PSNH

Public Service of New Hampshire

SEABROOK STATION
Engineering Office:
1671 Worcester Road
Framingham, Massachusetts 01701
(617) - 872 - 8100

May 31, 1983
SBN-514
T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) PSNH Letter, dated December 1, 1982, "Revised Response to RAI 281.6; Post-Accident Sampling; (Chemical Engineering Branch)", J. DeVincentis to G. W. Knighton

Subject: Open Item Response (SER Section 9.3.4.2; Chemical Engineering Branch)

Dear Sir:

In response to the Open Item included in the Safety Evaluation Report (Section 9.3.4.2) regarding NUREG-0737, Item II.B.3 (Post-Accident Sampling Capability) we have enclosed a report which addresses the status of our compliance with each of the Item II.B.3 criteria. We have included commitments to complete the development of sampling and analysis procedures, shielding studies, and sampling system time and motion studies not later than six months prior to fuel load.

The enclosed report supplements the information included in Reference (b) and OL Application Amendment 48. Note that the method of obtaining a containment recirculation sump sample has been modified.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John R DeVincentis

J. DeVincentis
Project Manager

Dist Per: PM

H001
Add: NSIC
Chem Engr Br

ALL/fsf
Enclosure
cc: Atomic Safety and Licensing Board Service List

II

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.

RESPONSE: The post-accident sampling system provides the capability to obtain liquid samples from reactor coolant loops 1 and 3 (hot legs), containment recirculation sumps, pressurizer relief tank, ECCS pump room sumps and gas samples of the containment atmosphere within three hours from the time a decision is made to take a sample. All electrically powered equipment (i.e., solenoid valves and sample pumps), whose operation is required to perform post-accident sampling, is powered from an emergency backup power source.

Specific details concerning time spans to enter and exit the sample panel area, operate the sample panel manual valves, perform manual sample dilutions, and transfer sample to the shield cart for analysis will be performed when construction activities allow an accurate appraisal but no later than six months prior to fuel load.

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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 non-volatile 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 to minimize personnel radiation exposures (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.

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

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

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

Onsite radiological and chemical analysis capability will be established to meet the three-hour time frame. The post-accident sampling subsystem provides the capability to obtain liquid samples from reactor coolant loops 1 and 3, ECCS pump room sumps (RHR/CBS Sumps "A" and "B" and PAB Sump "A"), the pressurizer relief tank, and the RHR pump discharge (containment recirculation sump sample). Gas samples of the containment atmosphere under post accident conditions can be drawn from the installed hydrogen analyzer system.

Radionuclides will be measured on grab samples by gamma ray spectroscopy using germanium detectors. Hydrogen concentration in containment will be measured on grab samples of the containment atmosphere. Dissolved gas will be determined by degassing an aliquot of liquid and obtaining a grab sample of gas for hydrogen analysis. Chloride and boron concentration on dilute liquid samples will be determined by analysis of grab samples.

Background levels will be reduced in the counting room through the use of a shielded cave. Personnel radiation exposure will be maintained ALARA through the use of lead shield carrying devices and remote handling devices where appropriate.

A procedure will be developed for relating radionuclide concentrations to core damage levels taking into account core temperature and sample location. Specific radionuclide identification and concentration will be accomplished through the use of a full spectrum scan.

Sampling procedures and core damage level determination procedures are currently under development and will be available six months prior to fuel load.

CRITERION: (3) Reactor coolant and containment atmosphere sampling during postaccident 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.

RESPONSE: Neither the reactor coolant/sump nor containment atmosphere sampling system requires an isolated auxiliary system to be placed in service for the purpose of sampling.

RCS Sampling

In the case of the RCS sampling system, samples can be supplied to the panel directly from reactor coolant loops 1 and 3 (hot legs), the discharge of either RHR pump (containment recirculation sump sample), the discharge of the pressurizer relief tank sample pump, the discharge of the PAB sump "A" sample pump, and the discharge from either RHR/CBS equipment vault "A" or "B" sump sample pump. See revised FSAR Figure 9.3-5c, attached, and FSAR Figure 9.3-5a.

