NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
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NORTHEAST NUCLEAR ENERGY COMPANY

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March 20, 1984

Docket No. 50-423
B11092

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

References: (1) B. J. Youngblood to W. G. Counsil, Request for Additional Information for Millstone Nuclear Power Station, Unit No. 3, dated January 16, 1984.

(2) W. G. Counsil to B. J. Youngblood, Millstone Nuclear Power Station Unit No. 3, Transmittal of Amendment 7 to the FSAR and Responses to Selected Requests for Additional Information, dated March 9, 1984.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Submittal of Responses to 480 Series Questions

In Reference (1), Enclosure 1, you requested additional information in the area of containment systems. By Reference (2) eleven responses were provided. Attached are the remaining responses to your requests. These responses are as they will appear in the next Amendment to the FSAR, (Amendment No. 8). As such you should now have all responses to the 480 series questions forwarded to date except for request numbers 480.11, 19 and 34. These will be submitted on or before April 10, 1984. Please note that we have renumbered the requests forwarded in Reference (1) by adding five to each request number. This was done to take into account five CSB requests for additional information forwarded as a result of the NRC's mini-review.

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If you have any questions 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

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.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

ATTACHMENT 1

Remaining 480 Questions not forwarded in Amendment 7

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NRC Letter: January 16, 1984

Question No. Q480.11 (Section 6.2.1)

In FSAR Section 6.2.1.3, it is stated that the mass and energy release data for the postulated LOCA were generated using the methodology described in a reference letter, NS-TMA-2075, from T.M. Anderson, W, to J.F. Stolz, NRC, April 25, 1979 (Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version). This report is still under staff review and has not been approved.

Provide mass and energy release data for the spectrum of LOCAs analyzed using the acceptable method in WCAP-8312-A and, in addition, provide a comparison for the worst case LOCA of the containment pressure response using the mass and energy release data from WCAP-8312-A versus NS-TMA-2075.

Response:

The response to this question will be submitted at a later date.

NRC Letter: January 16, 1984

Question No. Q480.12 (Section 6.2.1)

In FSAR Section 6.2.1.4.1, it is indicated that the mass and energy release analysis for the MSLB were based on a proprietary report - Model F Steam Generator Mass and Energy Release Data, 1978. Provide us with the report for review and evaluation. Tables 6.2-57 and 6.2-58 in the FSAR were blank pages, and were indicated as proprietary data, having been forwarded to the NRC under separate cover. The review branch has not been able to retrieve this data. Therefore, provide these two tables for our confirmatory analysis.

Response:

Refer to revised FSAR Section 6.2.1.4.1, included in Amendment 6.

NRC Letter: January 16, 1984

Question No. Q480.13 (Section 6.2.1)

It is stated in FSAR Section 6.2.1.4.4, that the computer codes, TRANFLO and MARVEL, were used to calculate the steam generator blowdown. Some model modifications were made by Westinghouse during the course of the staff's review of these two codes. The modifications, pertaining to the steam generator water level calculations, and heat transfer to steam during tube bundle uncovery, are described in letters from F. Rahe (Westinghouse) to J. Miller (NRC) dated November 23, 1981, and February 17, 1982. Clarify whether these model changes have been incorporated into the Millstone 3 analyses. If not, provide justification or assess the impact for not including these changes.

Response:

Westinghouse has performed studies to assess the sensitivity of containment response to mass and energy releases following a steamline rupture and subsequent tube uncovery. The results pertaining to a dry-type containment are applicable to Millstone 3. However, this is proprietary information. Please refer to the Westinghouse letter, dated February 17, 1982 from E. P. Rahe, Jr. to J. R. Miller of the NRC. This discusses, in a proprietary manner, the effect on containment response due to MARVEL code modifications.

NRC Letter: January 16, 1984

Question No. Q480.17 (Section 6.2.2)

According to Section 6.2.2, the containment recirculation pumps are located outside of containment. The tandem mechanical seals of the pumps are filled with demineralized water which is maintained at a pressure slightly greater than the pump discharge by a seal head tank. This arrangement assures that outer seal failure will not result in radioactive fluid leaking out of the containment boundary. Since the seal head tank prevents leakage of containment recirculation water out of the containment, provide the following:

(a) Verification that the tank and associated piping are Seismic Category I and Safety Grade; and,

(b) The design basis used for determining the capacity of the seal head tank.

Response:

a. A seal head tank and associated cooling coil (pipe) are attached directly to each containment recirculation pump. The pumps are seismic Category I. Additionally, the seal head tank and associated piping are qualified Seismic Category I. They are also designed to ASME Section 3 criteria and fabricated from ASME materials.

b. The design criteria used for determining the capacity of the seal head tank are:

heat dissipation

make-up reservoir

The tank and coil are sized to provide a radiation area for heat dissipation and sufficient fluid to make-up for seal system leakage (7 days design leakage without make-up). The tank is equipped with high and low level alarms and a piston arrangement which, in conjunction with the pumping ring, provides the 1 psi overpressure to the tandem seal.

NRC Letter: January 16, 1984

Question No. Q480.18 (Section 6.2.2)

Provide the following additional information with respect to the containment emergency sump performance:

(a) Specify the total flowrate of water (GPM) drawn from the sump under minimum and maximum safeguards operation following a LOCA. Give the actual pump flow rates (GPM) for these conditions.

(b) Provide the approach flow velocity at the sump screens for the above conditions.

(c) Provide a table listing the quantities and location of the various types of insulation employed inside the containment, for piping systems 8 inches in diameter and larger.

Response:

a. The containment recirculation system flowrates for minimum and maximum ESF operation are given in Table Q480.18-1.

b. The water approach velocities at the containment sump fine mesh screens are given in Table Q480.18-1.

   A further discussion of sump water velocity is included in revised FSAR Section 6.2.2.4.2.

c. Quantities and locations of insulation inside containment are provided in revised FSAR Section 6.2.2.2 and new FSAR Table 6.2-71.

TABLE Q480.18-1

CONTAINMENT RECIRCULATION SYSTEM FLOWS AND SUMP SCREEN APPROACH VELOCITY

	Minimum ESF		Maximum ESF	
	Before Switchover[1]	After Switchover	Before Switchover	After Switchover
Spray Flow per Pump (gpm)	3880	3880	3150	3880
Injection Flow per Pump (gpm)	-	4200	-	4200
Total Flow From Sump (gpm)	7760	8080	12,600	16,160
Sump Screen Approach Velocity (ft/sec)[2]	0.142	0.148	0.230	0.296

NOTES:

1. Switchover to the recirculation mode of safety injection.

2. Assumes 50 percent screen area blockage.

the water in the containment structure sump after a DBA, including the contents of the RWST, is between 7.0 and 7.5.

