similarly, the pump's wearing rings, both impeller and bowl, are Q480.3
hardened to provide wear resistance if particulate matter is in pumpage. The impeller wearing ring is hardfaced with a carbide material and the bowl ring is a heat treated stainless steel brought to a hardness to provide a difference in Brinell hardness between the stationary and rotating ring.

The design of the pump's sealing system precludes the effect of particulates in the pumpage. A tandem seal arrangement (two seals) is used with a seal fluid, demineralized water which is cool and clear of solids, to keep both seals clean. A nominal overpressure of 5 psig is maintained in the seal cavity so that if the inner seal should leak, flow is into the pumpage, and if the outer seal should leak, clean water is lost to the environment. A seal cooler and circulator ensures that the seal fluid is properly cooled.

Since the pump is open-line shafted, a minor amount of wear could be expected across bearings with a differential pressure, however, the degree in less than design and would not impare the function of the bearings.

It is not expected that the seals would see the effects of particles in the pumpage.

By design, the clearness provided along with the materials of construction ensure the long-term reliability of the pumps to perform as required.

6.2.2.3 Design Evaluation

The analyses of the effects of the containment heat removal systems on the containment structure are made using the LOCTIC computer program (Section 6.2.1).

The quench spray system sprays chilled water from the RWST into the containment atmosphere, and the containment recirculation system sprays cooled containment structure sump water into the containment atmosphere.

NRC Letter: January 31, 1983

Question No. Q480.4 (Section 6.2.3.3)

For analyses of the secondary containment system function following a high energy line break accident within the primary containment, provide the following information for each secondary containment volume (or justify not supplying the information):

1. Pressure and temperature as functions of time.

2. Primary containment wall temperature as a function of time.

3. Purge flow rate and recirculation flow rate as function of fan differential pressure.

4. Discussion of the manner in which heat transfer from the primary containment atmosphere to the secondary containment atmosphere is calculated, including a description of the heat transfer coefficients and material properties.

5. Initial conditions assumed for the secondary containment structure and atmosphere and justification therefor.

6. Manner in which equipment heat loads within the secondary containment are considered.

7. The decrease in the secondary containment volume due to thermal and pressure expansion of the primary containment structure, and a description and justification of the methods used to calculate the volume reduction.

Identify all high-energy lines within the secondary containment structure, and provide analyses of the pressure/temperature response for line ruptures for any of these lines that are not provided with guard pipes.

Response:

The thick, reinforced concrete primary containment walls and dome effectively isolate the containment enclosure building and contiguous buildings from high energy line breaks inside the primary containment. There will be no rise in pressure and temperature in the containment enclosure building or contiguous buildings from a high energy line break in the primary containment.

The supplementary leak collection and release system (SLCRS) does not perform its design function following a high-energy line rupture outside the containment. No credit is taken for the secondary containment system in the determination of radiological consequences from high-energy line ruptures outside the containment.

6. Regulatory Guide 1.29 for seismic design classification of system components. This system is classified as Seismic Category I.

7. Regulatory Guide 1.52 for air filtration and adsorption units as indicated in Section 1.8.1.52

Q480.4 Other design bases include:

1. The maintenance of negative pressure in areas contiguous to the containment

2. The filtration and adsorption by impregnated charcoal of contaminated air for radioactive iodine removal with minimum charcoal efficiency in excess of 99 percent of methyl iodine (CH_3I) and 99.5 percent of elemental iodine

3. The provision of continual monitoring for radioactive particulate in the discharge of the air at an elevated release point

4. Containment enclosure building integrity for normal wind load and frame integrity for tornado load; displacement integrity for LOCA conditions and containment integrated leakage rate test conditions

6.2.3.2 System Description

The containment enclosure building is Seismic Category I building and is comprised of structures with uninsulated metal siding and built up roofing over an insulated metal roof deck. The containment enclosure building design incorporates horizontal and vertical sliding joints to ensure that the integrity of the containment enclosure building will be maintained during maximum possible pressure transients of the containment structure under DBA conditions (Section 3.8.4). The design parameters for the containment enclosure building are listed in Table 6.2-64.

To provide the required air tightness within the enclosure building, the metal siding, metal deck side joints and end laps have two continuous lines of caulking at all joints. Neoprene gaskets and sheets are used to provide a flexible seal between the containment enclosure and the other buildings.

The supplementary leak collection and release system (SLCRS) is shown on Figure 6.2-46.

The principle components and design parameters for the SLCRS are listed in Table 6.2-63.

The SLCRS contains two 100-percent capacity exhaust fans, two 100-percent capacity filter banks, and related ductwork.

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. 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).

6.2.3.3 Safety Evaluation

The SLCRS is not normally in operation. The system starts on receipt of a SIS signal, and becomes operative within 15 seconds, including 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 60 seconds after the fan gets up to its designed speed, assuming a wind velocity of 22 mph. This velocity is not exceeded 95 percent of the time, based upon onsite meteorological records.

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. Q480.4

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

NRC Letter: January 31, 1983

Question No. Q480.5 (Section 6.2.5.2)

With regard to the hydrogen analyzers to be provided to continuously monitor the combustible gas concentrations within the containment following a LOCA, provide the following information:

1. A discussion of the operating principle and accuracy of the combustible gas analyzers;

2. A description of the tests conducted to demonstrate the performance capability of the analyzers or a reference to the report where such information may be found;

3. The locations of the multiple sampling points within the containment;

4. A discussion of the capability to monitor combustible gas concentrations within the containment independent of the operation of the combustible gas control systems; and

5. Failure mode and effects analyses of the containment combustible gas concentration monitoring systems.

Response:

At present the specifications for the hydrogen analyzer system (Cat I) are being developed independent of the DBA hydrogen recombiners.

Vendor selection and purchase orders are expected to be placed in June 1983.

Therefore, the response to this question will be provided in August 1983. This will include the FSAR amendment which will provide a system description of the containment hydrogen analyzer system.

NRC Letter: January 31, 1983

Question No. Q630.1 (Section 13.5.1)

Provide or reference the FSAR section that addresses the administrative procedures for the initial test program in accordance with SRP 13.5.1, Items I.B and II.B of NUREG-0800.

Response:

Refer to Section 13.5.1, Amendment 2.

13.5 PLANT PROCEDURES

The Station Superintendent issues administrative control procedures to implement auditable requirements or commitments governing station operation. Station procedures are written on a unit and station services level controlling the specifics of station operations, including specifications; maintenance and modification; periodic test, inspection, calibration, and special processes; and plant/equipment operating procedures.

13.5.1 Administrative Procedures

13.5.1.1 Conformance with Regulatory Guide 1.33

Regulatory Guide 1.33, issued February 1978, Quality Assurance Program Requirements, and ANSI 18.7/ANS 3.2 - 1976/Standard for Administrative Controls for Nuclear Power Plants, will be used as guidance for the preparation of administrative and station procedures.

13.5.1.2 Preparation of Procedures

Administrative control procedures are developed and in use for Millstone 1 and 2. Changes necessary to support Millstone 3 will be approved in accordance with the Technical Specifications (Chapter 16) at least 6 months prior to fuel load. Administrative control procedures are reviewed and approved by the Station Operations Review Committee (SORC) and the Station Superintendent. In addition, station procedures are issued to cover aspects of plant operation. Station procedures are approved by the Unit Superintendent in accordance with Technical Specifications for procedures specific to a single unit and by the Station Superintendent for common site procedures.

A startup manual approved by the SORC serves as an administrative control procedure to incorporate specific startup tests and administrative requirements specified in Section 14.2. Q630.1

13.5.1.3 Procedures

Administrative control and station procedures cover a wide range of topics. Major areas are described below.

<u>Standing Orders to Operating Personnel</u>

Administrative control procedures define the authorities and responsibilities of operating personnel. The procedures specify the number of personnel with reactor operator and senior reactor operator licenses required to be on site or at the controls in all plant conditions. (Figure 13.5-1 shows the area of the control room which has been designated "at the controls.") These procedures contain the necessary directives to implement Sections i, j, k, l, and m of 10CFR50.54. The administrative and station procedures direct the proper maintenance, review, and disposition of operating records,

EROLS QUESTIONS

MILLSTONE NUCLEAR POWER STATION - UNIT 3
DOCKET NO. 50-423

NRC Question	EROLS Section	Keywords
Geosciences Branch (GSB)		
231.1	2.5.2	Site Faulting investigations
Hydrologic Engineering Section (HGEB)		
240.1	2.4	Flood plains
240.2	7.1.9	Postulated core melt accident
Environmental Engineering Branch (EEB)		
E100.2	---	Differences between time of EROLS and ERCPS
E290.1	5.5.6	Copy of McDowell and Haalk 1975
E291.1	2.4	Tables 2.4-2 through 2.4-4
E291.2	2.4	Tables 2.4-2 through 2.4-5
E291.3	2.4	Metals
E291.4	2.4	Water quality program
E291.5	2.7	Noise measurement locations
E291.6	3.3.2	Service water
E291.7	3.3.2	Maximum monthly average
E291.8	3.4.1	Chlorine injection system
E291.9	3.6.2	Residual chlorine in the service water system
E291.10	3.6.2	FAC average concentration
E291.11	5.1	References: NUSCO, 1981b and Stone & Webster, 1976
E291.12	5.3	Service water interface with circulating water
E291.13	5.3	Reference for Section 5.3

EROLS QUESTIONS (Cont)

NRC Question	EROLS Section	Keywords
E291.14	12.0	Discharge outlet change
E291.15	12.0	NPDES permit
E291.16	12.0	Clean Water Act 316(a) thermal variance application
E291.17	12.0	Clean Water Act 316(b) demonstration
E291.18	App C	Effluent toxicity testing program
Siting Analysis Branch (SAB)		
E311.5	2.1.3	Population data in miles
Antitrust and Economic Analysis Branch (AEAB)		
E320.1	---	Production cost analysis
E320.2	---	30 and 40 yr present worth fuel and O&M costs
Radiological Assessment Branch (RAB)		
470.1	2.3.1	Distributional data
470.2	5.2.4	Dose rate estimates for man
470.3	5.2.4.4.1, 5.2.4.6	Dose rates
470.4	---	Invertebrate annual harvest

NRC Letter: January 31, 1983

Question No. Q231.1 (Section 2.5)

Revise the Environmental Report to accurately reflect the results of the site faulting investigations as presented in the FSAR. Page 2.5-2 (second and third paragraphs) of the ER is incorrect (based on the more detailed geologic submittal presented in the FSAR) as far as:

1. The number of faults mapped at the Unit 3 site,

2. The age of most recent faulting, and

3. The type of faulting.

Response:

EROLS Section 2.5 has been revised in Amendment 2 to reflect consistency with the FSAR.

2.5 GEOLOGY

For more detailed information, see FSAR Section 2.5.

2.5.1 Topography

The site is located on a small peninsula near the mouth of the Niantic River. The most striking topographic feature in the region around the site is the north-south trending ridges and valleys. The region is drained by a number of small brooks and the Thames, Niantic, and Connecticut Rivers. The maximum relief in the site area is approximately 61 meters (200 feet). Glacial till covers much of the bedrock surface on the hills and in smaller valleys.

Figure 2.5-1 shows the topography and surficial geology of the site vicinity. The site area slopes gently upward from Long Island Sound northward to an elevation of approximately 9 meters (30 feet). Wave action has eroded the blanket of till from the promontories of Millstone Point, exposing rock at several places. The reworked material was deposited as beach sand in the protected areas.

Much of the plant area has been graded and backfilled during the construction of the three units at Millstone Point.

2.5.2 Geology

The site is located in a geologically complex region characterized by metamorphosed and folded rocks of Ordovician-Silurian age. The area has been affected by four orogenies: the Avalonian (575 million years ago [m.y.a.]), the Taconian (465-445 m.y.a.), the Acadian (400-370 m.y.a.) and the Alleghenian (230-260 m.y.a.). The surrounding region has also been affected by rifting ranging from Triassic to Jurassic. Since then the region has been stable with the exception of epeirogenic uplift during the Cretaceous and Tertiary times and isostatic rebound resulting from the removal of the weight of ice covering the region during Pleistocene time.

The geology of the eastern portion of Connecticut is made difficult to decipher by the complex folding and faulting of the Late Paleozoic Era. The tectonic features of eastern Connecticut are shown on Figure 2.5-2. As shown on this figure the site lies on the east limb of the recumbent Hunts Brook syncline which mantles the Lyme Dome, just west of the site.

Much of the rock that underlies eastern Connecticut is a series of Early Paleozoic metavolcanic and meta-sedimentary rocks and granitic gneisses (Goldsmith and Dixon 1968). The site is underlain by the Monson Gneiss of pre-Silurian age and the Westerly Granite of Pennsylvanian or younger age. The Monson Gneiss is thinly layered with light feldspathic and dark biotitic and hornblendic layers (Goldsmith 1967). The foliation is well developed in the site area with an average trend of N67W and dips at 48 degrees northeast. The Monson Gneiss has been intruded by granitic and pegmatitic sills related to the Westerly Granite. | 2 Q231.1

A number of faults are prominent in Eastern Connecticut (Figure 2.5-2). The Honey Hill - Lake Char fault system lies approximately 24 km (15 miles) north of the Millstone site. The east-west segment of Honey Hill dips at low angles to the north; the Lake Char section dips westward at approximately 10 degrees (Dixon and Lundgren 1968). The system is thought to have been active beginning in Middle or Late Devonian time and continued into the Permian Period (170-225 m.y.a). The latest episode of faulting is related to the Jurassic-Triassic rifting which resulted in the formation of the Jurassic-Triassic basin and a number of other high angle normal faults throughout southern New England. These faults generally trend north-south. | 2 Q231.1

Juro-Triassic rifting is evident at the Millstone site. Eleven fault zones with numerous minor associated faults have been uncovered during excavation, and all but one of these features are normal faults trending approximately north-south. A small thrust fault was found, which was related to, or is older than, the Jurassic activity. | 2 Q231.1

Samples of the fault gouge found in the fault zones were radiometrically age dated. The results indicate that the last activity associated with the faulting occurred approximately 142 m.y.a. Therefore, the faulting at the site is considered to be incapable. | 2 Q231.1

During the Pleistocene epoch, all of New England was covered with ice. The topography was not greatly altered by glaciation, but the ice scoured the land, leaving scattered deposits (Figure 2.5-1) throughout the area after advancing to and beyond the southern New England coast. The ice began to retreat from the Connecticut coast approximately 15,000 years ago (Flint 1975).