The system interface valves include both manually-operated and remotely-operated valves. All manually-operated valves required for system alignment are equipped with accessible handwheel reach-rod extensions, which are located on a wall adjacent to the sample panel.

All remotely-operated valves are environmentally qualified for the conditions in which they need to operate and are cycled from either the Control Room or local panel.

The post-accident sampling system will provide a means to override the safeguards signals that automatically close the sample isolation valves.

Containment Atmosphere Sampling

The containment atmosphere sampling system draws samples from the hydrogen analyzer suction line and returns the sample flow to containment through the hydrogen analyzer return line. Inside containment, both of these lines are open-ended to the containment atmosphere.

The sample supply and return taps are located in the main steam and feedwater pipe chase building and tie into the hydrogen analyzer suction line upstream and downstream of Valve CGC-V13, for Train "A" (CGC-V35 for Train "B") see revised FSAR Figure 6.2-95, attached.

The system interface valves are manually operated.

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 $\geq$ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Response (4)

The amount of dissolved gases in reactor coolant will be determined by extracting a gaseous sample from the post-accident sampling panel using a shielded syringe if necessary. This sample will be analyzed for hydrogen and gamma spectrum only. The procedure for this analysis is currently under development and will be available six months prior to fuel load.

<u>Criterion</u>: (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.

<u>Clarification</u>: 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 a 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.

<u>Response (5)</u>

Grab sample analysis for chloride on a diluted liquid sample will be completed within 96 hours of drawing the sample. Seabrook Station utilizes seawater for cooling water but the design incorporates a double barrier between primary containment systems and the cooling water. Samples may be diluted up to a factor of 1000 at the sample station. However, the analysis employed will provide for a minimum detectable threshold of 10 ppm Cl.

The post accident sampling system will provide for the capability of taking an undiluted sample consistent with ALARA principles. This undiluted sample will be retained for analysis within 30 days.

Procedures for drawing both the diluted and undiluted chloride samples and for the analysis of the diluted chloride sample are under development and will be available six months prior to fuel load.

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, 10 CFR 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 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).)

RESPONSE: A shielding analysis will be performed, no later than six-months prior to fuel load, to ensure that operator radiation exposure from reactor coolant/containment atmosphere sampling and analysis is within the acceptable limits of 5 rem whole body and 75 rem extremities. The operator exposure will include entering and exiting the sample panel area, operating the sample panel manual valves, performing manual sample dilutions, and transferring sample to shielded cart for analysis.

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 (7)

Boron analysis will be conducted on a diluted liquid grab sample. Procedures are under development and will be available six months prior to fuel load.

Criterion: (8) If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of anlayzing the samples. Established planning for analysis at off-site 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 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 (8)

The Seabrook Station post-accident sampling system does not utilize any inline monitoring capabilities.

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

(a) Identify and quantify the isoptopes 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 μCi/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 accompolished through the use of sufficient shielding around samples and outside sources, and by use of a ventilation system design which will control the presence of airborne radioactivity.

<u>Clarification:</u>

(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.

(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 assure an accuracy within a factor of 2.

Response (9)

(a) Isotopes of the nuclide categories of noble gases, iodines, cesiums and non-volatile isotopes will be identified and quantified to levels corresponding to the source terms given in Regulatory Guides 1.4 and 1.7. There will be provisions for a 1000 to one dilution of the sample at the sampling station. This will be sufficient for transporting a small aliquot to the counting room using lead shield carrying devices and remote handling devices. If necessary, the sample can be counted using collimated counting geometry. Liquid sample measurement capabilities will permit measurement of radionuclides concentration in the range from approximately 1 μCi/g to 10 Ci/g. Procedures for this analysis will be available six months prior to fuel load.

(b) Background levels of radiation will be restricted in the counting room and the primary laboratory through the use of shielding and ventilation to provide results within an acceptably small error. The counting room has 30 inch concrete walls and roof. Ventilation will be controlled to both areas to limit the ingress of airborne radioactivity. All wet chemical analysis will be performed in an operating fume hood. Samples will remain behind shielding during both storage and analysis.

<u>Criterion</u>: (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.

