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

November 1, 1982

Docket No. 50-245
B10571

Director of Nuclear Reactor Regulation
Attn: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: (1) R. A. Clark letter to W. G. Counsil, dated June 30, 1982.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
NUREG-0737 Item II.B.3 Post Accident Sampling System

In Reference (1) Northeast Nuclear Energy Company (NNECO) was requested to submit documentation on how each criterion of NUREG-0737 Item II.B.3 has been satisfied.

The Attachments to this letter document NNECO's response to Reference (1) and constitutes our resolution of NUREG-0737 Item II.B.3. We plan to have the necessary modifications completed and procedures in place by December 31, 1982. We remain available should the Staff require any clarification of this issue.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

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Docket No. 50-245

ATTACHMENT I

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1
POST ACCIDENT SAMPLING SYSTEM
NUREG-0737 ITEM II.B.3

NOVEMBER, 1982

Criterion: (1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Clarification: Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of off-site power (i.e. designate an alternative backup power source, not necessarily the vital (Class IE) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

Response: Millstone Unit No. 1 has the capability of obtaining reactor coolant and containment atmosphere samples as recommended by NUREG-0737. Attachment II provides an overall description of the Millstone Unit No. 1 Post Accident Sampling System (PASS). The entire sampling operation including preparation, sample recirculation, sample isolation, purge of the system piping, sample retrieval, transport to the chemistry laboratory and analysis for both reactor coolant and containment atmosphere samples can be completed in three hours. This has been verified during operational testing and training exercises. The chemistry laboratory and the sample station are located approximately 100 feet apart at the same elevation, facilitating rapid sample transport.

PASS uses 110 volt AC, 15 amp power from a normal lighting circuit which is considered a reliable power source. Since neither Regulatory Guide 1.97 nor NUREG-0737 require an alternate power source for the PASS, and NNECO concluded that no unique power source provisions were necessary, one was not included in the design.

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

(a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Clarification: (2) (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non-volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

(b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

(c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

(d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Response: 2(a) The chemistry laboratory has a Nuclear Data 6620 Gamma Spectrometer Counting System with the capability to identify and quantify gamma emitting nuclides. This system is interfaced with two Ge(Li) crystals (15% efficient) and one intrinsic (Ge) crystal (19% efficient).

These crystals are calibrated with fourteen different geometries for counting liquid, gaseous particulate and charcoal filter samples.

A procedure is being developed to estimate core damage based on primary coolant sample results and other measured parameters. The Rogovin Report and other reports written since the TMI accident are being considered in the development of the procedure. The procedure will address evaluation of short and long lived volatile radionuclide levels.

It is expected that this procedure will be written, approved and implemented by November 30, 1982.

2(b) The capability to obtain and transport a grab sample is described in Attachment II. The chemistry laboratory has a Perkin-Elmer Sigma 3B Gas Chromatograph for analyzing hydrogen. This instrument has been modified to accept post accident gaseous samples.

2(c) Regulatory Guide. 1.97 Rev. 2 sampling and analysis capabilities are discussed below:

Primary Coolant and Sump

- o Gross Activity - see item 2(a)
- o Gamma Spectrum - see item 2(a)
- o Boron Content - A Perkin-Elmer I.C.P.-AA Model 5000 Photometer has been modified to accept post accident samples in range of 1 ppm. to 3000 ppm.
- o Chloride Content - An E.G.G. Polographic Analyzer Model 384-4, modified to accept post accident samples in the range of 0.1 ppm to 20 ppm, is scheduled to be operational by January 1, 1983.
- o Total Gas or Dissolved Hydrogen - see Attachment II
- o Dissolved Oxygen - The original design criteria of NUREG - 0578 did not call for the monitoring of dissolved oxygen, therefore this capability was not incorporated in our design.
- o pH - see Attachment II

<u>Containment Air</u>

- o Hydrogen Content - see item 2(b)
- o Oxygen Content - see item 2(b)
- o Gamma Spectrum - see item 2(a)

2(d) The major design concept of the PASS is based on taking a manual grab sample. The pH probe is the only in-line analyzing instrument and is discussed in the general system description in Attachment II.

Criterion: (3) Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system (RWCUS)) to be placed in operation in order to use the sampling system.

Clarification: System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Response: The Post Accident Sampling System (PASS), Reactor Coolant portion, has the capability to sample the influent branches from the Low Pressure Coolant Injection (LPCI) crosstie line, the Shutdown Cooling System supply off recirculation loop 'A' and the Sampling System supply off recirculation loop 'A'. The sampling and recirculation operations would not require an isolated auxiliary system. In a post accident situation, any one to all three of the above mentioned systems could be in operation during an accident and sampling as well as recirculation would only require opening the PASS isolation valves.

The containment atmosphere PASS ties into the Hydrogen Analyzer sample lines but does not require it to be operating for sampling and recirculation.

Criterion: (4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification: Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is 0.1 ppm by measurement of a dissolved hydrogen residual of 10cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with ALARA, direct monitoring for dissolved oxygen is recommended.

Response: The method for relating total dissolved gases is presented in Attachment II. No provision is made for the measurement of dissolved oxygen since it was not specified in the original design criteria of NUREG-0578 nor has it been required.