2.5.3 References for Section 2.5

Dixon, H.R. and Lundgren, L., Jr. 1968. Structure of Eastern Connecticut. In: Studies of Appalachian Geology: Northern and Maritime, Zen (Ed.), John Wiley and Sons Inc., New York, p 219-230.

Flint, R.F. 1975. The Surficial Geology of Essex and Old Lyme Quadrangles. State Geological and Natural History Survey of Connecticut, Quadrangle Report No. 31.

Goldsmith, R. 1967. Bedrock Geologic Map of Niantic Quadrangle. U.S. Geological Survey, Quadrangle Map GQ-575.

Goldsmith, R. and Dixon, H.R. 1968. Bedrock Geology of Eastern Connecticut. In: Guidebook for Field Trips in Connecticut, New England Intercollegiate Geologic Conference, 60th Annual Meeting, Yale University New Haven, Conn., F-O, p 1-5.

NRC Letter: January 31, 1983

Question No. Q240.1 (Section 2.4)

Description of floodplains, as requested by Executive Order 11988, Floodplain-Management, have not been provided. The definition used in the Executive Order is:

Floodplain: The lowland and relatively flat areas adjoining inland and coastal waters including flood prone areas of offshore islands, including at a minimum that area subject to a one percent or greater chance of flooding in any given year.

a. Provide descriptions of the floodplains adjacent to the site. On a suitable map(s) provide delineations of those areas that will be flooded during the one percent (100-year) flood, both before and after plant construction or operation.

b. Provide details of the methods used to determine the floodplains in response to a. above. Include your assumption of, and basis for, the pertinent parameters used in the computation of the flood flows and water elevations. If studies approved by the Federal Insurance Administration (FIA) are available for the site and other affected areas, the details of the analyis used in the reports need not be supplied. You can instead provide the reports from which you obtained the floodplain information.

c. Identify, locate on a map and describe all plant structures and topographic alternations in the floodplains.

d. Discuss the hydrologic effects of all items identified in response to c. above. Discuss the potential for altered flood flows and levels, offsite. Discuss the effects on offsite areas of debris generated from the site during flood events.

e. Provide the details of your analysis used in response to d. above. The level of detail is similar to that identified in item b. above.

Response:

The response to this question will be submitted in June 1983.

NRC Letter: January 31, 1983

Question No. Q240.2 (Section 7.1.9)

Calculate the radiological consequences of a liquid pathway release from a postulated core melt accident. The analysis should assume, unless otherwise justified, that there has been a penetration of the reactor basemat by the molten core mass, and that a substantial portion of radioactivity contaminated sump water was released to the ground. Doses should be compared to those calculated in the Liquid Pathway Generic Study (NUREG-0440, 1978). Provide a summary of your analysis procedures and the values of parameters used (such as permeabilities, gradients, populations affected, water use). It is suggested that meetings with the staff of the Hydrologic Engineering Section be arranged so that we may share with you the body of information necessary to perform this analysis.

Response:

As stated in Section 7.1.9 of the Millstone 3 EROLS, "The analyses of the probabilities and consequences of accidents beyond the design bases of the Millstone 3 plant will be comprehensively discussed in the relevant portions of the Millstone 3 Probabilistic Safety Study (PSS), which the applicants presently estimate will be completed within 6 months after the docketing of the FSAR" (August 1983).

NRC Letter: January 31, 1983

Question No. QE100.2

In addition to other requested information, provide a summary and brief discussion, in table form, by section, of differences between currently projected environmental effects (including those that would degrade and those that would enhance environmental conditions) and the effects discussed in the environmental report and environmental hearings associated with the construction permit review. On a similar basis, indicate changes in plant or plant component design, location or operation that have been made or planned since the construction permit review.

Response:

A review of the ERCPS and EROLS will be conducted to document substantive informational changes since 1973 (ERCPS). Significant changes in design, new information, and changes in conclusions or impacts will be presented in Tabular (matrix) form in June 1983.

NRC Letter: January 31, 1983

Question No. QE290.1 (Section 5.5.6)

Provide a copy of reference, McDowell and Haalk 1975.

Response:

A copy of the following reference has been provided under separate cover on April 1, 1983.

McDowell and Haalk 1975. Report on Expected Impact on Wildlife and Forest Due to a Proposed Widening of the Current Right-of-way of the Millstone - Manchester 345 kV Line.

NRC Letter: January 31, 1983

Question No. QE291.1 (Section 2.4)

Indicate the bases for the data presented in ER Tables 2.4-2 through 2.4-4. That is, indicate whether the data are averages for all sampling locations; indicate whether the data in these tables and Table 2.4-5 are depth composites or surface or subsurface grab samples.

Response:

Please refer to Figure 2.4-13, WATER QUALITY SAMPLING LOCATIONS.

In Table 2.4-2, data are averages of the sampling results at the following locations (stations) during various tidal phases:

For January 1974

Station No.	Tidal Phase: Ebb	Tidal Phase: Flood
2	S	S
3	S	S
4	S	S
5	S	NS
6	S	S
7	NS	S
8	S	S
9	NS	S

NOTE: S = Sampled, NS = Not Sampled

For February through December 1974

Station No.	Tidal Phase: High Slack	Max. Ebb	Ebb	Low Slack	Max. Flood	Flood
1	S	NS	NS	S	NS	NS
2	S	NS	NS	S	NS	NS
3	NS	S	NS	S	NS	NS
4	S	NS	NS	NS	S	NS
5	NS	NS	S	NS	NS	S
6	NS	NS	S	NS	NS	S
7	NS	NS	S	NS	NS	S
8	NS	NS	S	NS	S	NS
9	NS	NS	S	NS	NS	S

NOTES:

S = Sampled, NS = Not sampled.
Samples from Stations 3 and 4 were not analyzed on ammonia-nitrogen, nitrite-nitrogen, nitrate-nitrogen, organic-nitrogen, total phosphate, ortho-phosphate, condensed phosphate, and organic carbon.

In Table 2.4-3, data are averages of the sampling results at the following locations (stations) during various tidal phases:

For January through December 1974

Station No.	Tidal Phase: High Slack	Ebb	Low Slack	Flood
1	S	NS	S	NS
2	S	NS	S	NS
5	NS	S	NS	S
6	NS	S	NS	S
7	NS	S	NS	S
8	NS	S	NS	S
9	NS	S	NS	S

NOTE: S = Sampled, NS = Not Sampled

In Table 2.4-4, data are averages of the sampling results at the following locations (stations) during various tidal phases:

For January 1974

Station No.	Tidal Phase: Ebb	Flood
2	S	S
3	S	S
4	S	S
5	S	NS
6	S	S
7	NS	S
8	.	S
9	NS	S

For February through December 1974

Station No.	Tidal Phase: High Slack	Max. Ebb	Ebb	Low Slack	Max. Flood	Flood
1	S	NS	NS	S	NS	NS
2	S	NS	NS	S	NS	NS
3	NS	S	NS	S	NS	NS
4	S	NS	NS	NS	S	NS
5	NS	NS	S	NS	NS	S
6	NS	NS	S	NS	NS	S
7	NS	NS	S	NS	NS	S
8	NS	NS	S	NS	S	NS
9	NS	NS	S	NS	NS	S

The sample depths for Tables 2.4-2 through 2.4-4 are indicated on Figure 2.4-13.

The samples for Table 2.4-5 were collected from subsurface water (about 10 inches below the surface).

NRC Letter: January 31, 1983

Question No. QE291.2 (Section 2.4)

Indicate or provide a reference for the maximum values and their locations by months for the parameters shown in Tables 2.4-2 through 2.4-5.

Response:

The maximum values and their locations by months for parameters in Tables 2.4-2 through 2.4-5 are presented in the following tables (Table QE291.2-1 to Table QE291.2-4):

TABLE QE291.2-1

MAXIMUM CONCENTRATIONS OF PARAMETERS FOR LONG ISLAND SOUND WATER[6]

Parameter	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec
Ammonia-Nitrogen	0.12/[1] 2S	0.02[1]	0.08	0.04	0.16/ 2B	0.20/ 6M	0.28/ 5M	0.30/ 6M	0.40/ 2M	0.38/ 2S	0.14/ 9M	0.08 1M
Nitrite-Nitrogen	ND	0.001	0.004	0.002	0.002	0.001	0.001	0.002/ 6M	0.010	0.080	0.018/ 8B	0.003/ 1M
Nitrate-Nitrogen	0.95/ 2B	0.83/ 2S	0.96/ 2B	0.64/ 7M	1.46/ 8M	0.10	0.48/ 8S	0.08/ 6M	0.14/ 8S	0.076/ 8B	0.038/ 5M	0.17/ 6M
Organic-Nitrogen	0.42/ 2M	0.58/ 2S	0.36/ 1M	0.56/ 7M	0.60	0.70/ 1M	0.50	2.0/[5] 2S	1.2/ 9M	0.84/ 1M	0.706/ 8M	0.32
Total Phosphate	0.80/ 5M	0.7	0.4/ 2M	0.4/ 1M	0.1	0.9/ 2M	0.2	0.2	0.4	0.2	0.3/ 1M	0.3/ 8B
Ortho Phosphate	ND	0.3/ 8M	<0.1/ 8B	0.2/ 1M	<0.1	0.1	<0.1	<0.1	0.2/ 2M	0.1/ 8B	<0.1	0.1
Condensed Phosphate	0.18/ 2B	0.5/ 2S	0.1	0.3/ 1M	<0.1	<0.1	<0.1	0.2/ 7M	0.2/ 7M	0.2/ 9M	0.2	0.1
Organic Carbon	61.0/ 2M	111.0/[3] 1M	100.0/ 2B	20.0/ 6M	25.0/ 7M	21.0/ 2B	13.5	19.0/ 2M	41.5/ 8B	30.2/ 2S	9.3/ 9M	32.0
Oil and Grease	89.8/ 2M	29.44/ 3M	18.68/ 9M	22.28/ 8M	15.16/ 8S	29.42/ 1M	8.50/ 4B	14.16/ 6M	12.02/ 9M	7.88/ 4B	6.10/ 2B	8.16/ 3B
Sulfates	2,750/ 7M	2,600	2,800	2,550/ 6M	2,450/ 8B	2,900/ 1M	3,240/ 4S	2,850/ 3M	3,300/ 2B	2,425	2,400	2,525
MBAS	0.396/ 2S	0.60/[4] 2M	0.15/ 2B	0.45/ 6M	0.20/ 2M	0.225/ 1M	0.045	0.077	0.115/ 6M	0.240	0.133	0.224/ 3S
Suspended Solids	184/[2] 2M	54.1/ 3B	87.6/ 6M	45.0/ 3B	33.3/ 3B	45.55/ 6M	43.7/ 2B	53.5/ 3S	50.7/ 3B	38.4/ 3M	41.3/ 2B	38.6/ 7M

NOTES:

1. A/BC, where A - concentration in mg/l
   B - station number (see Figure 2.4-13)
   C - depth (B - bottom water, M - mid-depth water, S - surface water)

   e.g., 0.12/2S means the maximum value of 0.12 mg/l was recorded at Station No. 2, surface water sample.

TABLE QE291.2-1 (Cont)

When the same maximum concentration was recorded in more then one location, only concentration is shown, e.g., the maximum value of 0.02 was recorded at 2M, 8B, and 9M.

2. Unusualy high; the next high is 47.6/3B

3. Unusually high; the next high is 10.0/2B

4. Unusually high; the next high is 0.12

5. Unusually high; the next high is 0.66

6. Based on data collected monthly during 1974 water quality monitoring program by TRC (The Research Corporation of New England)

TABLE QE291.2-2

MAXIMUM CONCENTRATIONS OF QUARTERLY SAMPLES
FOR LONG ISLAND SOUND WATER

Parameter	March	June	September	December
Total alkalinity	276.8/2M[1]	246/2S	248	272/7M
Chloride	19,460/2B	19,375/1M	24,375/2B	18,750
Potassium	650	715/8M	575	675
Calcium	270	269	244	250
Magnesium	825/8S	1,805/8M	1,170/8M	900/1M
Arsenic	ND	ND	ND	ND
Molybdenum	0.60	1.0/1M	0.056/9M	ND
Titanium	ND	ND	ND	0.32/8M
Vanadium	0.40/7M	0.20	0.027/2S	ND
Cadmium	0.05	0.09/8M	0.013	ND
Berylium	ND	ND	ND	ND
Mercury	ND	ND	ND	ND
Total solids	35,750/2B	43,500/2B	41,017/7M	40,800/9M
Volatile solids	7,400/2B	14,950/2B	24,012/5M	12,800/6M
Tin	19.8/2M	ND	ND	ND
Phenol	0.009	ND	0.003	ND

NOTES:

A/BC, where A - concentration in mg/l
B - station number (see Figure 2.4-13)
C - depth (B - bottom water, M - mid-depth water, S - surface water)

When the same maximum concentration was recorded in more than one location, only concentration is shown.