<u>Clarification</u>: 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 anlaysis should be accurate within ± 5% of the measured value (i.e. at 6,000 ppm B the tolerance is ± 300 ppm while at 1,000 ppm B the tolerance is ± 50ppm). For concentrations below 1,000 ppm the tolerance band should remain at ± 50 ppm.

- Chloride: measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

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

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

- Oxygen: monitored to assess coolant corrosion potential.

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

- pH: measured to access coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within ± 0.3 pH units. For all other ranges ± 0.5 pH units 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

Constituient	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 anlayses 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 (10)

The accuracy, range and sensitivity of post-accident analyses capabilities will be detailed in the analytical procedures currently under development. The procedures will be available six months prior to fuel load.

CRITERION: (11) In the design of the postaccident 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 postaccident 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.

RESPONSE: (a) RCS Sampling

The Post-Accident Sampling System (see revised FSAR Figure 9.3-5c, attached and FSAR Figure 9.3-5a) is capable of obtaining samples from reactor coolant loops 1 and 3 under accident conditions where the Reactor Coolant System remains pressurized. Under these conditions, natural circulation could be established to remove decay heat and provide mixing of the RCS water.

Loop samples are taken from existing sampling points located in the hot leg of loops 1 and 3. These sampling points are the sample points utilized for routine primary sampling.

If an accident resulted in the depressurization of the primary system (i.e., a LOCA), long-term, post-accident samples would be provided from the discharge of the RHR pumps. These samples would be recirculation water which is pumped from the containment recirculation sump, through the RHR heat exchanger and back into the reactor vessel. Mixing in the vessel is accomplished as the recirculation water is forced to flow through the core to remove decay heat.

The sampling panel has the capability of obtaining a liquid sample from the pressurizer relief tank when the Reactor Coolant System is either pressurized or

depressurized. This is accomplished through the use of the PRT sample pump.

Equipment/valve leakage will be collected in RHR/CBS Sumps "A" and "B", as well as PAB Sump "A". Samples can be taken from these sumps and supplied to the sample panel from the discharge of the RHR/CBS sump sample pumps and the PAB Sump "A" sample pump, respectively.

The sampling system is purged using demineralized water which is flushed through the system. In the event of a loss of off-site power, the sample system is purged by establishing flow through the sample line, panel and returning the flow to the containment. See revised FSAR Figure 9.3-5c, attached.

The sample purge flows discussed above are within the region of turbulent flow which should promote mixing in the sample lines and help reduce sample plateout and distortion. Each sample line of the Post-Accident Sampling System will be kept as short as possible to limit the volume of fluid needed to be taken from the system.

The sample panel has two sample return lines. One line can be valved to return samples to the containment. This flow path is used under post-accident conditions when it is desirable to return the sample purge flow to the containment.

The second return line is used for normal operation which includes testing and operator training exercises. This line returns sample flow to the Floor and Equipment Drain System.

Flow restrictions in the sample line are provided by flow restrictors (orifice) and solenoid-operated isolation valves to limit reactor coolant loss from a rupture of a sample line.

Strainers/screens shall be incorporated in the sample lines or pump inlets, respectively, to prevent system blockage.