Criterion: (5) The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification: BWR's on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers (e.g. shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform a chloride analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at 0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Response: The chemistry lab will have the capability of measuring chloride concentration on undiluted samples down to 0.1 ppm. See item 2(c). It is our position that there is no need to save undiluted samples and have not made any provision for measuring dissolved oxygen as this is not a requirement.

Criterion: (6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees)).

Clarification: Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted man rem exposures based on person-motion for sampling, transport and analysis of all required parameters.

Response: Tables 1 and 2 present the predicted whole body and extremity exposures required to obtain and analyze primary coolant and containment atmosphere samples, respectively. The source terms assumed are delineated in NUREG-0737. As has been demonstrated, all doses are well within the 5 rem whole body, 75 rem extremities requirements of GDC 19.

Table 1

Dose Required To Obtain and Analyze Reactor Coolant Sample

		Whole Body Dose (mrem)	Extremity Dose** (mrem)
1.	Sample Collection		
	a. Area Dose - Control Panel	500	500
	b. Area Dose - Sample Panel	150	150
	c. Sample Dose - Sample Panel	18	178
	d. Transit Dose - Area & Sample	50	50
	Total Sample Collection	718	878
2.	Sample Analysis		
	a. Area Dose - Lab	NEG.*	NEG.
	b. Area Dose - Count Room	NEG.	NEG.
	c. Dose Due To Sample		
	(1) Dilution	16	890
	(2) Ge Li	NEG.	10
	(3) Boron	10	620
	Total - Sample Analysis	26	1520
	Total - Collection & Analysis	744	2398

*NEG = Negligible - less than 10 mrem.
** Includes whole body dose.

Table 2

Dose Required To Obtain and Analyze Atmosphere Sample

	Whole Body Dose (mrem)	Extremity Dose** (mrem)
1. Sample Collection		
a. Area Dose - Control Panel	500	500
b. Area Dose - Sample Panel	150	150
c. Sample Dose - Sample Panel	31	352
d. Transit Dose - Area & Sample	50	50
Total - Sample Collection	731	1052
2. Sample Analysis		
a. Area Dose Lab	NEG*	NEG
b. Area Dose - Count Room	NEG	NEG
c. Dose due to Sample		
(1) Gas Chromatography	100	200
(2) Ge Li	NEG	20
Total Sample Analysis	100	220
Total - Collection Analysis	831	1272

*NEG = Negligible - less than 10 mrem
** = Includes Whole Body Dose

Criterion: (7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants.)

Clarification: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Response: Millstone Unit No. 1 has the capability to measure boron concentration. See item 2(c).

Criterion: (8) If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Response: The PASS has the capability to obtain both diluted and undiluted backup samples as described in Attachment II. Backup analysis will be performed at the Haddam Neck Plant which has capabilities similar to those of the Millstone chemistry laboratory. The Haddam Neck Plant is located approximately 40 miles from the Millstone Nuclear Power Station.

Criterion: (9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 uCi/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification (9)(a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(9)(b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to ensure an accuracy within a factor of 2.

Response: 9(a) Samples may have any concentration from the normal operating range (typically 0.1 uCi/g) up to the calculated accident concentration of approximately 4 Ci/g.

Samples that are too radioactive to count will be diluted. The expected dose from performing a dilution of the highest activity sample is given in item 6. Dilution will be manual and the criteria for total dilution factor will be based on reducing sample activity to a level that can be counted. Thus the entire range of expected coolant activity can be analyzed.

9(b) Doses in the counting room were calculated assuming the source terms specified in NUREG-0737. Since the counting room is adjacent to the MP1 Reactor Building, the controlling accident is a LOCA at MP1. Due to distance and intervening shielding, an accident at MP2 would result in significantly lower dose rates. The MP1 LOCA sources considered include the piping in the MP1 Reactor building which could circulate primary coolant, airborne activity in the drywell, and airborne activity in the reactor building. Since samples will not be prepared or stored in the counting room, they were not considered in dose calculation. Credit was taken for decay in that it was assumed that sample analysis will not be required until 2 hours post shutdown. The calculated dose rate at this time in the counting room was 0.3 MR/hr.

This dose rate is insignificant in comparison with the activity level of the sample and hence will have no effect on the ability to achieve a factor of two accuracy.

Criterion: (10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Clarification: The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

In general this analysis should be accurate within $\pm$ 5% of the measured value (i.e. at 6,000 ppm B^{-} the tolerance is $\pm$ 300 ppm while at 1,000 ppm the tolerance band should remain at $\pm$ 50ppm.

- Hydrogen or Total Gas: monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of $\pm$ 10% is desirable between 50 and 2000 cc/kg but $\pm$ 20% can be acceptable. For concentration below 50 cc/kg the tolerance remains at $\pm$ 5.0 cc/kg.

- Oxygen: monitored to assess coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen the analysis should be accurate $\pm$ within 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains $\pm$ 0.05 ppm.

- pH: measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within $\pm$ 0.3 pH units. For all other ranges $\pm$ 0.5 pH unit is acceptable.

To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Consituient	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs^+	250	Cesium Nitrate
Ba^{+2}	10	Barium Nitrate
La^{+3}	5	Lanthanum Chloride
Ce^{+4}	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li^+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^-	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2.) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

Response: It is our position that the accuracies recommended in the clarification are achievable during normal conditions but not during post accident conditions. Accuracies and ranges that we conclude are appropriate based on our experience are listed below.