The borated water in the RWST is maintained at a maximum temperature of 50°F by circulating the RWST water through the refueling water coolers, which use chilled water from the chilled water system (Section 9.2.2.2). The RWST is insulated to limit the temperature rise of the water to 1/2°F, or less, per 24 hour period whenever the chilled water system is inoperable. Periodic sampling of the RWST water monitors the water's chemistry. Provisions are made to purify the water when necessary, by circulating the water through the fuel pool cooling and purification system (Section 9.1.3).

A vortex suppression assembly is installed in the RWST at the quench spray suction lines to eliminate vortex formation. The assembly consists of a single horizontal plate above both suction nozzles, supported off the bottom of the tank by vertical vanes. The quench spray pumps are automatically tripped at the low-low-low RWST level, which is set so that with allowance for negative instrument error, vortex formation will not occur. 440.36

The RWST also has a connection for supplying water to the ECCS. The RWST is provided with a manhole for inspection access during refueling periods.

Refer to Section 6.3.2.8 for a discussion of RWST design relative to instrument error, working allowance, ECCS switchover allowance, most limiting single failure, and compliance with design basis. 440.36

Each quench spray pump is capable of supplying approximately 4,000 gpm of sodium hydroxide/borated water solution to the two 360 degree quench spray headers located approximately 101 and 116 feet above the operating floor in the dome of the containment structure. The pumps are located in the engineered safety features building adjacent to the containment structure. Each quench spray discharge line contains a check valve inside containment and a motor-operated isolation valve outside the containment structure.

The preoperational test is described in Section 6.2.2.4. The design evaluation of the system is contained in Section 6.2.2.3. Small diameter drain lines, located downstream from the check valves within the containment structure, drain the quench spray headers should any water enter the headers during periodic testing. The size of the drain lines does not significantly decrease the capacity of the quench spray system during operation.

Containment Recirculation System

Each of the two containment recirculation subsystems consist of two containment recirculation coolers and pumps which share two 360 degree spray headers. Each containment recirculation spray header is fed by two risers, each riser running from one of the containment recirculation coolers in each of the subsystems. The two pumps in each subsystem are connected to different spray headers, but

they are both connected to the same emergency bus. Failure of one emergency bus does not prevent delivery of sufficient containment recirculation flow.

The four containment recirculation pumps take suction from a common containment sump, which is enclosed by a protective screen assembly. Three stages of trash rejection are provided: a 1 1/2-inch grating, coarse mesh, and fine mesh. The approximate screen sizes are as follows: coarse mesh, normal mesh (3/8 inch), fine mesh, and a
480.18 3/32-inch separation screen at the center of the sump. The grating and screens are erected vertically around the sump perimeter, as shown on Figure 6.2-38. The assembly is divided at the centerline by fine mesh screening so that failure of either half does not adversely affect the other half. The containment recirculation pumps from each subsystem take suction from each half of the sump. If half the screen assembly should become clogged, water is still available to all suction points via the screening separating the two sections of the sump. There is also a 1 1/2-inch grating at elevation 24 feet-6 inches which covers the sump and acts as a vortex breaker to prevent air entrainment in the pumps. Refer to Section 6.2.2.4.2 for a discussion of the containment sump model tests.

Both the fine mesh and the fine mesh screening separating the sump
480.3 halves have an opening size (3/32 - inch) that is smaller than the gap of the smallest coolant passage in the reactor core (1/8 - inch) and smaller than the orifice of the spray nozzles, (3/8 - inch).

All four containment recirculation pumps and motors are located outside the containment structure. The pumps are of the vertical deep well type, each mounted in a separate stainless steel well casing supported by the concrete containment structure mat. The pumps are located adjacent to the containment structure at an elevation sufficiently below the containment structure sump to ensure an adequate available net positive suction head (NPSH). Access to the motors for inspection and maintenance is provided. Each containment recirculation pump has a design flow of approximately 3,950 gpm. The containment recirculation pumps are started approximately 220 seconds after the containment depressurization actuation (CDA) signal. Each containment recirculation pump shaft is fitted with a tandem mechanical seal arrangement. The cavity between the mechanical seals is filled with demineralized water, which is maintained at a pressure slightly greater than the containment recirculation pump discharge pressure by a seal head tank. This arrangement ensures that, should the outer seal fail, leakage will be from the higher pressure demineralized water; thus, preventing leakage of the containment recirculation water which might be radioactive. Either type of failure is detected by a level alarm provided for the seal head tank.

The containment recirculation coolers are conventional shell and tube heat exchangers with containment recirculation water flowing through the shell, where the water is cooled by service water flowing in the tubes. The service water flow to each cooler is 6,500 gpm. The heat transfer duty for the coolers varies throughout the DBA. This is due

to the reduction in the temperature of the water on the containment structure floor. Each containment recirculation cooler is sized to have a UA of approximately 3.87 x 10^6 Btu/hr/°F. The service water temperature range is 33-75°F.

The containment recirculation coolers are welded at all points where there is a potential for leakage of radioactive containment recirculation water into the service water. Because the containment recirculation water pressure in the coolers is greater than that of the service water, only outleakage can occur and dilution of the borated water by service water is not possible. This ensures that the margin necessary for cold shutdown by boron is maintained.

The service water from each of the containment recirculation coolers is monitored by a radiation monitor which actuates an alarm if outleakage occurs. If outleakage is detected, the affected cooler is then isolated. Section 11.5 describes the radiation monitoring devices and techniques that are employed.

During normal unit operation, the containment recirculation coolers are kept clean and dry, with maximum heat transfer capability. For long term operation, on the order of weeks, there may be some fouling of the tubes on the service water side, with resultant loss in heat transfer capability. A fouling factor of zero is assumed, because the loss of the heat transfer capability is more than offset by the decrease in the decay heat production rate. The rate is such that one containment recirculation pump and cooler has more than a sufficient amount of heat removal capacity to hold the containment structure depressurized.