TABLE QE291.2-3

MONTHLY MAXIMUM METAL CONCENTRATION

Month	Tide	Parameter (Unfiltered Samples)				
		Iron	Manganese	Nickel	Zinc	Aluminum
Jan.	Ebb	0.25/3S[1]	0.06/5M	0.20	0.014/3M	2.60/8M
	Flood	0.25/3M	0.05/8S	0.20/6M	0.013/3B	3.20/8M
Feb.	Ebb	0.13/4S	0.035	0.14/8S	0.005/8B	1.3/9M
	Flood	0.40/3B	0.045	0.12/7M	0.007/3B	1.3/8S
Mar.	Ebb	0.70/5M[2]	0.062/8B	0.30	0.030/2B	1.7/8S
	Flood	0.19	0.095/3B	0.30	0.013	2.1/8S
Apr.	Ebb	0.25/3B	0.050	0.18/9M	0.050/3M	5.0/7M
	Flood	0.18/3B	0.050	0.24/8M	0.025/6M	2.7/4B
May	Ebb	0.22/1M	0.04	0.22	0.22/6M	2.85/9M
	Flood	0.18	0.04	0.27	0.205/9M	2.36/8S
June	Ebb	0.14/1M	0.075	0.19/2M	0.013	1.00/6M
	Flood	0.14	0.11/2S	0.16/2S	0.011/4B	1.00/6M
July	Ebb	0.17/4M	0.05	0.10	0.043/1M	10.0/4M
	Flood	0.10	0.03	0.10	0.035/7M	7.5/3S
Aug.	Ebb	0.11	0.05/4S	0.035	0.013/7M	2.8/9M
	Flood	0.11/8M	0.042/1M	0.065/9M	0.010/8B	2.1/3S
Sept.	Ebb	0.28/2S	0.04	0.073/9M	0.010/4B	7.2
	Flood	0.48/6M	0.05/2S	0.073	0.020/3M	2.8/1M
Oct.	Ebb	0.14/5M	0.02	0.28/4M	0.035/6M	ND
	Flood	0.14/3S	0.04/7M	0.10/8S	0.057/9M	ND
Nov.	Ebb	0.02	0.01	0.05/4S	0.025	ND
	Flood	0.02	0.01	0.05	0.055	1.00/2M
Dec.	Ebb	0.09/8S	ND	ND	0.040	0.6/9M
	Flood	0.09	ND	ND	0.050/3M	0.6

NOTES:

(1) A/BC where A - Concentration in mg/1
B - Station number (see Figure 2.4-13)
C - Depth (B - bottom water; M - mid-depth water, S - surface water)

(2) Unusually high; the next high is 0.19

TABLE QE291.2-4

MAXIMUM ANNUAL METAL CONCENTRATIONS IN ppb (µg/l) IN UNFILTERED SEAWATER SAMPLED FROM SELECTED SITES NEAR MILLSTONE POINT, CONNECTICUT[1]

Selected Site	1980	1979	1978	1977	1976	1975	1974	1973
				Copper				
Unit 1 Intake	2.4	3.9	3.3/3.6[2]	3.2/7.6	20/4.5	8/<1	15	9
Quarry Cut	3.6	5.0	6.0/3.9	4.8/4.1	10/2.8	12/<1	15	45
Giants Neck	2.8	4.4	4.0/2.4	5.0/13	16/7.9	12/<1	18	10
Twotree	2.2	3.8	3.6/0.7	5.7/4.1	12/3.2	8.9/<1	16	14
				Zinc				
Unit 1 Intake	6.7	9.2	15.4/1.7	6.9/4.7	26/2.7	26/18	14	10
Quarry Cut	5.1	8.4	16.5/1.5	8.4/4.5	20/5.3	15/4	16	15
Giants Neck	8.5	10.4	29/5.3	10.0/24	35/22	28/5.9	15	19
Twotree	5.6	9.6	11.8/1.5	11.0/5.2	39/4.3	24/6.4	9	75
				Iron				
Unit 1 Intake	280	130	3.7/105	1.7/490	2.4/195	1.6/148	173	122
Quarry Cut	333	130	0.9/133	3.2/400	2.4/195	10.5/101	197	721
Giants Neck	465	254	1.3/413	3.1/3,800	3.5/300	1.6/88	223	536
Twotree	257	133	1.7/128	1.1/310	8.9/251	1.7/146	241	348
				Chromium				
Unit 1 Intake	<1	<2	<2/<2	<1/<1	<1/<1	<1/2	6	4
Quarry Cut	<1	<2	<2/<2	<1/<1	2.4/<1	<1/2	3	4
Giants Neck	<1	<2	<2/<2	<1/10.0	<1/<1	<1/2	4	8
Twotree	<1	<2	<2/<2	<1/<1	1.8/<1	<1/2.7	2	5
				Lead				
Unit 1 Intake	1.3	3.7	8/<1	3.6/3.3	1.3/6.7	2/1	4	8
Quarry Cut	2.3	3.1	1.5/11	<1/3.0	14/12	1/1	15	5
Giants Neck	2.3	2.8	1.2/6.7	2.4/1.0	1/8.3	3/6	6	4
Twotree	1.7	2.1	2.0/<1	1.9/4.0	13/21	1/1	4	5

NOTES:

(1) For the years 1975 through 1978 the data are available only for soluble and insoluble constituents separately.

(2) A/B, where A is soluble concentration
B is insoluble concentration

NRC Letter: January 31, 1983

Question No. QE291.3 (Section 2.4)

The information on metals in Section 2.4 doesn't always indicate the location when metals concentration values are discussed. These data are helpful when reviewing this section and should be provided.

Response:

As discussed in EROLS Section 2.4.3.4.3 Trace Metals, an ongoing monitoring program was established by NUSCO to determine potential impacts of the Millstone Nuclear Power Station on trace metal concentrations in Long Island Sound. Samples were collected from four locations in the vicinity of the station: Millstone 1 Intake; Quarry Cut (station discharge); Giants Neck; and Twotree Island.

However, in presenting the 1974 baseline study results, the location of the sampling was not always indicated. It is supplemented by the following. Please refer to Figure 2.4-13 for station number designation.

Iron

Iron concentration between 100 and 200 ppb were record in September 1974 at the following stations:

Ebb Tide: 2M, 4M, 8S, 8B, 7M, 9M[1]

Flood Tide: 1M, 2S, 3B, 4B, 8M, 8B, 7M

The concentration greater than 200 ppb was recorded at the following sampling location:

2S (280 ppb), 3S (280 ppb), 3M (220 ppb), 4S (220 ppb), and 6M (480 ppb)

Lead

Lead concentration greater than 5 ppb were recorded in 5 of 24 samples which were collected from the Quarry Cut and the Millstone 1 Intake between February 1973 and February 1974.

The maximum concentration of 15 ppb was detected in the Quarry Cut sample.

Molybdenum

Molybdenum concentration of 1 mg/l was detected in the Quarry Cut (station discharge) samples in June 1974.

Cadmium

Soluble cadmium concentration of 13 g/l was reported in 9 of 44 samples collected in September and December 1974.

These samples were collected at the following stations:

Ebb Tide: 5M, 6M, 7M, 8S, 8B

Flood Tide: 5M, 6M, 8B, 8S

Phenol

Phenol concentration of 9 g/l was detected in the 8S and 8B samples in March 1974.

Phenol concentration of 3 g/l was detected in the 2S and 8M samples in September 1974.

In all other cases, if the location in Section 2.4 is not specified when concentrations of metals are discussed, the data are applicable to all locations or they are averages of the data collected from each location.

NOTES:

1. B = Bottom water sample
   M = Mid-depth water sample
   S = Surface water sample

2. Refer to Figure 2.4-13 for station numbers

NRC Letter: January 31, 1983

Question No. QE291.4 (Section 2.4)

The information in Section 2.4 and elsewhere in the ER is referenced to the 1974 water quality program. The ER in Section 6 indicates that more recent surveys have been conducted. Indicate whether and to what degree the analyses and conclusions drawn in the ER with regard to site area water quality would be changed by this additional information.

Response:

The comprehensive baseline water quality survey was conducted during 1974 only by The Research Corporation of New England, Inc. (TRC). The objective of more recent surveys is to determine possible contributions by the Millstone Station condenser cooling system to the heavy metal concentrations in adjacent Long Island Sound. The heavy metals selected for analyses in seawater include copper, zinc, iron, chromium, and lead. The results are discussed in Section 2.4.3.4.3 and presented in Table 2.4-5. The analyses and conclusions drawn in the ER with regard to site area water quality are not changed by this additional information (NUSCO 1982).

Reference:

NUSCO 1982. Annual Report 1981. Monitoring the Marine Environment of Long Island Sound at Millstone Nuclear Power Station, Waterford, Conn.

NRC Letter: January 31, 1983

Question No. QE291.5 (Section 2.7)

Indicate the reference points (e.g., site property boundary, reactor building centerlines) for the noise measurement locations.

Response:

Distance to the community noise measurement locations identified in Figure 2.7-1 and described in Table 2.7-1 were measured directly from a USGS map of the area. The USGS map is a 1970 photorevised edition with only the location of the Millstone 1 Nuclear Power Station clearly identified. The reference point for the community measurement locations is clearly identified on Figure 2.7-2 Amendment 2, which is a copy of the USGS map and represents the intersection of the Millstone 1 turbine building and the reactor building which provides the only distinguishable reference point for the Millstone 1 Station.

2.7 NOISE

2.7.1 Site Characteristics

The Millstone station, with two of its units operating and the third under construction, is situated on the tip of a small peninsula extending southward into Long Island Sound. Small residential communities, as well as Route 156, bound the site to the north and northeast. The residential areas are mostly year-round suburban communities, but many of the commercial businesses and other land uses are centered on summer tourism and vacationing. The resulting seasonal effects include slight increases in population and traffic volume during the summer months.

Noise-sensitive areas were determined through the use of United States Geological Survey maps and an inspection of the site environs by the survey personnel. Measurement locations were chosen in the residential communities of Jordan Cove, Pleasure Beach, Millstone Road, and Black Point to ensure that a complete and accurate description of the ambient sound levels could be drawn for all areas in the vicinity of the Millstone site, and that a comparison of sound levels could be made with those obtained from previous noise surveys at similar locations.

In all, eight measurement locations were selected as representative of the different noise-sensitive areas surrounding the station. Two locations were chosen in each of the Jordan Cove and Pleasure Beach areas. These communities have an unobstructed view of the Millstone site, and plant noise is generally audible. Three locations were chosen in the Black Point area and a single location in the Millstone Road community, where the plant was generally not audible. The eight measurement positions described in Table 2.7-1 are shown on Figures 2.7-1 and 2.7-2. 2 QE291.5

2.7.2 Ambient Sound Levels

The statistical descriptors selected to delineate the ambient sound levels include residual, equivalent, and day-night sound levels. Residual sound levels are represented by the L_{90} percentile level, which is the sound level exceeded 90 percent of the time. This residual level represents the minimum or background sound level. The equivalent sound level (L_{eq}) is the level of steady noise which would have the same total sound energy as the fluctuating noise actually measured in the community. The day-night sound level (L_{dn}) is similar to L_{eq}, but has a 10-dB weighting applied to noise occurring during the night since nighttime noise is considered more annoying than the same noise during the day. The L_{eq} is calculated by combining the daytime hourly L values for the 15-hour period from 0700 to 2200 hours with the 10 dB weighted nighttime hourly L_{eq} values for the 9-hour period from 2200 to 0700 hours. Table 2.7-2 provides a statistical summary of hand-held and automatically monitored measurements at each site. Table 2.7-3 furnishes more detailed information.

Bride Lake
Dodge Pond
Mt Ararat
McCook Point
8
Wigwam Rock
Crescent Beach
N I A
7
Crescent Park
B
Pond Pt
Threefoot Rock
Attawan Beach
ROCKY NECK STATE PARK
PATAGANSSET
Giants Neck
Watts Island
Huntley Island
4
Black Point Beach Club
ands End
Seal Rock
Griswold Island
North Brother
Long Rock
South Brother
Johns Rock
Blackboys
Black Point
0
0.5
1
SCALE-MILES
0
1
2
SCALE-KILOMETERS
LEGEND:
SOUND LEVE

MONITORING STATIONS

FIGURE 2.7-2
LOCATIONS OF SOUND LEVEL MEASUREMENTS IN THE
MILLSTONE STATION AREA
MILLSTONE NUCLEAR POWER STATION
UNIT 3
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

NRC Letter: January 31, 1983

Question No. QE291.6 (Section 3.4.2)

Indicate the temperature rise of the service water upon passage through the unit under normal operating and under shutdown conditions.

Response:

The temperature rise of the service water under all operating conditions is given in Table 3.4-1 as revised in Amendment 2.

TABLE 3.4-1

SERVICE WATER FLOW AND HEAT LOAD REQUIREMENTS UNDER ALL OPERATING CONDITIONS

	Normal Operating Condition	Normal Unit Cooldown Condition	DBA Coincident with LOP: Minimum Engineered Safety Features	DBA Coincident with LOP: Normal Engineered Safety Features	Loss of Power (LOP): Hot Shutdown	Loss of Power (LOP): Cold Shutdown	
Flow (gpm)	27,288	27,426	15,037	29,484	20,898	10,883	2 QE291.6
Flow (lb/hr)	1.516×10^7	1.523×10^7	8.3×10^6	1.62×10^7	1.18×10^7	6.1×10^6	
Heat Load $\left(10^6 \frac{BTU}{hr}\right)$	213.72	235.74	429.71	855.82	160.72	93.68	2 QE291.6
ΔT°C (F)	8.2 (14.8)	9.1 (16.3)	30.0 (54.0)	30.5 (54.9)	8.1 (14.5)	9.1 (16.3)	2 QE291.6

NRC Letter: January 31, 1983

Question No. QE291.7 (Section 3.3.2)

Indicate the reason for using the term "maximum monthly average" when referring to the continuous flow values (in gpm) in Table 3.3-1.

Response:

Please refer to Section 3.3.2 as revised (Amendment 2).

3.3 STATION WATER USE

3.3.1 Water Sources

Seawater withdrawn from Long Island Sound supplies the once-through circulating water system and the service water system. Freshwater from the Town of Waterford public water system supplies the unit's domestic water system. fire protection system, and makeup water treating system.

The makeup water treating system supplies deionized water to the primary grade water, auxiliary boiler, condensate storage, and auxiliary feedwater system.

3.3.2 Water Uses

A schematic diagram of station water use is shown on Figure 3.3-1.

Table 3.3-1 shows the maximum, average, and minimum use under normal operating conditions or during temporary shutdown for the various plant water systems. Under normal operating conditions, water use will vary because systems are dependent upon loadings and demands. Temporary shutdown will occur when the reactor is shut down for relatively long periods, such as during maintenance and refueling. Some of the plant systems will be operating at reduced capacities (Table 3.3-1) during a temporary shutdown. 2 QE291.7

3.3.2.1 Cooling Systems

The circulating water system for Millstone 3 will draw water from the Niantic Bay area of Long Island Sound through the circulating and service water pumphouse (located on the west side of Millstone Point and north of the Millstone 2 intake). The water is pumped through the condenser to condense steam exhausted from the turbine generator. During its passage through the condenser, the circulating water will be heated to approximately 9.4°C (17°F) above its inlet temperature.

Small amounts of radioactive wastes are released after treatment to reduce the radioactivity to a level below the concentrations permitted by Federal Regulations (Section 3.5). Circulating water, after passing through the condenser, will be used to dilute the treated radioactive waste and small quantities of chemical wastes. Sections 3.6 and 3.7 summarize the maximum cumulative chemical releases expected from operations at the site.

The heated circulating water will be discharged into the quarry (which is located on the southeast extremity of Millstone Point) where it will be combined with the heated discharges from Millstone 1 and 2. Section 3.4 gives details of the heat dissipation system.