- o Gross activity, gamma spectrum: A reasonable accuracy is a factor of 10.
- o Boron: $\pm$ 50 ppm below 1000 ppm and $\pm$ 5% above
- o Chloride: $\pm$ 10% for concentrations between 0.5and 20.0 ppm and $\pm$ 0.1 ppm below 0.5 ppm.
- o Total gas: $\pm$ 20% between 50 and 2000 cc/kg and $\pm$ 10 cc/kg from 50 to 30cc/kg, the lower level of detection for our system.
- o Oxygen: The ability to measure oxygen is not a requirement and we have not made provision for it.
- o pH: $\pm$ 0.3 pH units between 5 and 9, $\pm$ 0.5 pH units for other ranges.

The standard test matrix is a new criterion that was not specified in NUREG-0578. The analytical instrumentation selected for post accident sampling (i.e. ICP-AA 5000, Gas Chromatograph and EGG Polographic Analyzer model 384-4) was chosen for its ability to operate in the post accident sampling environment.

Semi-annual training is in excess of that required for senior reactor operators. Annual training is considered adequate. Technical Specifications regarding the PASS will be proposed subsequent to receipt of the model Technical Specifications.

Criterion: (11) In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(11)(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

Response: (11)(a) During design of the post accident sampling and analysis capabilities, consideration was given to the following items.

Reactor Coolant Portion

The reactor coolant portion of the PASS is provided with a flush module and pump. This flushes the radioactive sample and return lines with demineralized water at approximately one gallon per minute (350PSIG).

The reactor coolant portion has the capability to sample at three points within primary coolant boundary that are representative of core condition. These three sample points are as follows: recirculation loop 'A' pump suction, recirculation loop 'A' pump discharge and the suppression pool (torus). A sample taken during recirculation pump operation is representative of the core condition around the shroud. The recirculation pump provides adequate mixing of primary coolant by the upward flow of cooling water into the core and the downward flow back to the pump suction. In the improbable case of recirculation loop pipe break a representative sample could be obtained after Low Pressure Coolant Injection System (LPCI) initiation. LPCI takes a suction on the torus and injects into the core. The leaking water from the pipe break would flow into the drywell and then overflow back to the torus. Therefore sufficient mixing would occur in the torus after a run time on the LPCI system and a sample from this system would be indicative of the conditions existing around the core shroud area.

If a sample line or return line on the reactor coolant portion should break, PASS isolation valves, located immediately off the primary coolant boundry, serve to stop potential leakage. The redundant sample and return lines would be available and allow continued sampling. These remote operated isolation valves are installed to ASME Section III Class 2 requirements and are seismically and environmentally qualified.

Containment Air Portion

Containment air inlet and outlet lines are heat traced and insulated to prevent condensation of any vapor in the atmospheric sample. This ensures a representative sample for analysis.

Containment air sample and return lines are provided with a nitrogen purge. This purge system supplies nitrogen at approximately 4-8 standard liters per minute (100 psig).

This portion of the PASS samples containment atmosphere at 2 locations in the drywell, an upper and lower sample point. This allows obtaining a representative sample should primary containment atmosphere stratify.

Isolation valves are provided on the containment air sample and return line. If one of the lines should break, these isolation valves, located immediately off the drywell penetration, could be closed to prevent leakage.

PASS piping is sized to maintain turbulent flow thereby minimizing crud buildup and plate-out of radioactive products. Additionally the sample and return lines were designed for the shortest tubing runs possible.

All PASS isolation valves are normally in the closed position. They are provided with a key lock type switch and are maintained locked closed during normal operation. Therefore only operator intervention can open them and a containment isolation signal is not necessary.

(11)(b) Radioactive airborne contamination control is provided by blowers which maintain a slight negative pressure in the sample modules. These blowers exhaust into the existing plant ventilation which exhaust to the atmosphere via the stack. Normally the ventilation systems exhaust directly to atmosphere with radiation monitoring detection. . NNECO has concluded that leakage from the sampling system will be insignificant compared to leakage from other systems, and, thus filtered ventilation of the sampling station should not be required. This conclusion was stated in our December 15, 1980 letter to D. G. Eisenhut.

Docket No. 50-245

ATTACHMENT II

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1
POST ACCIDENT SAMPLING SYSTEM
NUREG-0737 ITEM II.B.3

NOVEMBER, 1982

MP1 PASS SYSTEM DESCRIPTION

1.0 INTRODUCTION

The Millstone Unit No. 1 PASS has the capability to obtain samples of reactor coolant and containment atmosphere under accident conditions in accordance with the requirements of NUREG 0578 and clarifications provided by NUREG 0737. The PASS is comprised of two independent units, designated Reactor Coolant PASS and Containment Air PASS. The reactor coolant PASS is designed to obtain representative samples of reactor coolant or liquid from the containment. The containment air PASS is designed to obtain a containment air sample. Once these samples are obtained, radiological and chemical analyses can be performed on-site or the samples can be transported off-site for analysis. Samples can be obtained within one hour and analysis can be performed within two hours after a decision is made to take a sample.