In all spray headers, a combination of spray nozzle orientations is used to obtain maximum coverage. The mean surface diameter of the spray droplets is less than 1,000 microns at a design pressure drop of 40 psi, for the QSS and 25 psi for the CRS. The average vertical fall height, considering the location of the spray headers and spray particle trajectories, is in excess of 101 ft for the quench spray and 87 feet for the containment recirculation spray.

The four containment recirculation pumps and the associated suction line valve and motors outside the containment structure are designed and installed to account for the differential movement between the pumps in the ESF building and the containment structure. Restraints and supports are used as appropriate.

The containment structure floor is sloped and channeled to ensure that sufficient water is provided to the containment structure sump at the time that the containment recirculation pumps are started. All cubicles drain to the containment structure floor. This design ensures that almost all water discharged into the containment structure during a LOCA reaches the sump.

During recirculation, leakage could occur through valve packings and from leaks in the suction and discharge piping of the containment

recirculation pump. Valves are appropriately selected to reduce this potential leakage to a negligible amount.

Consistent with letters from the ACRS (Hanover 1969) concerning vital piping which must function during a DBA, passive failure of the containment recirculation suction piping during a DBA is not considered credible during the short term period following the start of the DBA. However, if a passive failure occurs following depressurization where subatmospheric pressure has been achieved, the valve pit area floods with containment recirculation water. The flooding provides a water seal to prevent inleakage of outside air into the containment structure and the subsequent return of the containment structure to atmospheric pressure.

Insulation

Removable type encapsulated insulation is used on most piping within containment. Encapsulated insulation consists of multiple layers of Type 304 austenitic stainless steel sheets filled with fiberglass
480.18 composition and encased by inner and outer jacketing of type 304 austenitic stainless steel sheets. The minimum thickness of the inner jacketing is .010 inches and of the outer jacketing is .018 inches. Design details permit tight interlocking of adjacent sections of the assembled insulation. Where removal of insulation is required, quick release mechanical fasteners are provided. Some
480.18 piping 3 inches and smaller is insulated with encapsulated fiberglass blankets enclosed in stainless steel lagging.

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
480.18 encapsulated 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.

The allowance for 50 percent plugging or blockage of the sump is conservative. Lighter particles will float on the water surface
480.3 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.

The remaining piping requiring general thermal insulation, is insulated with fiberglass or foam glass type insulation 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. 480.18 480.18

Small amounts of insulation,such as "Min-k" and "Foamglas", are utilized in areas where the installation of encapsulated insulation is impractical. Refer to Table 6.2-71 for quantities and locations of the various types of insulation employed inside the containment. 480.18

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 480.3

(-)22 foot-6 inch elevation without vortex suppression grating. Since the minimum LOCA water elevation during pump operation is estimated at (-)23 feet-10 inches, vortex suppression is required. Tests with the suppression grating in place indicate that the sump performance is acceptable at the minimum estimated sump water level.

The maximum calculated sump water velocity at the fine mesh screens is less than 0.15 ft/sec assuming no screen blockage. The results of debris transport tests (Brocard 1982 Table B-3) indicate that a sump water approach velocity greater than 0.2 ft/sec is necessary to induce any debris to flow toward the screens. If 50 percent blockage of the fine mesh screens is assumed, the calculated sump water velocity increases to about 0.3 ft/sec. The debris transport tests indicate that velocities greater than 0.3 ft/sec are required to transport significant amounts of debris to the sump screens. 480.18

Additionally, a full flow test of the service water through the tube side of each containment recirculation cooler ensures that the required flow and head for effective system operation is achieved (Section 9.2.1.4). Following the test, the coolers are flushed with demineralized water, and left in a drained and ready condition, making further testing of the coolers unnecessary.

Proper functioning of interlocks, time delays, alarms, instruments, and valves during both the spray mode and switchover to recirculation mode will be verified during a simulated system actuation test. Valve speed and positioning will be verified in the control room and by local visual observation.

For inservice inspection and flow testing, the containment recirculation pumps are capable of being flow tested monthly. Closing the containment isolation valves in the pump suction and discharge, opening the locked closed valves in the test line from the RWST, opening the valve to the low pressure safety injection discharge line, and opening the valve in the residual heat removal return line to the RWST, permits flow testing of the containment recirculation pumps. The pump developed head and the measured flow will be compared to the pump performance curve (Figure 6.2-40) to verify inservice inspection acceptability.

Pressure retaining components are inspected for leaks from pump seals, valve packings, flanged joints, and safety valves during system testing. In addition, Safety Classes 2 and 3 pressure retaining components are subject to periodic inservice inspection, as described in Section 6.6.

Means are provided for airflow tests through the containment recirculation spray nozzles at intervals indicated in the Technical Specifications. This test will also be performed during the final stage of preoperational testing for this system to verify that the nozzles are not plugged.

The containment quench and recirculation spray systems are principal plant safety features and are normally inoperative during reactor

operation. Complete system tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, and main feedwater and containment isolation. A containment spray system test would require the system to be temporarily disabled. The method of assuring operability of this system is, therefore, to combine system tests normally performed during plant refueling shutdowns, with more frequent component tests, (i.e. motor-operated valves) which can be performed during reactor operation. The system test, at or between each major fuel reloading, demonstrates proper automatic operation of the containment spray systems. With the pumps blocked from starting, a test signal is applied to initiate automatic actuation and verify that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry including operation of the 240-second time delay relay in the containment recirculation pump startup system.

During reactor operation, the control room instrumentation, which initiates the containment spray system is checked periodically and the initiating circuits tested monthly. The testing of analog channel inputs is accomplished in a similar manner as the reactor trip system. The engineered safety features logic system is tested by means of a semi-automatic tester to simulate digital inputs from the analog channels. The semi-automatic tester uses short duration pulses to prevent master relay actuation. Verification of logic actuation is indicated by a test light. Upon completion of the logic checks, verification that the circuit from the master relay to the slave relays is complete, is accomplished by use of a built-in slave relay tester to check continuity. In addition, the active components (pumps and valves) are tested periodically, as indicated in the Technical Specifications. The test checks the operation of the starting circuits and verifies that the pumps are in satisfactory running order. Testing of containment quench and recirculation spray systems instrumentation and controls is discussed in Section 7.3.