The service water, in a manner similar to the circulating water, will be used as the coolant for various heat exchangers and will not come in contact with radioactive material or components in the unit. After passing through these heat exchangers, the service water will

NRC Letter: January 31, 1983

Question No. QE291.8 (Section 3.4.1)

Describe the design and likely operational parameters (e.g., frequency, duration and amount of biocide addition) of the chlorine injection system for the circulating water system.

Response:

In the event that thermal backwashing or tube cleaning proves unsuccessful, provisions for a chlorine injection system have been incorporated into the design of the circulating water system. The existing service water chlorination system (Section 3.4.2) has been designed with the capability to retrofit a chlorination system to provide sequential, intermittent chlorination downstream of the traveling water screens in each circulating water intake bay. This system would back up the Amertap system to provide condenser slime control. Should a more extensive, continuous chlorination program be required to control hardshell fouling in the intake structure, additional chlorination equipment would be necessary. Chlorination frequency, duration, and concentration are indeterminate at this time since this option is not expected to be added to the circulating water system. However, any chlorination program would be within the EPA Effluent Limitation Guidelines in 40CFR423.

213-cm (84-inch) diameter pipelines. The pipe is copper nickel clad steel inside the pumphouse, reinforced concrete outside the pumphouse, and concrete encased fiberglass inside the turbine building. The 213-cm (84-inch) diameter pipes transition to 244-cm (96-inch) diameter pipes just upstream of the condenser to improve flow characteristics inside the inlet water boxes. The condenser is a single-pass, 46,758 square meter (503,300 square foot), triple shell condenser with 28.6-mm (1-1/8 inch) diameter titanium tubes. Each condenser shell is served by two circulating water pumps. The circulating water discharges from the condenser outlet water boxes through six independent 4.3- by 4.3-meter (50-foot) long 213-cm (84-inch) diameter pipelines. The pipes are copper nickel clad steel followed by concrete encased fiberglass which combine into one 4.3 by 4.3 meter (14 by 14 foot) reinforced concrete tunnel. An additional flow of approximately 114 cubic meters/min (30,000 gpm) from the service water system enters the tunnel immediately downstream of the condenser. This 503-meter (1,650-foot) long tunnel runs to a seal pit structure at the quarry where the circulating water passes over a weir, into the quarry, and finally through a channel into Long Island Sound. The water discharges from the seal pit structure at an average velocity of about 0.76 meters/sec (2.5 fps).

A 152 cm (60-inch) diameter recirculating tempering line is provided from the upstream end of the discharge tunnel to the circulating and service water pumphouse to prevent ice formation around the intake during the winter.

The circulating water pumps are arranged in pairs such that the three pairs of pumps serve the three condenser shells, as shown on Figure 3.4-1. Each pair of pumps is interconnected at the circulating and service water pumphouse by lateral passageways and at the condenser inlet and outlet water boxes by cross connecting 168-cm (66-inch) diameter motor operated valves. This arrangement provides for recirculation of the discharged water for back flushing of the condenser, and for biofouling control of the intake lines, and the pumphouse. An Amertap tube cleaning system is provided for each condenser flow path to maintain a high level of tube cleanliness. This eliminates the need for a chlorine injection system in the circulating water system. However, in the event that thermal backwashing or tube cleaning proves unsuccessful, provisions for a chlorine injection system have been incorporated into the design of the circulating water system. The existing service water chlorination system (Section 3.4.2) has been designed with the capability to retrofit a chlorination system to provide sequential, intermittent chlorination downstream of the traveling water screens in each circulating water intake bay. This system would back up the 2
Amertap system to provide condenser slime control. Should a more extensive, continuous chlorination program be required to control QE291.8
hard shell fouling in the intake structure, additional chlorination equipment would be necessary. Chlorination frequency, duration, and concentration are indeterminate at this time, since this option is not expected to be added to the circulating water system. However, any chlorination program would be within the EPA Effluent Limitation Guidelines in 40 CFR 423.

Six independent condenser tube cleaning systems, one for each condenser flow path, are installed to provide a mechanical means of cleaning the condenser tubes and thus provide for the control of biofouling in the condenser. Each system consists of elastomeric sponge rubber balls oversized in comparison to the condenser tube diameter, which are injected into the circulating water system upstream of the condenser. The balls are forced through the condenser tubes by the differential pressure between condenser inlet

NRC Letter: January 31, 1983

Question No. QE291.9 (Section 3.6.2)

Based on operating experience with Units 1 and 2 or on other bases, provide an estimate of the total residual chlorine concentration of the service water system at the point where it discharges into the discharge tunnel.

Response:

A chlorination study was conducted in 1976 and 1977 for Millstone Units 1 and 2[1], to correlate levels of total residual chlorine at the quarry cut with levels of free available chlorine at the condenser outlets (discharge structures) into the quarry. No significant correlation of the total residual levels at the quarry cut and the free available levels at the discharge structure can be demonstrated from the available data. It is estimated that the total residual chlorine concentration of the Millstone 3 service water system at the point where it discharges into the Unit 3 discharge tunnel will be 2.5 ppm (the worst case). The 2.5 ppm concentration is an extremely conservative estimate. The service water chlorination equipment is designed to chlorinate the service water to a concentration of 2.5 ppm. If it is assumed no chemical reaction occurred between the chlorine and marine organisms (i.e., no chlorine is consumed during chlorination process), the concentration will remain the same at the point where the service water discharges into the discharge tunnel. The service water (30,000 gpm) will mix with the circulating water (912,000 gpm) in the discharge tunnel. Even in this worst case, it is calculated that the total residual chlorine will be reduced to 0.08 ppm at the discharge outfall structure into the quarry.

Reference:

1. NUSCO 1978. Annual Environmental Operating Report, January 1, 1977 - December 31, 1977 Part A, Section 4.7.

A gaseous chlorine solution is injected into the service water system to control biofouling. Chlorination of the service water occurs three times a day for 30-minute periods, for a total of 1 1/2 hours per day. Chlorination is controlled by grab sample monitoring such that the concentration of free available chlorine at the point where the mixture of service water and circulating water is discharged to the quarry is maintained at 0.1 ppm average or less. After mixing with the quarry water, the concentration of free available chlorine is reduced to a concentration below detectable limits (i.e., less than 0.05 ppm). In addition, the chlorine demand of the circulating water will further reduce the free residual chlorine concentration below that which would occur through dilution alone. It is estimated that approximately 3,720 kg/yr (8,200 lb/yr) of chlorine (as Cl_2) will be used for service water chlorination. |2 QE291.9

3.6.3 Floor and Equipment Drainage

Radioactive and potentially radioactive floor drainage is conveyed to the liquid radwaste treatment system (Section 3.5). Nonradioactive floor and equipment drainage, resulting from pump seal leaks, pump seal and bearing water, floor washing, etc, is discharged to the yard storm sewer. Oil contaminated floor drainage is conveyed to oil/water separators before discharge. The oil removed is collected in drums and hauled offsite for recycle or disposal. The amount of floor drainage discharged to the yard storm sewer on a daily basis is variable. There are three oil/water separators, each having a design capacity of 379 liters/min (100 gpm), for the Millstone 3 plant areas. Oil and grease concentrations in the separate effluent are limited to 10 mg/l, average and 20 mg/l, maximum.

3.6.4 Other Liquid Wastes

3.6.4.1 Steam Generator Blowdown

The design of the steam generator blowdown system provides a means of controlling the suspended solids concentration and the chemical composition of the steam generator shell water. The system is capable of blowing down water from each of the four steam generators at various blowdown rates up to a maximum of 341 liters/min (90 gpm) per steam generator. Blowdown from each steam generator is conveyed to the blowdown flash tank in which pressure is maintained at a point slightly above the normal operating pressure of the fourth point feedwater heater shells. Characteristics of steam generator blowdown are presented in Table 3.6-3.

Steam from the flash tank is conveyed to the feedwater heaters. The remaining liquid in the flash tank drains by pressure differential to the condensate side of the condenser. Contaminants are removed from the liquid in the condensate polishing demineralizers, which are located downstream of the condenser. By using the above system, steam generator blowdown will not be discharged to the environment under normal plant operating conditions. During an extended plant outage, the steam generator shells may be drained through the blowdown lines to the condensate polishing system waste

NRC Letter: January 31, 1983

Question No. QE291.10 (Section 3.6.2)

Indicate the time period that the 0.1 ppm FAC average concentration is based upon. Indicate the anticipated maximum concentration value.

Response:

The average free available chlorine concentration of 0.1 ppm or less is based on chlorination, three times a day for 30 minute periods, for a total of 1 1/2 hours per day.

The anticipated maximum concentration of free available chlorine is 0.25 ppm.

NRC Letter: January 31, 1983

Question No. QE291.11 (Section 5.1)

Provide a copy of the references, NUSCO 1981b and Stone & Webster Engineering Corporation, 1976.

Response:

A copy of the following references has been provided under separate cover on April 1, 1983.

NUSCo. 1981b. Feasibility of Modifying the Millstone Units 1 and 2 Cooling Water Intake Screen Wash System to Improve the Return of Fish to Long Island Sound. Submitted to Connecticut DEP.

Stone & Webster Engineering Corporation (SWEC) 1976. Biological Modeling of the Effect of Entrainment on Four Selected Fish Species at the NEP 1 and 2 Site, Charlestown, R.I.

NRC Letter: January 31, 1983

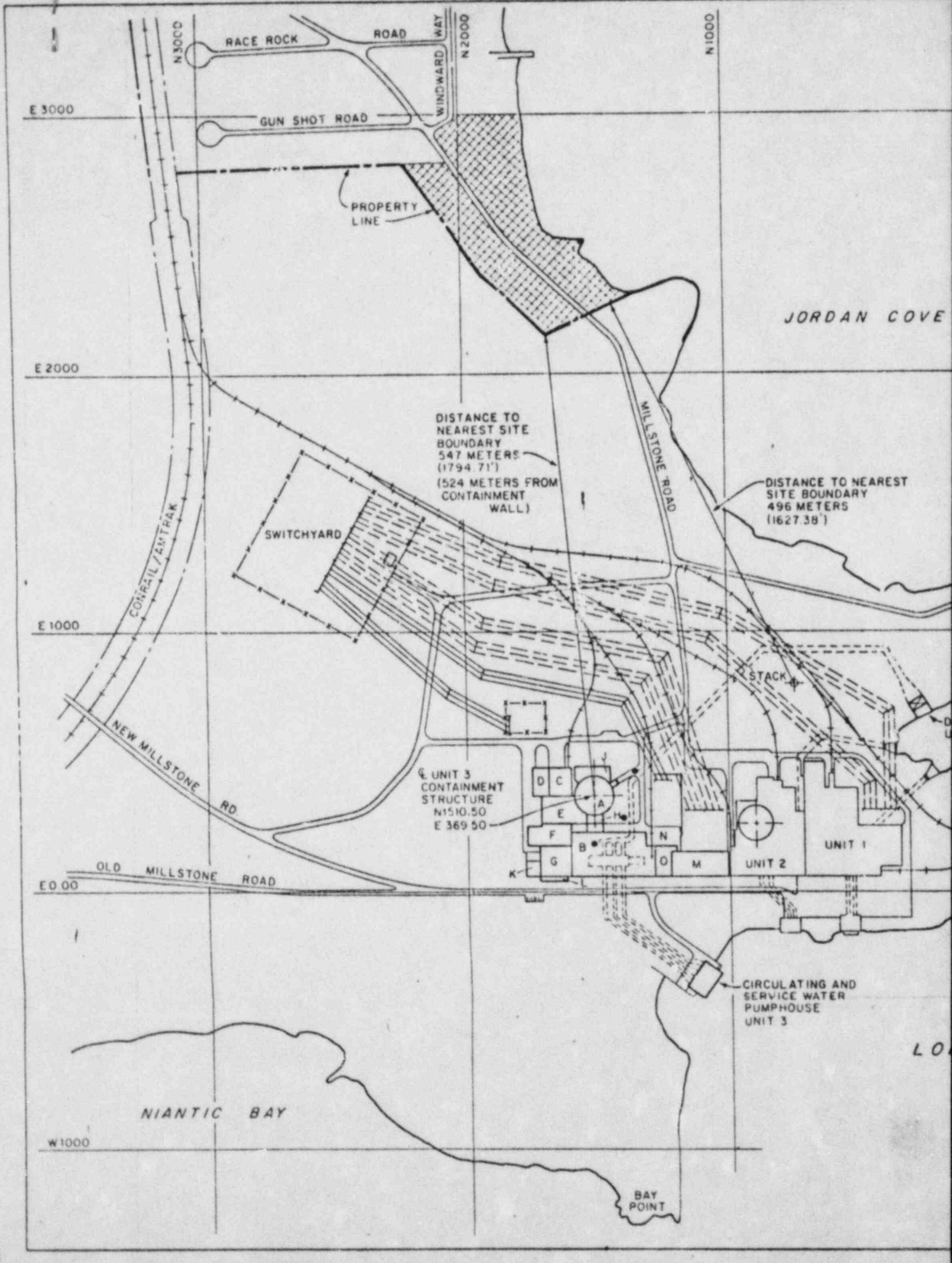
Question No. QE291.12 (Section 5.3)

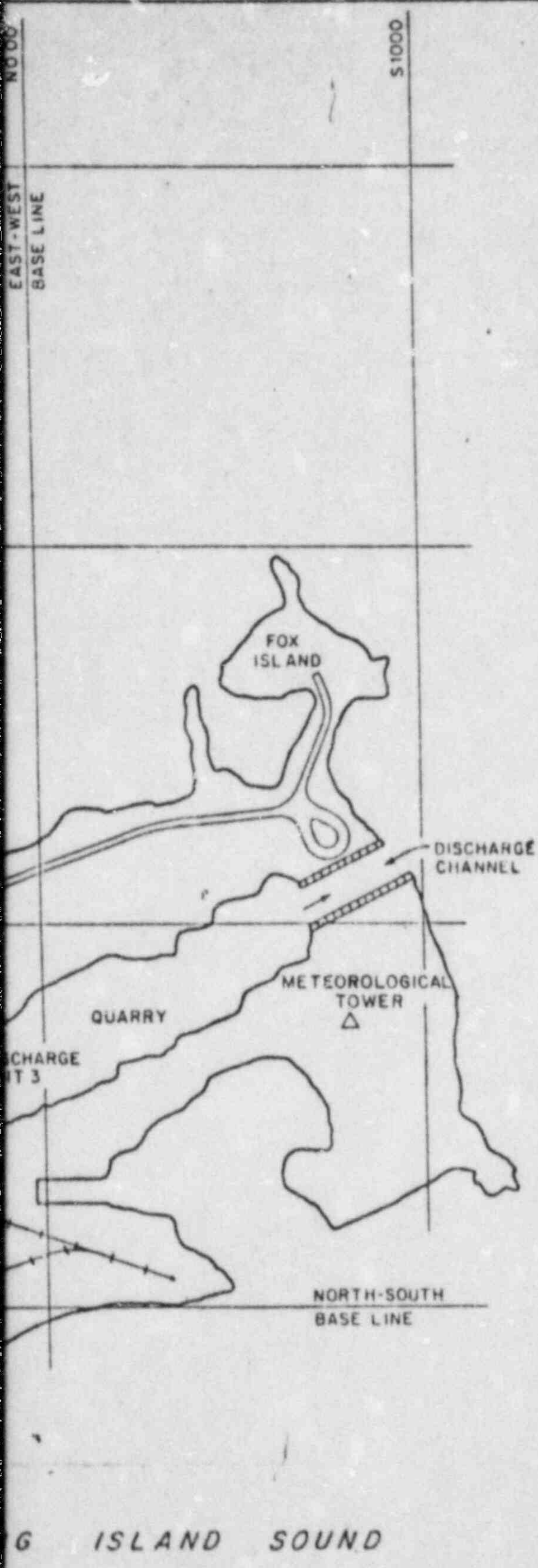
Indicate the location, physically in relation to the Unit 3 discharge structure, where the service water mixes with the circulating water. Section 3.6 indicates that the biocide control point is to be here, but Section 5.3 indicates that the control point is to be where the mixed waters enter the quarry. Indicate which situation is correct.