2.0 REACTOR COOLANT PASS

2.1 Equipment Purpose and Description

The reactor coolant PASS is a dual module unit consisting of one sample module and one remote operating module. Samples are trapped within the sample module. The equipment within the sample module is operated remotely via the remote operating module. The motive force for obtaining RC samples is the differential pressure between the primary plant and the collection area to which sample effluent is directed. Two sampling modes may be chosen, depending upon whether the sample is to be shipped off-site for analysis or analyzed on-site. Samples to be analyzed off-site are collected in a 2 ml shielded, removable sample chamber within the sample module. Samples to be analyzed on-site are collected in shielded containers within the sample module.

2.1.1 Sample Module

This module contains the valves and components required to physically collect the sample. All components are located within a stainless steel cabinet measuring approximately 22" wide by 24" deep, by 36" high, which sits on a 2' high stand. An exhaust blower is built into the top of the cabinet and discharges into the plant radioactive exhaust ventilation system. Doors are provided on the cabinet for access to remove samples and to perform maintenance. The module is located in the Millstone Unit No. 1 turbine building against the reactor building wall. At this location levels of radiation created at the module during the purge of RC through the sample lines will not result in significant exposure to the operator at the remote module or to other individuals.

Sample influent and effluent lines are connected to the sample module. Influent samples are taken from three points at Millstone Unit No. 1. RC samples can be taken via the reactor cleanup system, or the shutdown cooling system, and containment liquid samples can be obtained from the pressure suppression pool (torus) via the low pressure coolant injection (LPCI) system. Influent samples pass through sample coolers prior to being delivered to the sample module.

Effluent lines connected to the module are directed to either the torus or back to the RC recirculating system via the shutdown cooling system. Return of the effluent via the shutdown cooling system would be chosen if the pressure suppression pool has not been contaminated and it is desirable to confine the radioactive effluent to the primary plant.

2.1.2 Remote Operating Module

This module contains the sample system mimic board, electrical controls, and instrumentation read-out necessary to remotely operate the sample module. The remote operating panel is located outside the chemistry laboratory in the area which will have low radiation levels during an accident, approximately 100' from the sample module. The remote operating module is connected to the sample module through electrical umbilical cables which carry power and instrumentation indication lines. Nitrogen gas supply lines, used to operate valves and purge the radioactive gas sample after sampling is completed, are hard-piped to the sample and remote modules. The face of the module is normally protected by a lockable closure to prevent damage and unauthorized operation. 110 volt AC, 15 amp power is used to operate the remote operating module.

2.1.3 Reactor Coolant Auxiliary Valve Operating Panel (RCAVOP)

This panel contains switches and electrical controls to operate the sample system isolation valves and the system flush valves. The RCAVOP is located adjacent to the remote operating module.

2.1.4 Deionized Water Flushing Module

Deionized water flushing module is a modular unit designed to provide deionized water flushing capability at approximately one gallon per minute and up to 375 psig. The module is located adjacent to the remote operating module.

The flushing system is a self-contained system consisting of a water storage tank, positive displacement pump, and controls to operate the equipment. The equipment is mounted on a bedplate to form a modular unit.

2.1.5 Design Parameters

Design Pressure:	400 psig
Design Temperature:	Ambient
Pump Discharge Pressure:	Variable: Set at 375 psig
Pump Flow Rate:	Variable: Set at one gpm
Electric Power:	110 Volt AC, 15 Amp

2.2 Sampling Capability

2.2.1 Grab Sample of Reactor Coolant

A two milliliter sample, containing either pressurized or unpressurized liquid, can be obtained using the removable shielded sample chamber. This grab sample would normally be utilizied for off-site analysis.

2.2.2 In-Line Reactor Coolant Sample

The required volume of sample liquid is extracted from a shielded five ml sample chamber via a septum and syringe. This sample is normally used for on-site analysis at the chemistry laboratory.

2.2.3 Depressurized and Diluted Reactor Coolant Gases

The required volume of sample gas is extracted from a shielded five ml sample chamber via a septum and syringe. This sample is normally used for on-site analysis at the chemistry laboratory.

2.3 Analysis Capability

The Millstone Unit No. 1 reactor coolant PASS has the following analysis capability.

2.3.1 In-Line Analysis

Total dissolved gas (2000cc/Kg @ STP).

pH (0-14).

Indication of the in-line analysis is provided at the remote operating panel.

2.3.2 Laboratory Analysis

The following analyses will be performed at the Millstone chemistry laboratory or at the Connecticut Yankee laboratory if the Millstone laboratory is inaccessible.

2.3.2.1 Boron

A 100 μl sample of reactor coolant is extracted from the sample system via septum and syringe and injected into a sealed container of deionized water to produce a 1500:1 dilution. The sample is then transported in specially designed containers to the laboratory for analysis.

2.3.2.2 Radionuclide Gamma Spectrum and Gross Radioactivity

A 10 ml sample of diluted reactor coolant gas is extracted from the sample system via septum and syringe, injected into a sealed sample container, and transported in specially designed containers to the laboratory for analysis.

A portion of the liquid sample prepared for boron analysis is used in the laboratory to analyze reactor coolant for gamma spectrum and gross radioactivity.

2.3.2.3 Chlorides

Later.

2.4 Design Features

Both operating and equipment failure modes are analyzed to maintain exposure to ALARA. Personnel radiation exposure is minimized through the use of remote control operation, flushing techniques, and minimal sample volume and shielding.