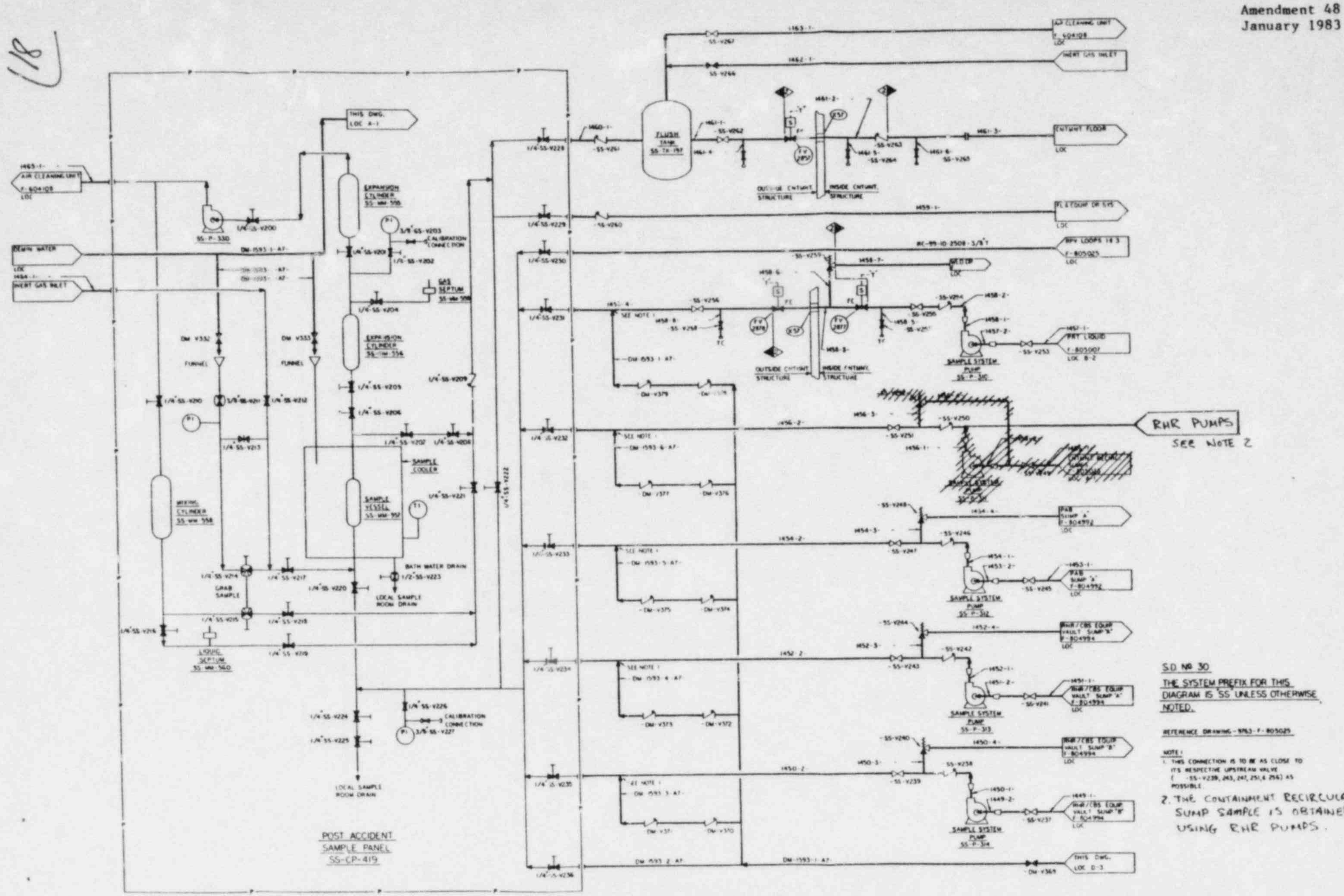
Containment Atmosphere Sampling

The Containment Atmosphere Sampling System draws air samples from two existing lines which are open to the containment at the dome area. Sampling from these areas should provide samples representative of the containment atmosphere. See revised FSAR Figure 6.2-95, attached.

Purging of the sampling lines is accomplished by establishing flow through the hydrogen analyzers using the analyzer sample pump. Removable sample cylinders are installed upstream of the analyzer with bypass lines to allow continuous purging. Redundant sample lines exist (one for each H_2 analyzer), therefore, blockage of a sample line will not prevent the capability to obtain samples.

Heat tracing has been provided to both sample lines and will maintain the sample temperature at approximately 300°F. This will ensure that moisture in the sample will not condense before reaching the sample cylinder. Sample lines will be kept as short as possible to limit the air volume needed to be taken from containment.