6.2.2.4.3 Chemical Addition Subsystem

The chemical addition subsystem is initially tested in conjunction with the ECCS pump and the quench spray pump preoperational test. The chemical addition tank (CAT) is normally filled with sodium hydroxide, but for the preoperational test the CAT is filled with water to a height of 10 percent higher than the RWST water level to simulate the difference in specific gravity to NaOH from that of water. This provides the same mass flowrate from the CAT during the test as during normal operation when the fluid levels in the CAT and RWST are the same. Water is pumped from the RWST into the cold legs of the reactor coolant system using the ECCS pumps. With the reactor vessel head removed, this injection water flows into the refueling cavity. A flow path for the quench spray pumps is also established for injection into the refueling cavity through the pump test line. Return flow to the RWST is isolated, and the internals of the check valve from the residual heat removal system are removed. The flow path is then through the residual heat removal and low pressure

safety injection system to the hot legs of the reactor coolant system. The purpose of this test is to verify that the CAT maintains hydrostatic balance with the RWST during drawdown and thus the ratio of flows from the CAT and RWST is constant.

Two tests will be performed, one for minimum safeguards and one for maximum safeguards. These tests will bound all possible flow scenarios and confirm that proper drawdown behavior is obtainable.

To test for minimum safeguards, one charging pump, one safety injection pump, one residual heat removal pump, and one quench spray pump are aligned to draw from the RWST. Only one of the two parallel CAT block valves is opened in order to maximize frictional pressure losses between the CAT and the RWST.

To test the operation of the system during maximum safeguards, two charging pumps, two safety injection pumps, two residual heat removal pumps, and two quench spray pumps are aligned to draw from the RWST. In this test, both of the two parallel CAT block valves are opened.

With the drawdown rate from the RWST and CAT established, verification of hydrostatic balance can be obtained. During this test, losses in the piping from the CAT are maximized when only one block valve is open, and minimized when both valves are opened. Therefore, any potential for hydrostatic imbalance is exaggerated. Quench spray and ECCS pump flows and the rates of level change in the RWST and CAT are measured using the flow and level indicators shown on Figures 6.2-36 and 6.3-1. The flow from each tank is determined from the rate of change of the tank level. The combined flow from the RWST and CAT is checked against the flow through the quench spray and ECCS pumps. A ratio of CAT flow to RWST flow is then obtained. From this flow ratio and the ratio of measured ECCS/QSS flows, a final adjustment can be made to the NaOH concentration in the CAT to provide the required minimum pH as discussed in Section 6.1.1.2. In this way, flow deviations are accommodated and the ability of the system to produce the additive concentrations to meet the design basis pH values in Table 6.1-2 is verified.

To maintain an adequate supply at the required concentration of NaOH, inspections are made periodically as indicated in the technical specifications to measure both the fluid level and the NaOH concentration of the CAT.

The CAT contains a caustic (NaOH) solution with a concentration 1.35 to 2.0 percent by weight with an ambient temperature below 125°F. The corrosion characteristics of stainless steel with such NaOH solutions are such that the design characteristics of the CAT will not be degraded over the 40-year life of the plant. Reaction of NaOH with the stainless steel forms $FeO-H_2O$, which is insoluble in an alkaline solution. This reaction does not reduce the alkalinity of the solution and, therefore, does not impair the effectiveness of the NaOH solution. The cover gas for the tank is air which contains

Containment isolation valve testing (Type C tests) is performed prior to initial criticality and periodically, thereafter, during each reactor shutdown for refueling, but in no case at intervals greater than 2 years.

A report of each periodic Type A test is submitted to the Nuclear Regulatory Commission (NRC). The report contains an analysis and interpretation of the Type A test results. In addition, the report has a summary analysis of the periodic Type B and C tests performed since the last Type A test.

If any periodic Type A test fails to meet the acceptance criteria, the schedule for subsequent Type A tests is subject to review and approval by the NRC. If two consecutive Type A tests fail, Type A testing must be performed during each refueling outage or at intervals not exceeding 18 months until two consecutive Type A tests meet the acceptance criteria, at which time the previous schedule may be resumed.

6.2.6.5 Special Testing Requirements

Type A, B, and C tests, as applicable, are conducted following containment structure modifications in accordance with Paragraph IV.A of Appendix J, 10CFR50.

A special test to verify the allowable in-leakage to the subatmospheric containment is not required as the integrated leak-rate test described in Section 6.2.6.1 adequately demonstrates the leak tightness of the containment.

An evaluation of in-leakage following a LOCA shows the containment pressure to be effectively subatmospheric at -0.5 psig 30 days following the accident. The inleakage analysis is based on the maximum specified out-leakage rate of 0.9 percent per day at approximately 45 psig adjusted to the pressure differences determined to be present following a LOCA. 480.22

The maximum in-leakage rate to the subatmospheric containment during normal operation is approximately 14 scfm at 9.5 psia, the lowest normal operating containment pressure. This corresponds to the out-leakage rate of 0.9 percent per day at 45 psig adjusted for the pressure differential and other important flow paramaters.

The containment structure enclosure will be evacuated by the supplementary leak collection and release system (SLCRS) to slightly negative pressure immediately following the design bases accident initiation of the engineered safety features actuation system (ESFAS). This will ensure all leakage from the primary containment (0.9 percent per day) is passed through the high-efficiency particulate air (99-percent efficient) filters of the SLCRS prior to release from the containment structure enclosure, engineered safety feature building, main steam valve building, hydrogen recombiner building or auxiliary building which are all connected to the SLCRS.

This filtration will ensure the reduction of primary leakage from 0.9 percent per day to less than 0.1 percent per day released to the environment. The SLCRS will be tested prior to loading fuel to verify that a slightly negative pressure can be obtained and maintained following an ESFAS actuation in the areas mentioned above. This test will be conducted again at each refueling or at intervals not to exceed 18 months. Some leakage through piping systems may bypass the secondary containment. This leakage is limited to the design leak rates through these piping systems. The bypass leakage penetrations, identified in Table 6.2-65, are tested in accordance with Section 6.2.6.3, and the combination of their leakage rates is compared with the maximum allowable rate (9 scfh). When the actual leakage rate approaches this limit, corrective action will be taken.