Radiation exposure to the operator taking the sample is estimated to be well below the exposure limits defined by 10 CFR 50, Appendix A, GDC-19.

The PASS system piping downstream of the sample coolers is designed for 2500 psig and 165°F. The inherent design of pH probes limits functional usage to pressure of 250 psig. The pH probe is therefore isolated during high pressure evolutions to protect the probe internals. The pH probe outer housing is designed to withstand system design pressure of 2500 psig in the event of inadvertent overpressurization to the probe internals.

All fluid boundary materials are of either 300 series stainless steel or Inconel.

The normal fail position of each solenoid valve was selected such that failures will still allow flushing the system to minimize radiation levels.

Solenoid valves are equipped with positive position indication at the remote operating panel.

Radioactive airborne contamination control is provided by a blower which maintains capture velocity into the sample module. This blower exhausts into existing plant ventilation.

Commercially available components are utilized to the maximum extent possible and have been selected based upon a reputation for high quality. Swagelok fittings are utilized wherever possible to be consistent with existing utility sample system components.

Leak rate test capability is provided for PASS containment isolation valves by installed test connections and isolation valves.

PASS piping is sized to maintain turbulent flow thereby minimizing crud buildup and plate-out of radioactive products.

Three sample points and two effluent return paths are available on the reactor coolant PASS providing operational redundancy.

Reactor coolant PASS solenoid isolation valves are operated from the control room control board. Therefore, the control room operators are responsible for making the PASS available to chemistry personnel by operating required containment isolation valves and isolating the PASS as conditions require. Final responsibility for system availability is then left with the control room, enhancing plant safety.

The valves required to be operated during the sampling operation are operated by chemistry personnel at the RCAVOP and remote operating module. Valve position indication is also provided for critical isolation valves at the control room to provide system status information to the control room operators.

Solenoid valves isolating the reactor coolant PASS from the interfacing systems are built to the ASME Code, Section III requirements, and are fully qualified for postulated accident conditions.

The reactor coolant PASS sample module, remote operating module, and demineralized water flush modules are identical to units installed at Millstone Units No. 2 and No. 3 and Connecticut Yankee, providing standardization and interchangeability. Operation of the systems is identical except where plant-specific conditions dictate.

3.0 CONTAINMENT AIR POST ACCIDENT SAMPLE SYSTEM

3.1 Equipment Purpose and Description

The containment air PASS has the capability of collecting a sample of containment air as required by NUREG 0578. The PASS is a dual module unit consisting of one sample module and one remote operating panel. The motive force for obtaining a sample is a sample pump which draws the sample from the oxygen/hydrogen analyzer system through the sample module, and back to the oxygen/hydrogen analyzer system. Inlet and outlet lines will be heat-traced and insulated to prevent condensation of any vapor in the atmospheric sample.

3.1.1 Sample Module

This module contains the valves and components required to physically collect a 10 ml sample of containment air. The equipment is housed within a wall-mounted cabinet. An exhaust blower is built into the top of the cabinet and discharges into the plant radioactive ventilation exhaust system. A door is provided on the cabinet for access to remove samples and to perform maintenance. Influent and effluent connections are provided for connection to the sample system piping.

3.1.2 Remote Operating Module

This module contains the sample system mimic panel and slave valves required to remotely operate the sample module. The face of the module is covered by a hinged, lockable cover to prevent damage to the controls and unauthorized use of the equipment. The module is located adjacent to the reactor coolant operating module outside the chemistry laboratory where radiation levels during an accident are low. The remote operating module is connected to the sample module through "umbilical" cables and piping. A control switch is provided to operate a remote valve to purge the sample lines with nitrogen following sample collection and isolation.

3.2 Containment Air Samples

A 10 ml sample of containment air is isolated in shielded sample chamber. Samples are withdrawn from the chamber via septum using a syringe, injected into a sealed container, and transported to the laboratory for subsequent analysis.

3.3 Design Features

Both operating and failure modes have been analyzed to maintain exposure to ALARA. Personnel radiation exposure is minimized through the use of remote control operation, flushing techniques, minimal sample volume, and shielding.

Radiation exposure to the operator taking the sample is estimated to be well below the exposure limit defined by 10 CFR 50, Appendix A, GDC-19.

Materials in contact with the containment air sample are of 300 series stainless steel and have been selected to ensure system integrity up to a pressure of 100 psig at 300°F.

Heat generated by flow of hot fluid through the sample module piping is dissipated by a blower which provides capture velocity air flow into the sample module and exhausts to the plant radioactive ventilation system.

Commercially available components have been utilized to the maximum extent possible and are selected based upon a reputation for high quality. Swagelok fittings have been utilized wherever possible to be consistent with existing plant equipment.

Isolation valves and breakdown connections are provided for solenoid valves and other equipment for maintenance considerations.

The containment air PASS piping and sample pump are sized to maintain turbulent flow, thereby minimizing crud buildup and plate-out of radioactive products.

Two sample paths and two return paths are available on the containment air PASS providing operational redundancy.

All valves are operated from the remote operating module or containment air auxiliary valve operating panel (CAAVOP). Isolation from containment is provided by valves in the oxygen/hydrogen analyzer system. These isolation valves are controlled from the control room.