Response:

Figure QE291.12-1 shows the physical location, in relation to the Millstone 3 discharge structure, where the service water mixes with the circulating water.

The biocide control point will be at the place where the mixed waters enter the quarry as indicated in Section 5.3


RACE ROCK ROAD
WINDWARD WAY
N3000
N2000
N1000
E 3000
GUN SHOT ROAD
PROPERTY LINE
JORDAN COVE
E 2000
MILLSTONE ROAD
DISTANCE TO NEAREST SITE BOUNDARY 547 METERS (1794.71')
(524 METERS FROM CONTAINMENT WALL)
DISTANCE TO NEAREST SITE BOUNDARY 496 METERS (1627.38')
SWITCHYARD
CONRAIL/AMTRAK
E 1000
STACK
NEW MILLSTONE RD.
℄ UNIT 3 CONTAINMENT STRUCTURE N1510.50 E 369.50
A
B
C
D
E
F
G
H
J
K
M
N
O
UNIT 2
UNIT 1
OLD MILLSTONE ROAD
E0.00
CIRCULATING AND SERVICE WATER PUMPHOUSE UNIT 3
NIANTIC BAY
W1000
BAY POINT



EXPLANATICN

A CONTAINMENT STRUCTURE
B TURBINE BUILDING
C FUEL BUILDING
D WASTE DISPOSAL BUILDING
E AUXILIARY BUILDING
F SERVICE BUILDING
G CONTROL BUILDING
H MAIN STEAM VALVE BUILDING
J ENGINEERED SAFETY FEATURES BUILDING
K EMERGENCY DIESEL GENERATOR BUILDING
L OFFICE BUILDING
M WAREHOUSE & UNIT 2 CONDENSATE POLISHING FACILITY
N AUXILIARY BOILER
O CONDENSATE POLISHING ENCLOSURE

LEGEND

PRIVATELY OWNED RECREATION AREA

• POINTS WHERE SERVICE WATER MIXES WITH CIRCULATING WATER

X POINT WHERE MIXED WATERS ENTER THE QUARRY

0 250 500
SCALE-FEET

0 125 250
SCALE-METERS

FIGURE OE 291.12-1
SITE PLAN
MILLSTONE NUCLEAR POWER STATION
UNIT 3
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

NRC Letter: January 31, 1983

Question No. QE291.13 (Section 5.3)

Provide a copy of the reference used for Section 5.3 (i.e., Waslenchuk, 1980).

Response:

A copy of the following reference has been provided under separate cover on April 1, 1983.

Waslenchuk, D.G., 1980. The Concentration, Reactivity, and Fate of Copper, Nickel, and Zinc in a Coastal Power-Station Cooling-Water Plume.

NRC Letter: January 31, 1983

Question No. QE291.14 (Section 12.0)

Provide a copy of the application to Corps of Engineers for construction of discharge outlet change and copy of Corps of Engineers' approval.

Response:

A copy of the above application and approval has been provided under separate cover on April 1, 1983.

NRC Letter: January 31, 1983

Question No. QE291.15 (Section 12.0)

Provide a copy of the (latest) NPDES permit for Millstone 1, 2, and 3 discharges.

Response:

A copy of the above permit has been provided under separate cover on April 1, 1983.

NRC Letter: January 31, 1983

Question No. QE291.16 (Section 12.0)

Provide a copy of the Clean Water Act 316(a) thermal variance application to Conn DEP and a copy of correspondence from Conn DEP documenting approval.

Response:

A copy of the above application and approval has been provided under separate cover on April 1, 1983.

NRC Letter: January 31, 1983

Question No. QE291.17 (Section 12.0)

Provide a copy of the Clean Water Act 316(b) demonstration to Conn DEP and a copy of correspondence from Conn DEP documenting approval of the intake structure and fish return sluiceway.

Response:

A copy of the above correspondence and approval has been provided under separate cover on April 1, 1983.

NRC Letter: January 31, 1983

Question No. QE291.18 (Section Appendix C)

Provide a copy of all reports available from the effluent toxicity testing program. Indicate which "other suitable organisms indigenous to the Millstone area" will be used in effluent toxicity testing.

Response:

No formal report providing summaries and interpretation of data from the effluent toxicity testing program has been prepared to date. Development of the experimental testing facility was not completed until early 1981 when the first series of tests on sheepshead minnow (Cyprinodon variegatus) were begun. During 1982, the program scope was expanded to include Mysidopsis bahia. Preparation of a report on these results is being planned for later in 1983. Copies will be made available.

Other indigenous species currently being considered are the silverside (Menidia menidia), an abundant local shore-zone forage fish, and winter flounder (Pseudopleuronectes americanus) which would be tested as larvae or juveniles. Because the experimental facility would have to be modified appreciably to accommodate the lower water temperature requirements of flounder, tests using this species will probably be delayed for several years.

NRC Letter: January 31, 1983

Question No. QE311.5 (Section 2.1)

It is noted in Tables 2.1-1 through 2.1-20 that the population data has been given in metric measurement. Please provide this data in the English system of miles to correspond with the distances given in Regulatory Guide 1.70, Section 2.1-3 Population Distribution.

Response:

Population per sector, based on distances in English system miles is provided in attached Tables QE311.5-1 through QE311.5-14. Per NRC agreement, distances do not correspond to those given in Regulatory Guide 1.70 but instead are provided in distances suitable for the Probabilistic Safety Study. Transient population distribution is listed by distance and direction in Tables QE311.5-15 through QE311.5-17.

TABLE QE311.5-1

1980 POPULATION DISTRIBUTION
0-10 miles

	Distance (miles)														
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	116	495	119	357	773	91	17	45	317	359	1,697	1,823	6,209
NNE	0	0	31	325	475	806	614	241	288	1,904	1,850	1,295	1,862	3,623	13,314
NE	23	153	57	439	410	191	1,036	2,595	5,649	6,537	5,717	4,020	3,728	2,643	33,198
ENE	6	68	160	210	111	108	514	4,127	1,182	140	7,223	1,364	4,475	4,387	24,075
E	0	16	528	165	212	250	844	1,871	209	76	751	0	621	2,263	7,806
ESE	0	0	73	89	68	11	0	0	0	0	0	0	219	415	875
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	147	157	304
SSW	0	0	0	0	0	0	0	0	0	0	0	0	6	112	118
SW	0	0	0	0	29	132	0	0	0	0	0	0	0	0	161
WSW	0	0	0	0	1,302	179	204	112	0	1,009	1,103	1,510	35	18	5,472
W	0	0	0	257	1,019	383	409	694	23	295	435	187	525	765	4,992
WNW	0	0	0	516	723	504	524	23	40	32	157	52	518	421	3,510
NW	0	0	37	580	364	147	645	297	89	319	573	52	513	392	4,008
NNW	0	122	458	198	438	272	246	232	400	155	455	75	1,076	1,268	5,395
TOTAL	29	359	1,460	3,274	5,270	3,340	5,809	10,283	7,897	10,512	18,581	8,914	15,422	18,287	109,437

TABLE QE311.5-2

1985 POPULATION DISTRIBUTION
0-10 miles

Direction	Distance (miles) 0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	119	507	122	365	791	93	17	46	324	368	1,746	1,886	6,384
NNE	0	0	32	333	487	824	628	246	293	1,916	1,860	1,325	1,907	3,775	13,626
NE	23	156	58	449	420	196	1,060	2,508	5,621	6,505	5,728	4,064	3,790	2,753	33,411
ENE	6	70	164	215	114	110	513	4,107	1,176	142	7,336	1,389	4,500	4,455	24,297
E	0	17	541	169	216	256	854	1,862	208	77	762	0	628	2,282	7,872
ESE	0	0	75	92	69	12	0	0	0	0	0	0	246	466	960
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	165	176	341
SSW	0	0	0	0	0	0	0	0	0	0	0	0	7	126	133
SW	0	0	0	0	30	139	0	0	0	0	0	0	0	0	169
WSW	0	0	0	0	1,367	188	215	118	0	1,082	1,182	1,619	38	18	5,827
W	0	0	0	264	1,062	402	429	729	24	316	466	201	563	791	5,247
WNW	0	0	0	531	748	529	550	24	42	33	166	56	557	453	3,689
NW	0	0	38	597	375	155	677	312	94	342	602	54	542	418	4,206
NNW	0	125	469	203	450	279	253	237	416	165	478	79	1,119	1,320	5,593
TOTAL	29	368	1,496	3,360	5,460	3,455	5,970	10,316	7,891	10,624	18,904	9,155	15,808	18,919	111,755

TABLE QE311.5-3

1990 POPULATION DISTRIBUTION
0-10 miles

	Distance (miles)														
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	121	515	124	371	804	94	18	47	330	374	1,792	1,956	6,546
NNE	0	0	32	338	494	837	637	249	298	1,948	1,892	1,346	1,943	3,980	13,994
NE	24	159	59	456	427	199	1,075	2,630	5,712	6,610	5,799	4,093	3,835	2,880	33,958
ENE	6	71	166	218	116	112	521	4,173	1,195	142	7,368	1,390	4,558	4,472	24,508
E	0	17	549	172	220	260	869	1,892	211	77	765	0	632	2,297	7,961
ESE	0	0	76	93	70	12	0	0	0	0	0	0	274	518	1,043
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	184	196	380
SSW	0	0	0	0	0	0	0	0	0	0	0	0	8	140	148
SW	0	0	0	0	31	142	0	0	0	0	0	0	0	0	173
WSW	0	0	0	0	1,395	191	219	120	0	1,173	1,282	1,755	41	18	6,194
W	0	0	0	275	1,092	411	438	744	25	341	505	217	611	814	5,473
WNW	0	0	0	553	775	540	562	25	43	34	170	61	603	488	3,854
NW	0	0	40	621	390	158	691	318	96	371	614	56	560	439	4,354
NNW	0	127	477	209	468	286	257	241	424	176	488	84	1,146	1,358	5,741
TOTAL	30	374	1,520	3,450	5,602	3,519	6,073	10,486	8,022	10,919	19,213	9,376	16,187	19,556	114,327

TABLE QE311.5-4

2000 POPULATION DISTRIBUTION
0-10 MILES

	Distance (miles)														
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	118	505	121	364	788	92	17	46	323	367	1,857	2,125	6,723
NNE	0	0	32	332	485	821	625	245	292	1,909	1,854	1,320	1,928	4,262	14,105
NE	23	156	58	440	419	195	1,055	2,576	5,593	6,473	5,766	4,183	3,948	3,068	33,961
ENE	6	69	163	214	113	110	510	4,086	1,170	146	7,536	1,422	4,661	4,574	24,780
E	0	17	539	169	216	255	852	1,853	207	79	783	0	646	2,349	7,965
ESE	0	0	75	91	69	12	0	0	0	0	0	0	309	584	1,140
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	207	220	427
SSW	0	0	0	0	0	0	0	0	0	0	0	0	9	158	167
SW	0	0	0	0	32	146	0	0	0	0	0	0	0	0	178
WSW	0	0	0	0	1,436	197	226	124	0	1,320	1,443	1,975	46	18	6,785
W	0	0	0	283	1,124	423	451	766	25	384	568	245	687	849	5,805
WNW	0	0	0	569	798	556	578	25	44	35	178	68	677	546	4,074
NW	0	0	41	640	402	163	712	327	99	417	632	57	589	476	4,555
NNW	0	124	468	208	482	287	253	237	427	193	502	91	1,187	1,437	5,896
TOTAL	29	366	1,494	3,459	5,697	3,529	6,050	10,331	7,874	11,002	19,565	9,728	16,751	20,666	116,561

TABLE QE311.5-5

2010 POPULATION DISTRIBUTION
0-10 miles

Direction	Distance (miles) 0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	115	491	118	353	765	90	17	45	314	356	1,922	2,316	6,902
NNE	0	0	31	322	471	796	605	240	292	2,025	1,970	1,281	1,902	4,876	14,831
NE	22	151	56	435	407	189	1,021	2,858	6,282	7,270	6,136	4,163	3,954	3,379	36,323
ENE	6	67	158	208	110	107	567	4,589	1,314	139	7,200	1,308	4,954	4,359	25,096
E	0	16	523	163	210	247	876	2,081	232	76	748	0	649	2,366	8,187
ESE	0	0	72	88	67	11	0	0	0	0	0	0	359	679	1,276
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	241	256	407
SSW	0	0	0	0	0	0	0	0	0	0	0	0	11	183	194
SW	0	0	0	0	28	131	0	0	0	0	0	0	0	0	159
WSW	0	0	0	0	1,285	176	202	111	0	1,586	1,733	2,372	55	17	7,537
W	0	0	0	319	1,099	378	404	685	23	456	683	294	825	875	6,041
WNW	0	0	0	641	842	497	517	23	40	31	166	82	809	631	4,279
NW	0	0	46	721	446	149	637	293	88	501	566	51	559	485	4,542
NNW	0	121	454	216	543	294	244	230	395	224	449	103	1,141	1,429	5,343
TOTAL	28	355	1,455	3,604	5,626	3,328	5,838	11,200	8,683	12,353	19,965	10,010	17,381	21,871	121,697