6.2.7 References for Section 6.2

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TABLE 6.2-71

PIPE INSULATION INSIDE CONTAINMENT
(8 inches and larger)

System and Line No.	Removable Encapsulated Thickness (inches)	Pipe Size (inches)	Linear Feet
Main Feedwater System			
FWS01607302	3	16	6
FWS01607402	3	16	6
FWS02001802	3	20	235
FWS02002202	3	20	80
FWS02002602	3	20	80
FWS02003002	3	20	235
Main Steam System			
MSS03009202	4	30	174
MSS03009302	4	30	82
MSS03009402	4	30	82
MSS03009502	4	30	174
480.18 Reactor Coolant System			
RCS00802401	3	8	16
RCS00802501	3.5	8	26
RCS00802901	3	8	16
RCS00803001	3.5	8	26
RCS00803401	3	8	16
RCS00803501	3.5	8	26
RCS00803901	3	8	16
RCS00804001	3.5	8	26
RCS01012201	3	10	7
RCS01013201	3	10	7
RCS01013801	3	10	7
RCS01014601	3	10	7
RCS01210301	4	12	13
RCS01212301	4	12	13
RCS01406401	4	14	77
RCS02900101	4	29	11
RCS02900201	4	29	12
RCS02900601	4	29	11
RCS02900701	4	29	12
RCS02901101	4	29	11
RCS02901201	4	29	12
RCS02901601	4	29	11
RCS02901701	4	29	12

TABLE 6.2-71 (Cont)

System and Line No.	Thickness (inches)	Pipe Size (inches)	Linear Feet
RCS03100301	4	31	28
RCS03100801	4	31	28
RCS03101301	4	31	28
RCS03101801	4	31	28
RCS27500401	4	27.5	9
RCS27500501	4	27.5	17
RCS27500901	4	27.5	9
RCS27501001	4	27.5	17
RCS27501401	4	27.5	9
RCS27501501	4	27.5	17
RCS27501901	4	27.5	9
RCS27502001	4	27.5	17
RHS01203301	2.5	12	65
RHS01203501	2.5	12	78
RHS01204302	2.5	12	10
RHS01204402	2.5	12	6
Safety Injection System			
SIL01000902	2	10	6
SIL01001202	2	10	9
SIL01004501	2	10	18
SIL01004701	2	10	18
SIL01004901	2	10	18
SIL01005101	2	10	18
General Anti-Sweat (Figerglass or Foam Plastic)			
Chilled Water System			
CDS010-45-4	1 1/2	10	185
CDS010-74-4	1 1/2	10	185
CDS010-46-4	1 1/2	10	185
CDS010-75-4	1 1/2	10	185
CDS010-104-2	1 1/2	10	10
CDS010-56-2	1 1/2	10	10
CDS008-44-2	1 1/2	8	5
CDS008-105-2	1 1/2	8	5
CDS006-115-4	1 1/2	6	8
CDS006-185-4	1 1/2	6	8
CDS006-186-4	1 1/2	6	8

480.18

NRC Letter: January 16, 1984

Question No. Q480.19 (Section 6.2.2)

Discuss the effectiveness of convection mixing and/or recirculation spray operation to mix combustible gases that may be generated within the containment following onset of a postulated LOCA. Describe the design features of the containment which promote mixing of the atmosphere, and identify the compartments which may not achieve effective mixing of combustible gases. Provide sketches to show the expected circulation patterns within the containment compartments.

Response:

The response to this question will be submitted at a later date.

NRC Letter: January 16, 1984

Question No. Q480.22 (Section 6.2.5)

Discuss the procedures that will be in effect and the surveillance method that will be used to monitor containment integrity, i.e., the in-leakage rate. Specify the maximum allowable containment in-leakage rate during normal plant operation that will ensure that an effective subatmospheric condition will prevail for at least 30 days following onset of a loss of coolant accident.

Response:

Monitoring containment in-leakage during normal operation is not required as the periodic integrated leak-rate test establishes containment integrity (FSAR Section 6.2.6.1).

Refer to revised FSAR Section 6.2.6.5 and 9.5.10.3 for an evaluation of the effect of in-leakage on containment pressure following a LOCA.

Containment isolation valve testing (Type C tests) is performed prior to initial criticality and periodically, thereafter, during each reactor shutdown for refueling, but in no case at intervals greater than 2 years.

A report of each periodic Type A test is submitted to the Nuclear Regulatory Commission (NRC). The report contains an analysis and interpretation of the Type A test results. In addition, the report has a summary analysis of the periodic Type B and C tests performed since the last Type A test.

If any periodic Type A test fails to meet the acceptance criteria, the schedule for subsequent Type A tests is subject to review and approval by the NRC. If two consecutive Type A tests fail, Type A testing must be performed during each refueling outage or at intervals not exceeding 18 months until two consecutive Type A tests meet the acceptance criteria, at which time the previous schedule may be resumed.

6.2.6.5 Special Testing Requirements

Type A, B, and C tests, as applicable, are conducted following containment structure modifications in accordance with Paragraph IV.A of Appendix J, 10CFR50.

A special test to verify the allowable in-leakage to the subatmospheric containment is not required as the integrated leak-rate test described in Section 6.2.6.1 adequately demonstrates the leak tightness of the containment.

An evaluation of in-leakage following a LOCA shows the containment pressure to be effectively subatmospheric at -0.5 psig 30 days following the accident. The inleakage analysis is based on the maximum specified out-leakage rate of 0.9 percent per day at approximately 45 psig adjusted to the pressure differences determined to be present following a LOCA. 480.22

The maximum in-leakage rate to the subatmospheric containment during normal operation is approximately 14 scfm at 9.5 psia, the lowest normal operating containment pressure. This corresponds to the out-leakage rate of 0.9 percent per day at 45 psig adjusted for the pressure differential and other important flow paramaters.

The containment structure enclosure will be evacuated by the supplementary leak collection and release system (SLCRS) to slightly negative pressure immediately following the design bases accident initiation of the engineered safety features actuation system (ESFAS). This will ensure all leakage from the primary containment (0.9 percent per day) is passed through the high-efficiency particulate air (99-percent efficient) filters of the SLCRS prior to release from the containment structure enclosure, engineered safety feature building, main steam valve building, hydrogen recombiner building or auxiliary building which are all connected to the SLCRS.

valves, and instrumentation. The design data for the major components in the containment vacuum system are shown in Table 9.5-8.

The containment vacuum ejector removes air from the containment structure to create a subatmospheric pressure prior to initial unit operation and after subsequent refueling operations. The motive medium for the ejector is 135 psig saturated steam which is supplied at 14,000 pounds per hour from the auxiliary steam system (Section 10.4.10). The containment vacuum ejector discharges directly to the atmosphere through a silencer.