The containment air PASS sample module and remote operating panel are identical to units installed at Millstone Units No. 2 and No. 3 and Connecticut Yankee, providing standardization and interchangeability. Operation of the systems is identical except where plant-specific conditions dictate.

Heat tracing is installed in the containment air PASS piping to ensure a representative sample.

Valves isolating the containment air PASS from the oxygen/hydrogen analyzer system will have valve position indication on the main control board thereby continuously advising operators of the PASS system status.

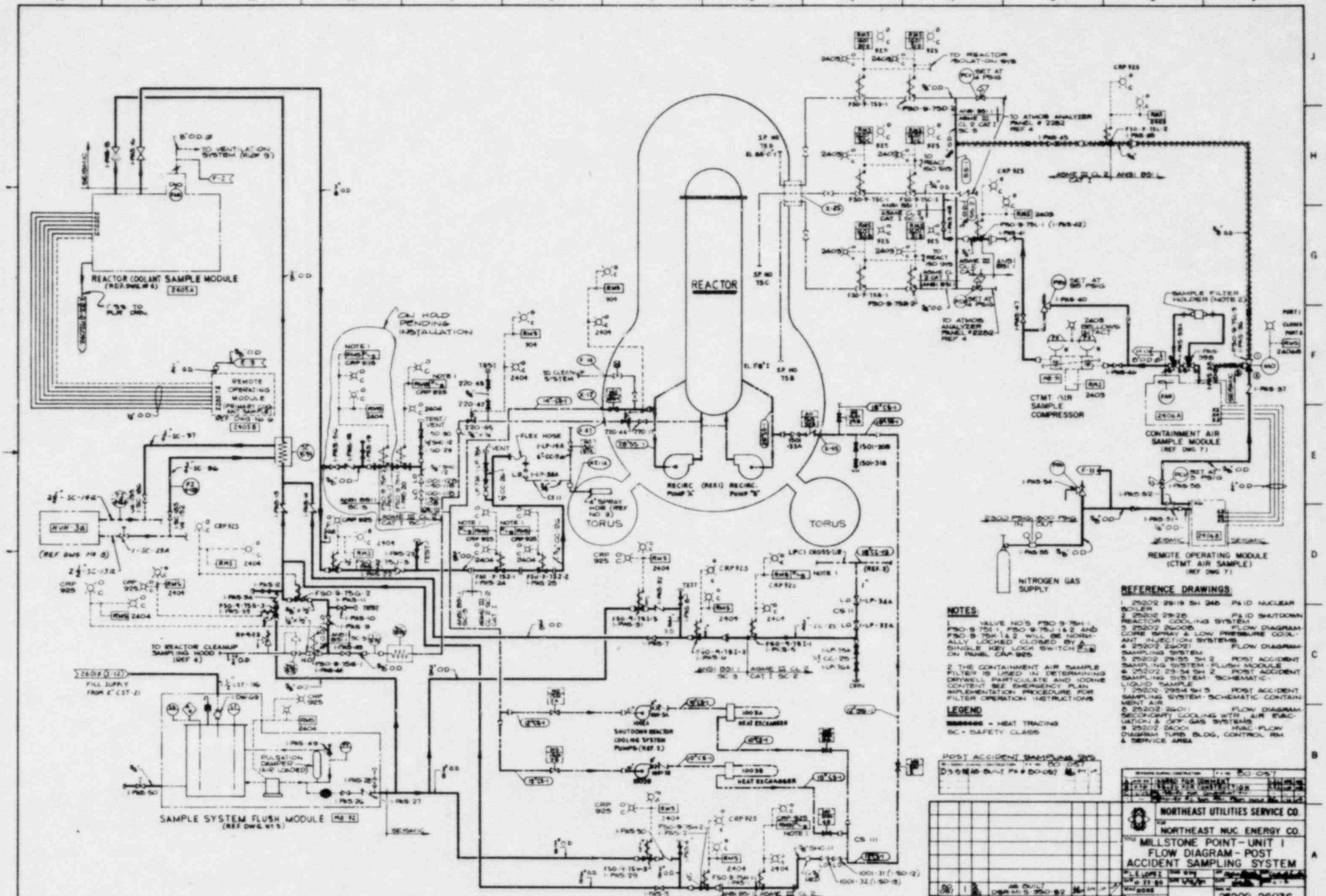
4.0 TEST

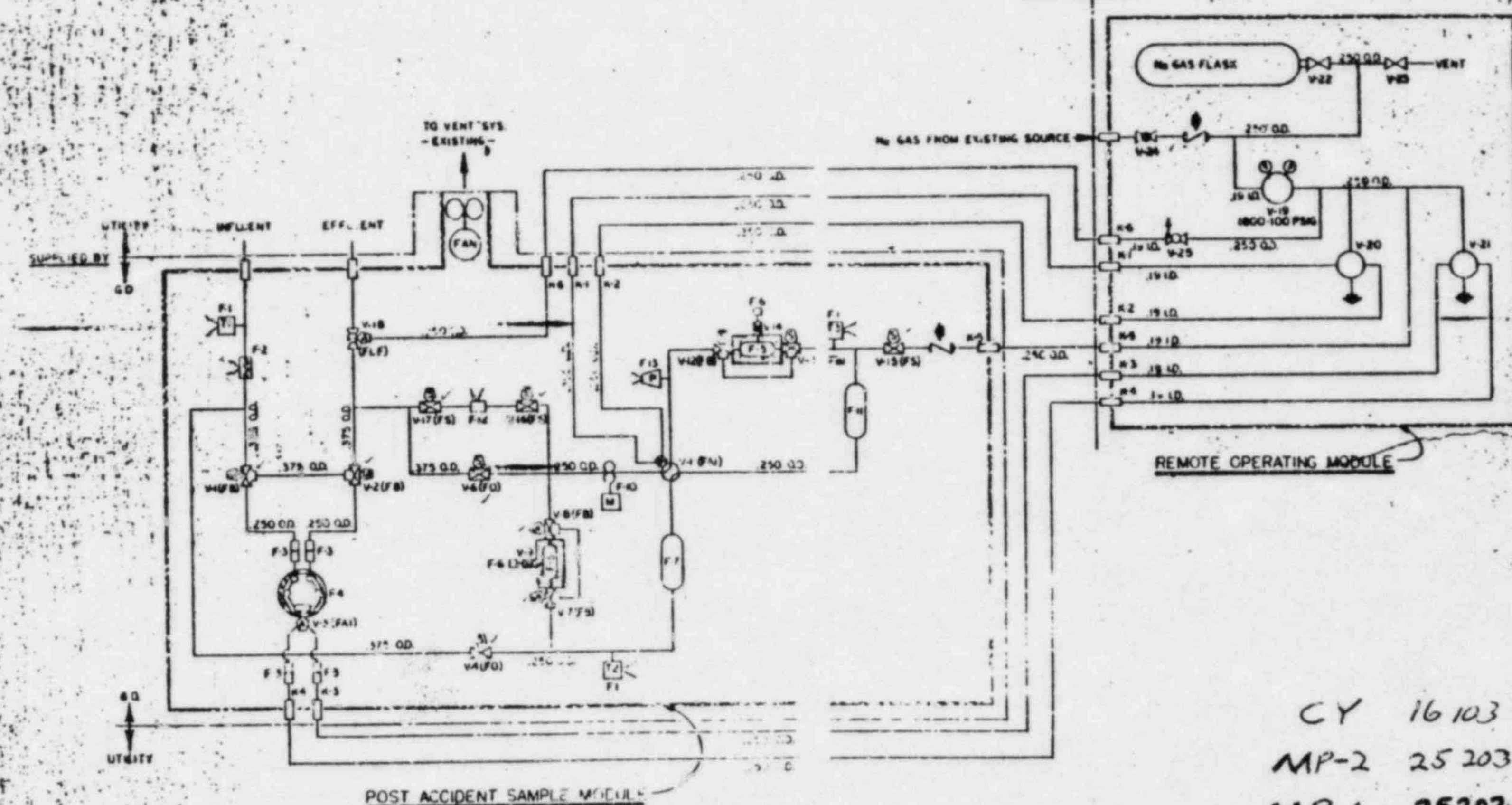
The reactor coolant and containment air PASS will be operationally tested periodically, semi-annually at a minimum, to ensure system availability.

The reactor coolant PASS will be tested by utilizing the demineralized water flush module as a source of influent to the sample module. Also, all remotely operated valves will be operationally tested on a regular basis. Operational testing will not normally be performed using reactor coolant, in order to maintain the system cleanliness.

The containment air PASS will be operationally tested at the same interval as the reactor coolant PASS.

5.0 DRAWINGS/FIGURES


REACTOR
TORUS
TORUS
RECIRC PUMP "A"
RECIRC PUMP "B"
REACTOR COOLANT SAMPLE MODULE
(REF DWG # 6)
2405A
REMOTE OPERATING MODULE
SAMPLE SYSTEM FLUSH MODULE
(REF DWG # 5)
ON HOLD PENDING INSTALLATION
TO REACTOR CLEANUP SAMPLING HOOD
(REF 4)