TABLE QE311.5-6

2020 POPULATION DISTRIBUTION
0-10 miles

Direction	Distance (miles) 0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	108	461	111	332	719	84	16	42	295	335	1,976	2,535	7,014
NNE	0	0	29	303	443	745	566	229	286	2,204	2,148	1,203	1,830	5,713	15,699
NE	21	142	53	408	382	177	955	3,307	7,382	8,543	6,702	4,078	3,896	3,746	39,792
ENE	5	63	148	195	104	100	657	5,393	1,545	127	6,568	1,108	5,370	3,961	25,344
E	0	15	490	153	196	232	910	2,445	273	69	682	0	646	2,367	8,478
ESE	0	0	68	83	63	11	0	0	0	0	0	0	411	778	1,414
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	276	294	570
SSW	0	0	0	0	0	0	0	0	0	0	0	0	12	210	222
SW	0	0	0	0	22	101	0	0	0	0	0	0	0	0	123
WSW	0	0	0	0	997	137	157	86	0	1,916	2,094	2,866	67	13	8,333
W	0	0	0	367	1,024	293	313	531	18	547	825	355	997	892	6,162
WNW	0	0	0	738	887	386	401	18	31	24	143	99	971	728	4,426
NW	0	0	53	830	504	122	494	227	68	606	439	40	488	472	4,343
NNW	0	113	427	224	625	300	226	216	334	262	349	118	1,034	1,365	5,593
TOTAL	26	333	1,376	3,762	5,358	2,936	5,398	12,536	9,953	14,340	20,245	10,202	17,974	23,074	127,513

TABLE QE311.5-7

2030 POPULATION DISTRIBUTION
0-10 miles

	Distance (miles)														
Direction	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-6.0	6.0-7.0	7.0-8.5	8.5-10.0	Total
N	0	0	98	418	100	301	652	76	14	38	268	304	2,020	2,781	7,070
NNE	0	0	26	275	401	674	509	214	200	2,458	2,403	1,090	1,722	6,738	16,790
NE	19	129	48	370	347	160	860	3,952	8,961	10,369	7,500	3,926	3,772	4,180	44,593
ENE	5	57	134	176	94	90	785	6,547	1,875	108	5,609	812	5,920	3,357	25,569
E	0	14	443	139	178	211	957	2,968	332	59	583	0	635	2,352	8,871
ESE	0	0	61	75	57	10	0	0	0	0	0	0	467	883	1,553
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0	0	313	334	647
SSW	0	0	0	0	0	0	0	0	0	0	0	0	14	238	252
SW	0	0	0	0	12	57	0	0	0	0	0	0	0	0	69
WSW	0	0	0	0	563	77	88	49	0	2,320	2,535	3,471	81	9	9,193
W	0	0	0	431	898	166	177	300	10	656	999	430	1,208	899	6,174
WNW	0	0	0	866	937	218	227	10	17	14	104	120	1,168	842	4,523
NW	0	0	62	974	579	80	279	128	39	733	248	22	371	437	3,952
NNW	0	103	387	233	731	307	201	196	242	307	197	136	859	1,238	5,137
TOTAL	24	303	1,259	3,957	4,897	2,351	4,735	14,440	11,770	17,062	20,446	10,311	18,550	24,288	134,393

TABLE QE311.5-8

1980 POPULATION DISTRIBUTION
0-50 Miles

Direction	Distance (miles) 0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6,209	3,585	3,801	4,215	2,394	3,819	22,992	15,034	16,350	9,419	9,416	97,234
NNE	13,314	3,449	7,323	12,988	13,256	16,095	7,918	10,604	14,725	14,430	15,579	129,681
NE	33,198	3,521	4,375	2,492	1,747	4,426	3,753	9,235	16,357	49,825	108,963	237,892
ENE	24,075	8,129	3,167	4,347	6,244	8,012	6,631	11,430	21,117	44,766	74,863	212,781
E	7,806	613	1,898	2,910	4,632	7,414	1,652	4,908	6,921	1,135	1,592	41,481
ESE	875	125	0	0	0	0	0	615	0	0	0	1,615
SE	0	0	0	0	154	889	0	0	0	0	0	1,043
SSE	0	0	119	125	395	1,676	0	0	0	0	0	2,315
S	304	0	292	226	2,128	6,674	262	0	0	0	0	9,886
SSW	118	721	138	581	1,826	6,602	8,465	8,756	518	0	0	27,725
SW	161	0	472	3,149	1,681	5,897	8,206	13,479	22,557	56,346	82,581	194,529
WSW	5,472	335	0	0	0	0	0	0	310	10,562	45,270	61,949
W	4,992	6,324	5,059	3,739	8,566	10,729	14,997	32,277	120,757	142,931	109,444	459,815
WNW	3,510	2,782	3,485	4,311	2,175	6,066	18,007	39,549	94,929	65,705	138,140	378,659
NW	4,008	673	979	1,287	1,810	6,905	21,332	39,104	118,751	280,731	98,897	574,477
NNW	5,395	1,104	1,089	1,389	2,881	6,688	9,153	16,221	82,445	56,488	46,438	229,291
TOTAL	109,437	31,361	32,197	41,759	49,889	91,892	123,368	201,212	515,737	732,338	731,183	2,660,373

TABLE QE311.5-9

1985 POPULATION DISTRIBUTION
0-50 Miles

Direction	Distance (miles) 0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6,384	3,709	3,912	4,284	2,455	3,975	23,609	15,179	16,554	10,076	9,761	99,880
NNE	13,626	3,588	7,467	13,148	13,437	16,421	8,508	11,390	15,482	15,051	15,949	134,067
NE	33,411	3,719	4,624	2,596	1,782	4,584	4,114	9,967	17,574	51,922	113,306	247,599
ENE	24,297	8,299	3,323	4,506	6,464	8,543	7,069	12,579	24,369	50,548	84,619	234,616
E	7,872	650	1,960	3,029	4,847	7,741	1,695	5,414	7,094	1,389	1,947	43,638
ESE	960	141	0	0	0	0	0	633	0	0	0	1,734
SE	0	0	0	0	173	998	0	0	0	0	0	1,171
SSE	0	0	134	141	444	1,881	0	0	0	0	0	2,600
S	341	0	328	254	2,388	7,492	294	0	0	0	0	11,097
SSW	133	809	155	652	2,051	7,409	9,502	9,828	582	0	0	31,121
SW	169	0	530	3,534	1,886	6,619	9,210	15,131	25,316	63,246	92,960	218,331
WSW	5,827	343	0	0	0	0	0	0	347	11,855	50,813	69,185
W	5,247	6,469	5,248	3,892	8,923	11,397	15,836	32,973	120,630	143,186	111,192	464,993
WNW	3,689	2,858	3,573	4,490	2,340	6,618	19,025	40,663	96,648	68,167	139,867	387,938
NW	4,206	722	1,034	1,350	1,911	7,399	22,266	41,515	122,277	280,976	101,434	585,090
NNW	5,593	1,191	1,188	1,500	3,103	7,290	10,122	17,299	84,245	50,860	48,278	238,669
TOTAL	111,755	32,498	33,476	43,376	52,204	98,349	131,250	212,571	531,118	755,276	769,856	2,771,729

TABLE QE311.5-10

1990 POPULATION DISTRIBUTION
0-50 Miles

Direction	Distance (miles) 0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	6,546	3,845	4,038	4,382	2,508	4,038	24,188	15,349	16,732	10,618	10,086	102,330
NNE	13,994	3,756	7,704	13,492	13,794	16,796	9,080	12,173	16,285	15,654	16,533	139,261
NE	33,958	3,995	4,972	2,701	1,783	4,660	4,432	10,726	18,817	53,874	115,930	255,048
ENE	24,508	8,470	3,404	4,611	6,662	8,984	7,708	13,903	25,912	52,328	85,862	242,352
E	7,961	678	2,036	3,125	5,030	8,086	1,877	5,978	7,758	1,486	2,084	46,099
ESE	1,043	157	0	0	0	0	0	694	0	0	0	1,894
SE	0	0	0	0	192	1,111	0	0	0	0	0	1,304
SSE	0	0	149	157	494	2,093	0	0	0	0	0	2,893
S	380	0	364	283	2,659	8,339	327	0	0	0	0	12,352
SSW	148	901	172	726	2,283	8,245	10,576	10,938	647	0	0	34,636
SW	173	0	590	3,934	2,100	7,367	10,251	16,841	28,181	70,395	103,171	243,003
WSW	6,194	348	0	0	0	0	0	0	387	13,194	56,556	76,769
W	5,473	6,556	5,355	3,992	9,230	11,849	16,307	33,430	121,678	144,531	113,678	472,079
WNW	3,854	2,933	3,649	4,691	2,480	7,057	19,843	41,837	98,320	70,148	141,846	396,653
NW	4,354	767	1,079	1,401	1,995	7,804	23,217	43,848	126,477	285,359	104,083	600,384
NNW	5,741	1,260	1,266	1,583	3,301	7,820	10,959	18,379	86,327	61,444	50,254	248,334
TOTAL	114,327	33,666	34,778	45,078	54,512	104,249	138,765	224,096	547,521	779,031	800,083	2,876,106

TABLE QE311.5-11

2000 POPULATION DISTRIBUTION
0-50 Miles

Direction	0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
	Distance (miles)											
N	6,723	4,184	4,312	4,476	2,556	4,177	25,449	16,407	18,056	11,685	10,729	108,754
NNE	14,105	4,075	8,098	14,028	14,355	17,458	9,619	13,057	17,607	16,591	17,446	146,439
NE	33,961	4,355	5,424	2,857	1,807	4,890	4,812	11,529	20,117	56,342	119,918	266,012
ENE	24,780	8,796	3,614	4,849	6,955	9,539	8,541	14,918	28,267	56,030	89,983	256,272
E	7,965	728	2,153	3,301	5,282	8,534	2,072	6,297	8,710	1,556	2,181	48,779
ESE	1,140	176	0	0	0	0	0	694	0	0	0	2,010
SE	0	0	0	0	217	1,251	0	0	0	0	0	1,460
SSE	0	0	168	176	557	2,358	0	0	0	0	0	3,259
S	427	0	411	320	2,995	9,394	369	0	0	0	0	13,916
SSW	167	1,015	194	819	2,571	9,292	11,912	12,324	729	0	0	3,9023
SW	178	0	665	4,431	2,366	8,300	11,550	18,971	31,743	79,298	116,215	273,717
WSW	6,785	352	0	0	0	0	0	0	436	14,864	63,711	86,148
W	5,805	6,644	5,606	4,254	9,787	12,951	17,866	34,599	125,287	148,472	117,327	488,598
WNW	4,074	3,030	3,761	4,971	2,728	7,948	21,572	43,721	101,040	73,366	145,793	412,004
NW	4,555	849	1,175	1,516	2,168	8,606	25,062	48,039	132,619	289,048	107,362	620,999
NNW	5,896	1,441	1,479	1,759	3,516	8,422	12,290	20,245	90,375	66,280	52,843	264,546
TOTAL	116,561	35,645	37,060	47,757	57,860	113,120	151,114	240,801	574,986	813,532	843,508	3,031,944

TABLE QE311.5-12

2010 POPULATION DISTRIBUTION
0-50 Miles

Direction	0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
	Distance (miles)											
N	6,902	4,557	4,665	4,774	2,577	4,003	26,468	17,255	18,778	12,147	11,256	113,382
NNE	14,831	4,564	9,030	15,654	15,978	18,456	10,363	14,292	19,329	17,342	17,925	157,764
NE	36,323	5,239	6,538	3,043	1,643	4,635	5,007	12,045	20,753	57,555	122,483	275,264
ENE	25,086	9,143	3,391	4,770	7,216	9,792	9,069	15,515	29,458	57,961	92,581	263,982
E	8,187	745	2,370	3,364	5,488	8,882	2,201	6,493	9,270	1,602	2,247	50,849
ESE	1,276	205	0	0	0	0	0	694	0	0	0	2,175
SE	0	0	0	0	253	1,456	0	0	0	0	0	1,709
SSE	0	0	196	205	648	2,744	0	0	0	0	0	3,793
S	497	0	477	372	3,485	10,932	429	0	0	0	0	16,192
SSW	194	1,181	226	951	2,992	10,813	13,863	14,341	848	0	0	45,409
SW	159	0	773	5,156	2,753	9,657	13,441	22,077	36,939	92,276	135,240	318,471
WSW	7,537	341	0	0	0	0	0	0	507	17,296	74,138	99,819
W	6,041	6,458	5,395	4,190	10,097	12,769	17,085	34,344	134,595	157,780	125,673	514,427
WNW	4,279	3,151	3,822	5,442	2,848	8,225	22,179	46,163	103,799	74,321	151,134	425,363
NW	4,542	905	1,218	1,553	2,230	8,938	27,032	52,038	143,630	318,041	112,252	672,379
NNW	5,343	1,493	1,551	1,767	3,700	8,857	13,049	22,278	96,017	72,458	56,864	283,877
TOTAL	121,697	37,982	39,652	51,241	61,908	120,159	160,186	257,535	613,923	878,779	901,793	3,244,855