The containment vacuum pumps maintain the containment subatmospheric pressure and control purge backup for the hydrogen recombiner system, meeting the intent of Regulatory Guide 1.7. The common discharge of the two containment vacuum pumps is directed through the radioactive gaseous waste lines (Section 11.3) which are connected to the Millstone 1 stack for elevated release. The system is not required to perform any safety related function. If both hydrogen recombiners fail, one of the pumps may be used for this operation. One pump draws air from the auxiliary building and discharges it into the containment while the second pump continues to remove air from the containment to the radioactive gaseous waste system.

Each containment vacuum pump is capable of removing containment structure air inleakage during normal operation and maintaining containment atmosphere pressure in the operating range of 9.5 to 11.5 psia.

9.5.10.3 Safety Evaluation

The maximum allowable containment atmosphere pressure during unit operation varies primarily as a function of service water temperature. (Technical Specification 16.3/4.6.1.5). The containment vacuum pumps are operated remote manually from the control room to maintain containment atmosphere pressure at or below the maximum permissible value.

Operation of the containment vacuum system is not required for at least several weeks after a DBA; therefore, the system is not an engineered safety feature. The design of the containment structure and containment heat removal systems will maintain subatmospheric containment pressure for 30 days following a LOCA. This allows ample time for repair or replacement of containment vacuum system equipment, if necessary. 480.22

Excessive depressurization of the containment structure is not considered credible. The containment vacuum pumps have a relatively small capacity when compared to the containment structure free volume. Uninterrupted operation of a containment vacuum pump for approximately 48 hours would be required to lower the containment atmosphere pressure from 9.0 psia to the minimum design pressure of 8.0 psia, assuming no air temperature change.

The steam ejector is used for evacuating the containment from atmospheric pressure to subatmospheric pressure during startup operations, in approximately 4 hours, compatible with the normal

NRC Letter: January 16, 1984

Question No. Q480.24 (Section 6.2.3)

Discuss the inspection and testing programs to assure the sealing capability of the enclosure building caulking and the overall leakage integrity of the enclosure building over the life of the plant. Specify the maximum allowable in-leakage rate of the enclosure building.

Response:

Testing is described in FSAR Section 6.2.3.4. The allowable in-leakage of the enclosure building is designed to be .0195 cfm/square foot of the surface area of the enclosure. Corrective action will be taken, as described in FSAR Section 6.2.6, should the SLCRS fail subsequent testing.

NRC Letter: January 16, 1984

Question No. Q480.25 (Section 6.2.3)

According to Figure 6.2-46, the Supplementary Leak Collection and Release System (SLCRS), which is required to draw a negative pressure in the containment Enclosure Building after a LOCA, contains one inlet line with a single valve. Discuss the consequences of this single valve failing closed after a LOCA. Discuss the design changes that will be made to ensure the operability of the SLCRS, assuming a single active failure.

Response:

The valve (damper) referred to here is a volume damper and does not change position during SLCRS operation. The damper is manually adjusted during initial balancing of the system. Thus, the damper is not an active component and not subject to an active failure.

NRC Letter: January 16, 1984

Question No. Q480.26 (Section 6.2.3)

According to Section 6.2.3.3, a radiation monitor will monitor the air being processed by the Supplementary Leak Collection and Release System to warn the operator of a potential problem that will require operator action. Discuss the design criteria for the radiation monitoring system, including redundancy requirements, and the safety grade and seismic classifications. Discuss the operator actions that can be taken and the time available for such actions.

Response:

Refer to FSAR Section 11.5.1 for the design criteria for the process radiation monitoring system.

The monitor was designed in accordance with the safety grade, seismic classification and redundancy criteria of a Type E - Category 2 Common Plant Vent Discharge Monitor as defined in Regulatory Guide 1.97, Rev. 2, Tables 1 and 3.

The design of the system is such that no operator intervention is anticipated. Both trains of filtration are started automatically subsequent to the accident. Once the containment has been depressurized, within 1 hour the operator has the capability of remote manual control of the supplementary leak collection and release system equipment from the control room (UP-1). The adequacy of the charcoal is substantiated by the testing required by Regulatory Guide 1.52.

NRC Letter: January 16, 1984

Question No. Q480.27 (Section 6.2.3)

In order for us to complete our review, we will require the following:

a. Analyses of the pressure and temperature response of the secondary containment to a loss-of-coolant accident within the primary containment and, if appropriate, a high energy line rupture within the secondary containment.

b. Identification of primary containment leakage paths that bypass the secondary containment including the design provisions for periodically leak testing the bypass leakage paths.

c. The proposed technical specifications pertaining to the functional capability testing of the secondary containment system and the combined leak rate for bypass leakage paths.

d. Analyses of the effect of openings in the secondary containment on the capability of the depressurization and filtration system to accomplish its design objective of establishing a negative pressure in a prescribed time.

Response:

a. An assessment of the environmental response of the containment enclosure building (secondary containment) to a LOCA within the primary containment shows the effect to be negligible. The elevated pressure and temperature from a LOCA cause expansion of the primary containment. The enclosure building, in turn, expands similarly since it is structurally supported from the primary containment. The enclosure panel joints are designed to accommodate the expansion without losing building integrity. Thus, the containment enclosure building pressure will neither increase nor will its integrity be adversely affected from a LOCA within the primary containment. For a further discussion of the containment enclosure building, refer to FSAR Section 3.8.4.

   The effect of heat transfer from the primary containment to the containment enclosure building is negligible. It would require hours to produce a temperature change at the exterior of the primary containment structure from a LOCA because the concrete thermally isolates the outside from the effects inside.

b. Primary containment bypass leakage paths are identified in FSAR Section 6.2, Table 6.2-65. Testing provisions are discussed in FSAR Section 6.2.4.

c. Proposed technical specifications will be submitted 1 year prior to fuel loading.

d. No opening in the enclosure building is postulated in designing the SLCRS. An in-leakage rate of .0195 cfm/square foot of the enclosure building area is considered in the design, as indicated in the response to NRC Question 480.24.

NRC Letter: January 16, 1984

Question No. Q480.28 (Section 6.2.4)

According to Section 6.2.4.11, the containment pressure setpoint which initiates the Phase A containment isolation signal, is set at 1.5 psig. Justify that the containment pressure setpoint is the minimum compatible with normal operating conditions. The guidelines contained in Section II.E.4.2 of NUREG-0737 should be used in the justification.

Response:

Millstone 3 has a subatmospheric containment, and under certain plant operating conditions, the containment will reach atmospheric conditions. In these cases, the containment isolation Phase A signal is neither necessary nor desired.