FILL SUPPLY FROM 2" CST-21
PULSATION DAMPER (AIR LOADED)
SHUTDOWN REACTOR COOLING SYSTEM PUMPS (REF 2)
HEAT EXCHANGER
HEAT EXCHANGER
CTMT AIR SAMPLE COMPRESSOR
CONTAINMENT AIR SAMPLE MODULE
(REF DWG 7)
2406A
SAMPLE FILTER HOLDER (NOTE 2)
REMOTE OPERATING MODULE
(CTMT AIR SAMPLE)
(REF DWG 7)
NITROGEN GAS SUPPLY
TO ATMOS ANALYZER PANEL #2262
REF 4
TO REACTOR ISOLATION SYS
TO CLEANUP SYSTEM
FLEX HOSE
SEISMIC
NOTES:
1. VALVE NO'S FSO-9-75H-1, FSO-9-75I-1, FSO-9-75J 1 & 2 AND FSO-9-75K 1 & 2 WILL BE NORMALLY LOCKED CLOSED BY A SINGLE KEY LOCK SWITCH ON PANEL CRP-925
2. THE CONTAINMENT AIR SAMPLE FILTER IS USED IN DETERMINING DRYWELL PARTICULATE AND IODINE CONTENT. SEE EMERGENCY PLAN IMPLEMENTATION PROCEDURE FOR FILTER OPERATION INSTRUCTIONS
LEGEND
- HEAT TRACING
SC - SAFETY CLASS
REFERENCE DRAWINGS:
1. 25202-29119 SH 2&3 P&ID NUCLEAR BOILER
2. 25202-29126 P&ID SHUTDOWN REACTOR COOLING SYSTEM
3. 25202-26006 FLOW DIAGRAM CORE SPRAY & LOW PRESSURE COOLANT INJECTION SYSTEMS
4. 25202-26021 FLOW DIAGRAM SAMPLING SYSTEM
5. 25202-29135 SH 2 POST ACCIDENT SAMPLING SYSTEM - FLUSH MODULE
6. 25202-29134 SH 2 POST ACCIDENT SAMPLING SYSTEM - SCHEMATIC - LIQUID SAMPLE
7. 25202-29134 SH 3 POST ACCIDENT SAMPLING SYSTEM - SCHEMATIC CONTAINMENT AIR
8. 25202-26011 FLOW DIAGRAM SECONDARY COOLING WTR, AIR EVACUATION & OFF-GAS SYSTEMS
9. 25202-26001 HVAC-FLOW DIAGRAM TURB BLDG, CONTROL RM & SERVICE AREA
POST ACCIDENT SAMPLING SYS
NORTHEAST UTILITIES SERVICE CO.
NORTHEAST NUC ENERGY CO.
MILLSTONE POINT - UNIT 1
FLOW DIAGRAM - POST ACCIDENT SAMPLING SYSTEM
25202-26036