TABLE QE311.5-13

2020 POPULATION DISTRIBUTION
0-50 MILES

Direction	Distance (Miles) 0.0-10	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	7,014	4,983	5,080	5,168	2,555	3,607	27,369	18,277	19,522	12,194	1.,703	117,472
NNE	15,699	5,159	10,292	17,946	18,246	19,640	10,980	15,459	21,209	18,000	18,318	170,948
NE	39,792	6,392	7,991	3,223	1,359	4,088	5,102	12,505	21,223	58,461	124,853	284,989
ENE	25,344	9,495	2,900	4,492	7,465	9,994	9,680	16,036	30,448	59,698	94,562	270,114
E	8,478	737	2,641	3,362	5,693	9,258	2,388	6,665	9,774	1,645	2,307	52,948
ESE	1,414	235	0	0	0	0	0	694	0	0	0	2,345
SE	0	0	0	0	289	1,667	0	0	0	0	0	1,956
SSE	0	0	224	235	741	3,142	0	0	0	0	0	4,342
S	570	1	547	425	3,989	12,513	490	0	0	0	0	18,535
SSW	222	1,352	259	1,089	3,426	12,379	15,870	16,418	971	0	0	51,986
SW	1,230	0	885	5,903	3,153	11,057	15,386	25,274	42,288	105,642	154,826	364,537
WSW	8,333	317	0	0	0	0	0	0	580	19,884	84,877	113,911
W	6,162	6,048	4,891	3,931	10,210	11,812	15,000	33,235	148,397	171,245	136,597	547,528
WNW	4,426	3,262	3,828	5,987	2,870	8,095	22,071	48,774	106,318	73,421	157,403	436,455
NW	4,343	940	1,228	1,546	2,223	8,948	29,062	55,697	156,933	361,845	117,347	740,112
NNW	5,593	1,478	1,551	1,665	3,703	8,978	13,273	24,299	102,680	79,395	61,252	303,947
TOTAL	127,513	40,399	42,317	54,972	66,002	125,178	166,671	273,333	660,343	961,350	964,045	3,482,123

TABLE QE311.5-14

2030 POPULATION DISTRIBUTION
0-50 Miles

	Distance (miles)											
Direction	0.0-10.0	10.0-12.5	12.5-15.0	15.0-17.5	17.5-20.0	20.0-25.0	25.0-30.0	30.0-35.0	35.0-40.0	40.0-45.0	45.0-50.0	Total
N	7,070	5,457	5,560	5,673	2,494	2,977	28,129	19,404	20,179	11,791	12,064	120,798
NNE	16,790	5,867	11,916	20,976	21,229	21,035	11,529	16,632	23,285	18,545	18,513	186,317
NE	44,593	7,855	9,835	3,405	946	3,215	5,069	12,652	21,064	58,371	125,139	292,144
ENE	25,569	9,857	2,112	3,993	7,608	9,742	9,676	16,009	30,476	59,578	96,651	269,271
E	8,871	702	2,972	3,260	5,758	9,358	2,450	6,690	9,831	1,649	2,312	53,853
ESE	1,553	267	0	0	0	0	0	694	0	0	0	2,514
SE	0	0	0	0	329	1,894	0	0	0	0	0	2,223
SSE	0	0	254	267	842	3,569	0	0	0	0	0	4,932
S	647	1	621	483	4,532	14,216	557	0	0	0	0	21,057
SSW	252	1,536	294	1,238	3,890	14,061	18,029	18,652	1,102	0	0	59,054
SW	69	0	1,006	6,706	3,581	12,560	17,480	28,713	48,039	120,011	175,888	414,053
WSW	9,193	281	0	0	0	0	0	0	659	22,495	96,421	129,049
W	6,174	5,405	4,066	3,456	10,126	9,995	11,436	31,179	166,893	189,069	150,460	588,259
WNW	4,523	3,373	3,782	6,626	2,789	7,522	21,182	51,621	108,640	70,611	164,681	445,350
NW	3,952	954	1,201	1,487	2,138	8,612	31,162	59,042	172,953	422,210	122,882	826,593
NNW	5,137	1,385	1,469	1,446	3,776	8,814	12,963	26,333	110,463	87,186	66,181	325,153
TOTAL	134,393	42,940	45,089	59,016	70,038	127,570	169,662	287,621	713,584	1,061,516	1,029,192	3,740,620

TABLE QE311.5-15

TRANSIENT POPULATION - PARK VISITATION

Park Name	Sector Location (miles)	Annual Attendance	Average Daily Attendance
Ocean Beach	3.4 E	500,000	1,370[1]
Fort Griswold	5.5 ENE	47,220	129[2]
Harkness Memorial	2.8 E	146,745	868[2]
Rocky Neck	3.8 W	533,312	1,630[3]
Submarine Memorial	5.6 NE	68,000	186[1]
Waterford Beach	2.8 ESE	53,000	663[4]

NOTES:

[1] Year-round use - average daily attendance based on 365 days/year

[2] Seasonal attendance between April 15 to September 30

[3] Seasonal camping with other year-round use - 53,275 campers from April 15 to September 30. 480,037 other visitors over 365 days.

[4] Based on attendance from mid-June through Labor Day.

Sources: State of Connecticut, Department of Environmental Protection, Parks and Recreation Unit, State Park Attendance, 1978.

American Automobile Association, Campbook, Northeastern Region, 1981 Edition.

Telecon: Ellis, C.S. (SWEC) 1981p to Bugbee, R. (Parks & Recreation Dept.), Waterford, Conn., December 21, 1981.

Ellis, C.S. (SWEC) 1981j to Submarine Memorial Assoc., Groton, Conn, Nov. 20, 1981.

Ellis, C.S. (SWEC) 1981h to Butler, Mrs., Ocean Beach Park, New London, Conn., November 19, 1981.

TABLE QE311.5-16

TRANSIENT POPULATION
EMPLOYMENT 1977

Direction	Distance (Miles) 0.0-1.0	1.0-2.0	2.0-3.0	3.0-4.0	4.0-5.0	Total
N	-	-	-	-	-	-
NNE	-	-	-	-	-	-
NE	-	-	315	-	655	970
ENE	-	-	-	-	2767	2767
E	-	-	-	-	-	-
ESE	-	-	-	-	-	-
SE	-	-	-	-	-	-
SSE	-	-	-	-	-	-
S	-	-	-	-	-	-
SSW	-	-	-	-	-	-
SW	-	-	-	-	-	-
WSW	-	-	-	-	-	-
W	-	-	-	-	-	-
WNW	-	-	-	-	-	-
NW	-	200	-	115	-	315
NNW	-	-	-	-	-	-
Total	-	200	315	115	3422	4052

NOTE:

Firms with 50 or more employees

Source: 1977 Facts Book Southeastern Connecticut Chamber of Commerce, New London, Conn.

TABLE QE311.5-17

TRANSIENT POPULATION
1981-1982 SCHOOL ENROLLMENTS

	0.0-1.0	1.0-2.0	2.0-3.0	3.0-4.0	4.0-5.0	Total
N	0	384	0	271	0	655
NNE	0	0	0	0	369	369
NE	0	0	1073	755	2,790	4,613
ENE	0	293	0	800	0	1,093
E	0	0	0	0	0	0
ESE	0	0	0	0	0	0
SE	0	0	0	0	0	0
SSE	0	0	0	0	0	0
S	0	0	0	0	0	0
SSW	0	0	0	0	0	0
SW	0	0	0	0	0	0
WSW	0	0	0	0	0	0
W	0	0	0	0	0	0
WNW	0	0	381	0	0	381
NW	0	0	785	372	0	1,157
NNW	0	0	0	0	1,716	1,716
Total	0	677	2,239	2,198	4,875	9,989

Source: Connecticut Education Directory, 1981-82. Conn. State Board of Education, Hartford, Conn.

NRC Letter: January 31, 1983

Question No. QE320.1 (Sections 8 and 11)

Provide the following:

A production cost analysis which shows the difference in system production costs associated with the availability vs. unavailability of the proposed nuclear addition. Note, the resulting cost differential should be limited solely to the variable or incremental costs associated with generating electricity from the proposed nuclear addition and the sources of replacement energy. If, in your analysis, other factors influence the cost differential, explain in detail.

a. The analysis should provide results on an annual basis covering the period from initial operation of the first unit through five full years of operation of the last unit.

b. Where more than one utility shares ownership in the proposed nuclear addition or where the proposed facility is centrally dispatched as part of an interconnected pool, the results of the anlaysis may be aggregated for all participating systems.

c. The analysis should assume electrical energy requirements grow at (1) the system's latest official forecasted growth rate, and (2) zero growth from the latest actual annual energy requirement.

d. The analysis should assume two capacity factors for the nuclear facility (1) 50 percent average annual capacity factor and (2) applicant's currently anticipated average annual capacity factor.

e. For each year (and for each growth rate scenario) the following results should be clearly stated: (1) system present worth production costs with the proposed nuclear addition available as scheduled; (2) system present worth production costs without the proposed nuclear addition available; (3) the capacity factor assumed for the nuclear addition (4) the average fuel cost and variable O&M cost for the nuclear addition and the sources of replacement energy (by fuel type)--both expressed in mills per kWh; and (5) the proportion of replacement energy assumed to be provided by coal, oil, gas, etc. for both purchased power and for self-generated power. The base year for all costs should be identified.

Response:

On March 26, 1982, the Nucler Regulatory Commission ammended its rules and regulations to provide that an applicant for a license to operate a nuclear power plant need not include in its Environmental Report - Operating License Stage any discussion of "need for power or alternative

energy sources. . ." 47FR12940 1982. In his letter to W.G. Counsil, dated January 31, 1983, to which the "Requests for Additional Information" were attached, Darrell G. Eisenhut, Director, Divison of Licensing, Office of Nuclear Reactor Regulation, indicated that the Staff will not include need for power and alternative energy sources issues in the licensing review for Millstone 3. Accordingly, because this question is in essence an inquiry into the need for power and alternative energy sources, a response is not necessary.

NRC Letter: January 31, 1983

Question No. QE320.2 (Sections 8 and 11)

Provide 30 and 40 yr present worth fuel and O&M costs (for the nuclear units(s)). Provide values for all variables assumed in calculating these costs (escalation, discount rates, etc).

Response:

See response to QE320.1.

NRC Letter: January 31, 1983

Question No. Q470.1 (Section 2.1)

Provide the following distributional data for each of the 22-1/2 degree radial sectors centered on the sixteen cardinal compass directions for the radial distances of 1.6, 3.2, 4.8, 6.4, 8.0, 16.1, 32.2, 48.3, 64.4, and 80.4 km for the reactor:

1. Present annual meat production (kg/yr)

2. Present annual milk production (l/yr)

3. Present annual vegetable production (kg/yr)

Response.

Estimated production of beef, milk, and vegetables (corn) is listed by distance and direction in Tables Q470.1-1, Q470.1-2, and Q470.1-3. Corn was discussed in lieu of all vegetables because garden vegetables were not given in quantities and corn was uniformly distributed throughout the three states. Since data was only available as state totals in Connecticut and Rhode Island and as county totals in N.Y., distributions, with the exception of those within 8 km, were calculated by assuming that production was evenly distributed over either the state or the county land area. Production within 8 km was distributed by first sector distance (see Table 2.1-22 of the EROLS), and actual survey data of beef, dairy, and vegetable gardens was used.

References:

1. Connecticut Department of Agriculture 1981. Connecticut Agriculture Statistics, 1980. Hartford, Conn. November 1981.

2. Rhode Island Department of Environmental Management, Division of Agriculture. Rhode Island Agriculture 1980. Providence, R.I.

3. State of New York, Department of Agriculture and Markets. New York Agricultural Statistics.

TABLE Q470.1-1

ESTIMATED 1980 BEEF PRODUCTION (10^3 kgs)
WITHIN 80.4 km OF MILLSTONE 3

	Distance (km)									
Direction	0-1.6	1.6-3.2	3.2-4.8	4.8-6.4	6.4-8.0	8.0-16.1	16.1-32.2	32.2-48.3	48.3-64.4	64.4-80.4
N	0	0	0	0	0	36.0	145.0	243.0	334.0	438.0
NNE	0	0	0	0	0	32.0	135.0	239.0	332.0	326.0
NE	0	0	0	0	0	32.0	145.0	221.0	52.0	0
ENE	0	0	0	0	0	35.0	131.0	34.0	0	0
E	0	0	0	0	0	6.0	25.0	0	0	0
ESE	0	0	0	0	0	0	0	0	0	0
SE	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0
SW	0	0	0	0	0	0	0	0	0	0
WSW	0	0	0	0	0	5.0	2.0	0	0	0
W	0	0	0	0	0	28.0	214.0	174.0	210.0	321.0
WNW	0	0	0	0	0	35.0	131.0	242.0	335.0	439.0
NW	0	0	0	0	0	35.0	141.0	227.0	328.0	434.0
NNW	0	0	0	0	0	36.0	146.0	243.0	335.0	428.0

TABLE Q470.1-2

ESTIMATED 1980
MILK PRODUCTION (10^3 liters) 80.4 km OF MILLSTONE 3

Direction	Distance (km) 0-1.6	1.6-3.2	3.2-4.8	4.8-6.4	6.4-8.0	8.0-16.1	16.1-32.2	32.2-48.3	48.3-64.4	64.4-80.4
N	0	0	0	0	0	817.0	3243.0	5445.0	7492.0	9819.0
NNE	0	0	0	0	0	709.0	3049.0	5356.0	7484.0	8212.0
NE	0	0	0	0	0	723.0	3251.0	5707.0	3404.0	3263.0
ENE	0	0	0	0	0	775.0	3035.0	2422.0	2636.0	1895.0
E	0	0	0	0	0	141.0	724.0	554.0	440.0	14.0
ESE	0	0	0	0	0	0	0	0	193.42	0
SE	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0
SW	0	0	0	0	0	0	0	0	0	0
WSW	0	0	0	0	0	109.0	48.0	0	0	0
W	0	0	0	0	0	632.0	2679.0	3911.0	4722.0	7191.0
WNW	0	0	0	0	97.0	785.0	2932.0	5428.0	7516.0	9847.0
NW	0	0	0	0	0	779.0	3151.0	5083.0	7354.0	9728.0
NNW	0	0	0	0	0	818.0	3265.0	5445.0	7514.0	9590.0

TABLE Q470.1-3

ESTIMATED 1980 CORN PRODUCTION (kg)
WITHIN 80.4 km OF MILLSTONE 3

Direction	Distance (km) 0-1.6	1.6-3.2	3.2-4.8	4.8-6.4	6.4-8.0	8.0-16.1	16.1-32.2	32.2-48.3	48.3-64.4	64.4-80.4
N	50	2,030	3,700	4,780	6,220	51,680	205,070	344,250	473,620	620,720
NNE	0	1,810	3,320	4,740	6,030	44,820	191,900	338,590	544,660	2,823,980
NE	490	1,790	3,440	4,830	5,630	45,690	205,590	705,470	6,024,190	8,640,230
ENE	210	2,000	3,460	4,830	1,720	48,990	457,500	4,441,010	6,975,080	5,021,020
E	50	2,070	2,660	2,850	310	8,930	491,570	1,465,730	1,163,120	35,810
ESE	0	990	240	0	0	2,900	580	0	511,800	0
SE	0	0	0	0	0	0	0	4,260	0	0
SSE	0	0	0	0	0	0	4,030	7,810	0	0
S	0	0	0	0	0	1,390	11,460	28,320	0	0
SSW	0	0	0	0	0	540	15,970	57,840	18,960	0
SW	0	0	520	0	0	0	9,140	31,770	99,330	117,490
WSW	0	0	2,470	330	920	6,900	3,020	0	510	40,310
W	0	160	3,060	4,690	4,170	39,980	169,340	247,240	298,530	454,620
WNW	0	730	3,420	4,570	6,150	49,650	185,340	343,130	475,180	622,550
NW	140	1,180	3,040	4,670	6,220	49,280	199,210	321,300	464,970	615,120
NNW	330	750	1,580	4,270	6,010	51,700	206,440	344,280	475,040	606,320

NRC Letter: January 31, 1983

Question No. Q470.2 (Section 5.2.4)

Section 5.2.4 entitled, "Dose Rate Estimates for Man," states that calculated doses to the 80 km population are based on the year 2006 population; however, nowhere in the EROLS are population estimates provided for this year. Please provide a table for the year 2006 similar to Tables 2.1-5 and 2.1-12 at the radial distances specified above.