Subatmospheric containments use a 1.5 psi margin above atmospheric pressure to provide adequate protection against inadvertent containment isolation. This margin is necessary to allow instrument inaccuracies and slight pressure fluctuations due to containment temperature variation.

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Question No. Q480.29 (Section 6.2.4)

According to Section 6.2.4.14, the containment isolation valves outside of containment are located no more than 10 feet from the containment wherever practical. According to Table 6.2-65, the isolation valves for the two auxiliary feedwater lines and the hydrogen recombiner suction and discharge lines are located 50, 90, 20, and 17 feet respectively from the containment. Justify the spacing between the isolation valves and the containment for these lines.

Response:

Refer to revised FSAR Table 6.2-65 for the response to this question.

These isolation valves for the two auxiliary feedwater lines and the hydrogen recombiner suction and discharge lines are relocated to 11, 10, 19, and 12 feet respectively from the containment. The instances which exceed the 10 foot guideline are a result of building arrangement considerations such as requirements for fittings, interferences with other valves, and stress considerations.

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NRC Letter: January 16, 1984

Question No. Q480.30 (Section 6.2.4)

For all branch lines located between the containment and outside or inside isolation valve, identify the containment isolation provisions for these branch lines and dicuss how General Design Criteria 54-57 are met.

Response:

Refer to revised FSAR Section 6.2.4.2 for the response to this question.

monitor discharge, and reactor coolant pump seal water return lines, all normally open containment isolation valves inside the reactor containment are air-operated valves, solenoid valves, or check valves. The fail closed feature of these valves is not affected by the most severe post-DBA environmental conditions. The containment isolation valves for the instrument air, containment atmosphere monitor discharge, and reactor coolant pump seal water return lines are motor-operated valves which are designed to operate under post-DBA conditions.

Closed systems used as one of the isolation barriers inside or outside the containment satisfy the following requirements:

1. The systems do not communicate with either the reactor coolant system or the containment atmosphere.

2. The systems are protected against missiles, pipe whip and jet impingement

3. The systems are designated Seismic Category I.

4. The systems are classified Safety Class 2 (Reactor plant component cooling system is designated Safety Class 3).

5. The systems are designed to withstand temperatures at least equal to the containment design temperature.

6. The systems are designed to withstand the external pressure from the containment structural acceptance test.

7. The systems are designed to withstand the loss of coolant accident transient and environment.

Valves used for containment isolation barriers are designed, constructed, and installed in accordance with Safety Class 2 and Seismic Category I Requirements. The design pressure of containment isolation valves is equal to, or greater than, the design pressure of the reactor containment. Containment isolation valve type is selected on the basis of fluid system requirements (pressure drops, radioactivity, etc.), seat leaktightness, and the standard industry practices for the applicable valve size. Containment isolation valve procurement specifications require strict seat and packing leaktightness tests in addition to the code requirements. Branch
480.30 lines located between the containment and the outside or inside isolation valve meet the same containment isolation criteria as the main line.

Design provisions are made to ensure the integrity of containment isolation valves and connecting piping under dynamic forces resulting from inadvertent closure. Details of these provisions are given in Section 3.7.3.

The containment isolation system design provides mechanical and electrical redundancy. The isolation valve arrangement ensures

containment integrity, assuming the occurrence of a single failure, by providing at least two barriers between the atmosphere outside the

480.31 (6.2.6) According to Section 6.2.6, isolation valves for the Secondary Main Feedwater and Main Steam lines will not be Type C tested. In order to meet the requirements of Appendix J, justify not including these isolation valves in your test procedures:

Response:

These lines are on the secondary side of the steam generators and are considered closed loops within containment. Operating procedures will maintain steam generator secondary side water level and pressure (to be greater than post-accident containment pressure) during recovery from a design basis accident (DBA). This will assure that leakage through the closed loops can only be into containment during a DBA.

The closed loops within containment and the water level and pressure control procedures constitute two barriers against release of radionuclides. Therefore, containment isolation valves in these lines need not be tested to meet 10 CFR 50 Appendix J objectives.

480.32 (6.2.6) With regard to leak rate testing of butterfly valves having resilient seals, such as the containment isolation valves in the purge/vent system, it is our position that passive valves (i.e., valves administratively controlled closed during plant operating Mode 1 through 4) be leak tested at least once every six months. Discuss your plans for complying with this position.

Response:

Passive, normally closed, butterfly valves used as containment isolation valves in systems connected to the containment atmosphere will not be leak tested at six month intervals. Section III.D.3 of 10 CFR 50 Appendix J only requires that such valves be tested ". . .during each reactor shutdown for refueling, but in no case at intervals greater than 2 years." These valves will be tested in accordance with 10 CFR 50 Appendix J.

480.33 (6.2.6) To fully satisfy Item II.E.4.2 of NUREG-0737, commit to verifying every 31 days that the containment purge valves are closed, by checking the purge valves themselves or by checking the purge valves themselves or by checking the purge valve position lights in the control room.

Response:

The containment purge valves will be verified to be closed at least once every 31 days (per technical specifications). This verification will be accomplished by checking the purge valve position lights in the control room.

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Question No. Q480.34 (Section 6.2.6)

According to Section 6.2.2.3, the isolation valves for the seal water injection lines serving the reactor coolant pumps are not Type C tested because the lines would be continually pressurized following an accident. Discuss how this would be accomplished assuming a single active failure or the termination of seal water injection.

Response:

The response to this question will be submitted at a later date.

NRC Letter: January 16, 1984

Question No. Q480.35

Provide a description of the instrumentation used to monitor containment pressure, temperature, and containment hydrogen concentration. In your description of containment instrumentation, provide the information requested in Section IIF.1 of NUREG-0737.

Response:

Four redundant capacitance-type Rosemount Model 1153A absolute pressure transmitters, range 0-60 psia, are provided on Millstone 3. Two additional redundant Rosemount transmitters (same model and type), range 0-200 psia, are provided for extended range measurement (three times containment design pressure). All six transmitters have indicators on the main control board and two of the four channels of the normal range transmitters are continuously recorded, while one of the two extended range channels is recorded. Accuracy of the transmitter is ±0.25 percent and response time is 0.2 seconds for transmitter output to reach 63.2 percent of the final value for a step change. All transmitters and channels are designed and qualified in accordance with Regulatory Guide 1.97, paragraph 1.3.1.