CY 16103 - 29436 SH.2

MP-2 25203 - 29429 SH.2

MP-1 25202 - 29314 SH.2

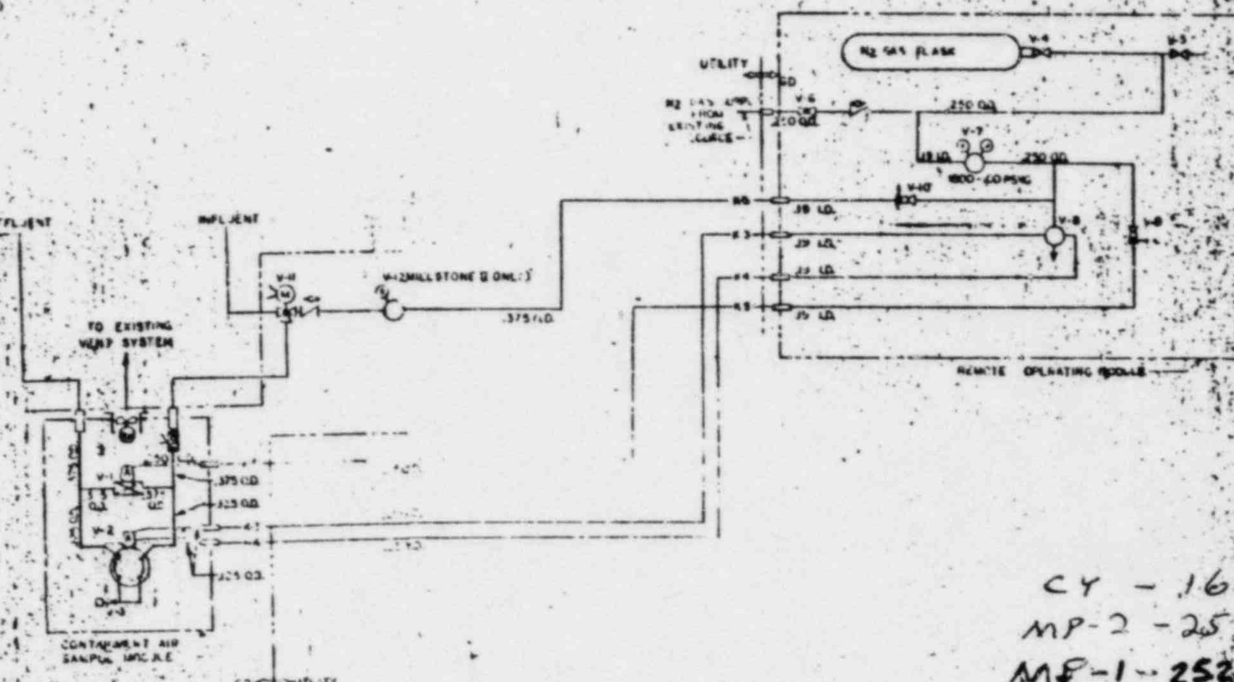
GENERAL DYNAMICS
Electric Boat Division
Reactor Plant Services
GROTON, CONN.

TITLE POST ACCIDENT SAMPLE SYSTEM SCHEMATIC — REACTOR COOLANT

SCALE NTS

DRAWING NO. RPS-10046

SHEET 2 OF 4



CY - 16183-29436 SH3
MP-2 - 25203-29437 SH3
MP-1 - 25202-29314 SH.3

GENERAL DYNAMICS
Electric Boat Division
Reactor Plant Services
GROTON, CONN.

TITLE POST ACCIDENT SAMPLE SYSTEM
SCHEMATIC CONTAINMENT AIR

SCALE NTS	DATE	DRAWING NO. RPS-10016	
DRAWN BY	EXAMINED		
REVISED	APPROVED	SHEET 3 OF 4	REV 1
ENGINEER			

22" 17" 11" 8.5" 8.5" 11" 17" 22"