Response:

The year in Section 5.2.4 has been corrected to 2010. Population estimates for the year 2010 are provided in Tables 2.1-6 and 2.1-13.

5.2.4 Dose Rate Estimates for Man

Calculated doses to the maximum offsite individual and for the population within the 80-km radius for the year 2010 are based on the gaseous and liquid releases shown in Sections 3.5.2 and 3.5.3. The mathematical models and assumptions used to calculate these doses are given in Appendix F. |2 Q470.2

5.2.4.1 Gaseous Pathways

Tables 5.2-8 through 5.2-19 present the calculated doses to maximum individual from gaseous pathways. These tables present the calculated total body and organ doses for four age groups - adult, teen, child, and infant. The analysis was performed for locations where an existing resident, milk cow, and milk goat were identified. Each analysis considers existing and potential pathways at the specified location. For example, it was assumed that a garden existed at all locations which had milking animals. The maximum individuals residing at those farms were analyzed for inhalation, ground deposition, ingestion of vegetation, and consumption of milk.

Tables 5.2-8 through 5.2-11 present the doses to the maximum individuals living at the estimated maximum resident location (0.81 km east northeast).

Tables 5.2-12 through 5.2-15 present the estimated doses to the maximum individual living at the maximum goat milk animal location (2.4 km north northeast).

Tables 5.2-16 through 5.2-19 present doses to the maximum individuals from consumption of cow milk.

Table 5.2-23 presents the comparison of the maximum individual calculated doses from gaseous effluents to the design objectives of Appendix I limits (U.S. NRC 1976).

Annual calculated gamma air dose and beta air dose values were determined. Table 5.2-23 compares these values to the 10CFR50 design objective limit values. The maximum values occurred at the site boundary in the east northeast direction at a distance of 650 meters from Millstone 3.

5.2.4.2 Liquid Pathways

Tables 5.2-20 through 5.2-22 present the calculated doses to the maximum individual from liquid pathways. The tables present the calculated total body and organ doses for three age groups; adult, teen, and child. It was assumed that an infant would not exceed the doses calculated for a child and, therefore, a separate table was not provided.

Table 5.2-23 presents a comparison of the maximum individual calculated doses from liquid effluents to the design objectives of Appendix I limits (U.S. NRC 1976).

NRC Letter: January 31, 1983

Question No. Q470.3 (Section 5.2.4)

Explain the difference in the doses discussed in Section 5.2.4.4.1, "Eighty-km Radius Population Dose," and Section 5.2.4.6, "Annual Population Doses." These appear to be the same doses, except that in one case population data for the year 2006 was used, and in the other case population data for the year 2010 was used.

Response:

The year in Section 5.2.4.4.1 has been corrected to 2010.

5.2.4.3 Direct Radiation from Facility

The station radiation shield analysis is based on conservative operating parameters, which include a nuclide inventory associated with 1 percent fuel defects. The shield walls are designed to meet the operational dose criteria of 0.25 mRem per hour in the yard areas. The most significant structures which contribute to the yard dose rate are the containment fuel building, waste disposal building, auxiliary building, refueling water storage tank, and the boron recovery tanks.

The calculated dose rate levels in the unrestricted areas are based on full power normal plant operations assuming fuel defects producing expected quantities and concentrations of radioactive nuclides consistent with NUREG 0017.

The annual dose rate at the site boundary is approximately 0.43 mRem per year. The annual dose rates computed as a function of distance from the site boundary are given on Figure 5.2-4.

5.2.4.4 Annual Population Doses

5.2.4.4.1 Eighty-km Radius Population Doses

Population dose commitments are calculated for all individuals living within 80 km of the facility, employing the same models used from individual doses (Regulatory Guide 1.109).

Table 5.2-24 presents the calculated annual total body and thyroid doses from gaseous and liquid pathways to the population projected for the year 2010, residing within an 80-km radius of the site. Due | 2
to the long travel times and high dilution factors involved in Q470.3
analyzing population doses from liquid pathways out to the 80-km radius, the dilution factor and travel time to the 16-km radius is conservatively assumed to be representative of the 80-km radius. Thus, population doses from ingestion of aquatic foods, swimming, and boating will be determined using the dilution factor and travel time calculated for the 16-km radius.

5.2.4.4.2 Contiguous U.S. Population Doses

In addition to the 80-km (50-mile) radius population doses, population doses associated with the export of food crops produced within the 80-km (50-mile) region and the atmospheric and hydrospheric transport of the more mobile effluent species, such as noble gases, tritium, and carbon-14, were calculated.

Table 5.2-25 presents the calculated annual total body and thyroid doses to the contiguous U.S. population.

NRC Letter: January 31, 1983

Question No. Q470.4 (Section Appendix 7)

Table F-2 should be amended to include both the sport and commercial invertebrate annual harvest for the 80 km region.

Response:

The total invertebrate harvest was calculated using seafood consumption rates for an average adult, teen, and child and the 2010 population. For purposes of calculating doses, it was conservatively assumed that 50 percent of the total was sports catch and 50 percent commercial.

TABLE F-2

PARAMETERS AND ASSUMPTIONS USED IN
EQUATIONS FOR ESTIMATING DOSES TO HUMANS

All parameters and assumptions used are recommended values to be used, in lieu of site specific data, from Regulatory Guide 1.109, Revision 1.

The following are site specific parameters or parameters for which there is no recommended value:

F = normal circulation flow rate (for 3-unit operation) = 4,160 cu ft/sec

T_p = transit time = see Table F-1

Note: T_p used in calculations was increased, where appropriate, by the distribution or holdup time recommended by Regulatory Guide 1.109, Revision 1.

P = fractional equilibrium ratio of C^{14} = 1 (continuous release); = 0.0073 (intermittent release)

Q_i = annual release rate of radionuclide i, Ci/yr (Tables 3.5-11 and 3.5-14)

f_p = fraction of year animals graze on pasture = 0.67 (8 months)

f_s = fraction of daily feed which is pasture grass when animal is grazing = 1 (100%)

H = absolute humidity of atmosphere at location of analysis 7.10 g/cu m

U_{ap} = usage factor (hr/year of exposure):

Maximum individual	Adult	Teen	Child
Swimming	100	100	56
Boating	52	52	29

80-km radius population:

	Adult	Teen	Child
Swimming	3.5	19	12
Boating	29	29	16.5

V_p = Total commercial U.S. fish harvest = 1.1E+09 kg/yr*

V_p' = Total commercial U.S. shellfish harvest = 5.2E+08 kg/yr

TABLE F-2 (Cont)

V_{dp}	= 80-km commercial fish harvest	= 9.7E+06 kg/yr
V_{dp}'	= 80-km sports fish harvest	= 9.7E+06 kg/yr
V_{dp}''	= 80-km invertebrate harvest**	= 2.8E+06 kg/yr
V_{dp}'''	= 80-km milk production	= 5.6E+08 l/yr
V_{dp}''''	= 80-km meat production	= 1.7E+07 kg/yr
V	= 80-km vegetation production	= 1.8E+08 kg/yr

NOTES:

* 1.1E+09 = 1.1×10^9

**Includes sports and commercial harvest (analysis conservatively assumes 50% sports and 50% commercial) | 2 Q470.4

Attachment 2

Responses to Enciosures 4 through 11 (Enclosure 4, Items 1-8)

Response to NRC Acceptance Letter, Enclosure 4

1. Safety Related Q-List

2. Fracture Prevention - QDC51

3. Instrumentation - Inadequate Core Cooling

4. Preservice and Inservice Inspections

5. Preservice Inspection and Testing of Snubbers

6. Effects of Containment Coatings and Sump Debris on ECCS

7. Seismic Qualification

8. Station Blackout Procedures

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 1 Safety-Related Structures, Systems, and Components (Q-list) Controlled by the QA Program - Staff requests for additional information regarding this issue have been sent to a number of OL applicants. A sample request from the Diablo Canyon review is provided as Enclosure 5.

Response:

FSAR Table 3.2-1, which provides a listing of safety related structures, systems, and components, is being reviewed against the items listed on Enclosure 5 to the NRC acceptance letter. A revision of this table to reflect those items applicable to Millstone 3 will be provided in August 1983.

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 2 Fracture Prevention of Containment Pressure Boundary (GDC 51) - Enclosure 6 provides the technical basis by which the staff determines compliance with GDC 51.

Response:

Refer to FSAR Section 3.1.2.51 Amendment 2.

postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws."

Design Conformance

Ferritic materials for the containment structure boundary are specified so that, when the liner, equipment latch, personnel lock, penetrations, and fluid system components, including valves required to isolate the system are exposed to postulated accident, test, operating and normal conditions, the corresponding and resultant stress levels are below the maximum stress level permitted by the crack arrest temperature (CAT) curve of NRL Report 6900 for each applicable correspondence temperature condition. RAI-4-2

Nil Ductivity Transition Temperature

Figure 23 of NRL Report 6900 shows the fracture analysis diagram (FAD), which plots stress vs temperature in excess of nil ductility transition temperature (NDTT). The containment structure liner is designed so that no stress exceeds the CAT curve shown in this FAD. This approach is very conservative and ensures that flaws of any size will not be propagated to a rapid (i.e., brittle) fracture.

Uncertainties in the determination of NDTT are minimized by using the drop weight test, ASTM E208, for material 5/8 inch or thicker. Charpy V-notch tests are required for all material which form part of the containment structure boundary.

3.1.2.52 Capability for Containment Leakage Rate Testing (Criterion 52)

Criterion

"The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure."

Design Conformance

The design of the containment structure and related equipment, which are subjected to the containment structure test conditions, as described in Section 6.2.6, allows for conducting periodic integrated leakage rate testing of the containment structure.

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 3

Instrumentation for Detection of Inadequate Core Cooling - (TMI Action Item II.F.2 in NUREG-0737) - Discussion of this item should address how core thermocouple readouts are provided in the control room including location and rate of printout (see Part (4) of Attachment 1 to Item II.F.2).

Response:

The requested information requires detailed information on the displays for Inadequate Core Cooling (ICC) Instrumentation. Since ICC instrumentation readouts and data acquisition are an integral part of the Safety Parameter Display System (SPDS) a schedule for addressing this request for information is not possible until the schedule for the implementation of the SPDS has been determined. We intend to address SPDS implementation schedule in our response to NRC generic letter 82-33 on April 15, 1983. We will provide a schedule for responding to this issue in June 1983.

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 4

Preservice and Inservice Inspections - Staff guidance in this review area has been sent to a number of pending OL applicants. A copy of that guidance is provided as Enclosure 7.

Response:

For a schedule for the submittal of the PSI (as-built) program and the first 10 year ISI plan, see the response to Q250.1.

REQUEST FOR ADDITIONAL INFORMATION

Enclosure 4, Item 5 NRC Letter: January 31, 1983

Preservice Inspection and Testing of Snubbers - The staff has recently established requirements to ensure snubber operability which have been transmitted to pending OL applicants. A copy of those requirements is provided as Enclosure 8.

Response:

Procedures for preservice inspection and testing of snubbers are being developed. Requirements outlined in Enclosure 8 of the NRC acceptance letter will be addressed in these procedures which will be provided in FSAR Chapter 14 as an amendment.

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 6 Effectives of Containment Coatings and Sump Debris on ECCS and Containment Spray Operation - A copy of the staff concerns on this issue, including a request for additional information which has been sent to a number of OL applicants, is provided as Enclosure 9.

Response:

Item (1) - Refer to the response to Q480.3.

Item (2) - A visual examination of the containment recirculation sump is required during each refueling for evidence of structural damage or corrosion. The Applicant will observe the containment sump during the containment closeout for any evidence of these problems.

Item (3) - The Applicant will factor recommendations identified as a result of vortex studies into the emergency operating procedures.

Items (4) and (5) - Refer to the response to Q480.3.

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 7 Seismic Qualification - A staff request for additional information in this review area has been sent to a number of pending OL applicants. A copy of that request is provided as Enclosure 10.

Response:

(1) Conformance with Regulatory Guides 1.92 and 1.100 can be found in FSAR Section 1.8. Differences to SRP's 3.9.2, 3.9.3, and 3.10 are documented in FSAR Section 1.9

(2) A list of all safety related equipment is being prepared using the tabular format of Attachment 1 to enclosure 10 of the NRC acceptance letter and will be submitted in September 1983.

(3) Specific information on equipment selected for audit by the NRC/SQRT will be provided in accordance with Attachment 3 to enclosure 10 of the NRC acceptance letter after submittal of the safety related equipment list.

REQUEST FOR ADDITIONAL INFORMATION

NRC Letter: January 31, 1983

Enclosure 4, Item 8 Procedures and Training for Station Blackout - In response to a recommendation in a recent decision by the Atomic Safety and Licensing Appeal Board (ALAB-603), to ensure that station blackout events can be accommodated, the staff is requesting licensees and OL applicants to implement emergency procedures and a training program for station blackout events. A copy of that request is provided as Enclosure 11.

Response:

Appropriate station procedures and training programs for station blackout events will be completed prior to the fuel load date of November 1985.