Two redundant 200 ohm platinum resistance temperature detectors are provided with indication on the main control board. One channel is continuously recorded. RTD calibration is 0-400°F, accuracy is ±1.0 percent, and response time is 5 seconds for sensor output to reach 63.2 percent of the final value for a step change. Both channels are designed and qualified in accordance with Regulatory Guide 1.97, Paragraph 1.3.1.

The Millstone 3 containment hydrogen monitoring system is designed as Category I (Class 1E) with dual redundant trains (Train A and Train B). This system addresses the requirements set forth in NUREG 0737, Item IIF.1(6).

Each train contains stand-alone analyzer and control cabinets which analyzes, monitors, alarms and trends containment hydrogen concentration.

The containment hydrogen monitoring system will sample hydrogen sources on an automatic/manual basis selectable from the control cabinet located in the hydrogen recombiner building control area.

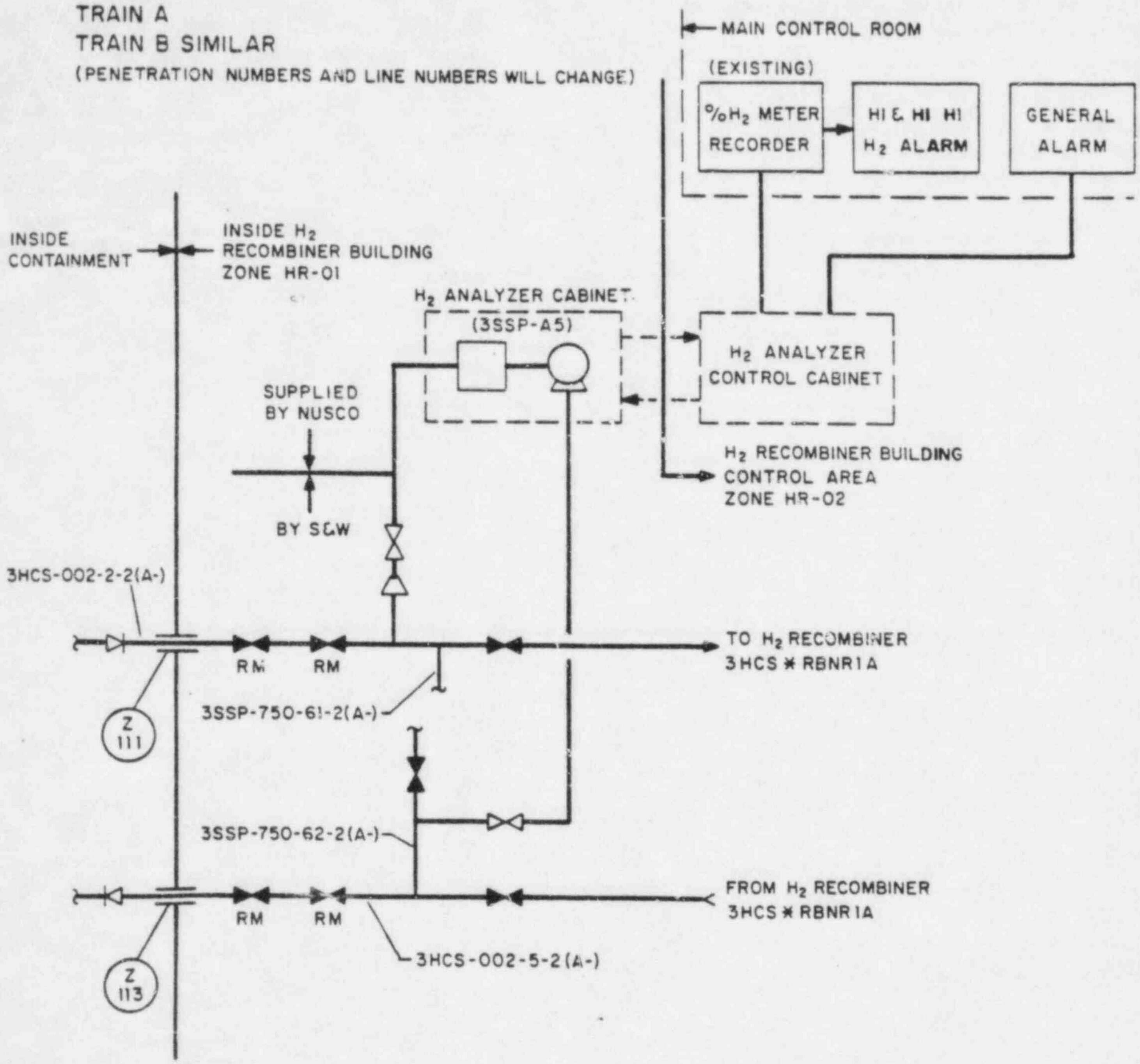
Withdrawal of the samples from existing hydrogen recombiner lines, measurement of hydrogen concentration, and return of the total sample to the containment are the basic functional assignments of the hydrogen analyzer cabinet.

The hydrogen analyzer control cabinet provides the data acquisition, computation, and automatic/manual control of all analyzer functions.

Hydrogen concentration will be measured and converted to an analog signal (0-10 percent H_2) for display on the digital panel meter, mounted on the control cabinet.

The system will have analog output for display (two meters), recording (Train A only), and alarming in the main control board. Input is also provided to the plant computer.

Figure Q480.35-1 shows the location and general arrangements for the hydrogen monitoring system.



LEGEND:

RM = REMOTE MANUAL OPERATION

NOTE

1. H_2 ANALYZER CABINET - 72"H x 24"D x 31"W - 750 LBS.
2. H_2 ANALYZER CONTROL CABINET - 70"H x 30"D x 26"W - 465 LBS.

FIGURE Q480.35-1

CONTAINMENT HYDROGEN MONITORING SYSTEM

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

480.36 (6.2.6) Discuss whether or not the containment liner weld channels will be vented during the Type A test; if not, provide justification.

Response:

The containment liner channels will not be vented during type A testing to prevent the entry of moisture into the channels. Moisture in the channels could eventually lead to localized liner corrosion. Vacuum drying to prevent such corrosion would be costly and difficult to verify.

The containment liner is double walled at the channels. All channels were leak tightness checked prior to closure (i.e. installation of threaded plugs). Periodic type A testing, without channel venting, will verify containment liner integrity in accordance with 10 CFR 50 Appendix J.