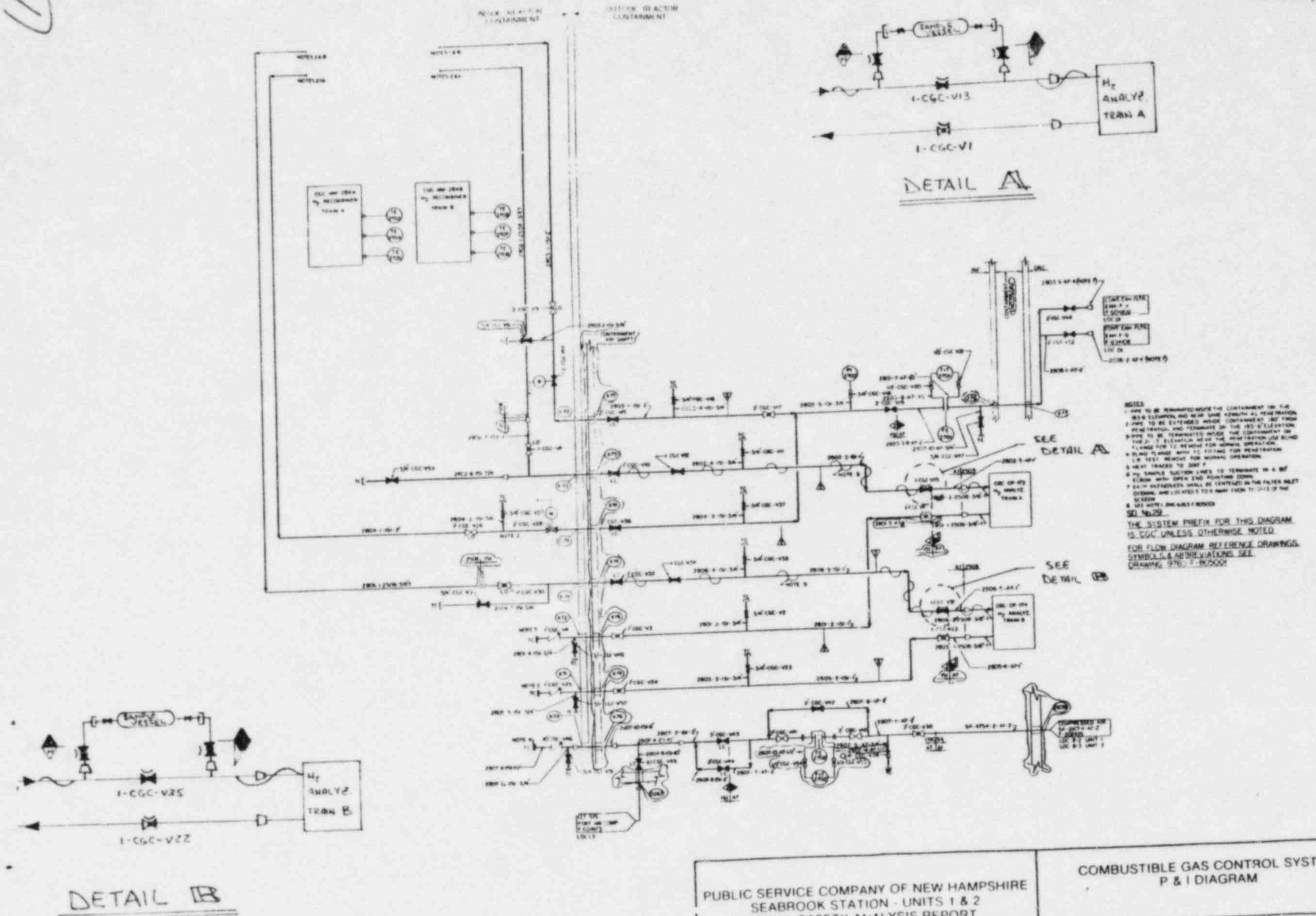
(b) The post-accident RCS sampling panel vacuum pump and vents discharge into the PAB Ventilation System. The PAB ventilation flow passes through roll, medium efficiency, HEPA and carbon filters before being discharged to the plant primary vent stack.



PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE
SEABROOK STATION - UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SAMPLE SYSTEM NUCLEAR-POST ACCIDENT
P&I DIAGRAM

FIGURE 9.3-5c



PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE SEABROOK STATION - UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	COMBUSTIBLE GAS CONTROL SYSTEM P & I DIAGRAM
	9763-F-805022 · FIGURE 6.2-95

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Rep. Beverly Hollingworth
Coastal Chamber of Commerce
209 Winnacunnet Road
Hampton, NH 03842

William S. Jordan, III, Esquire
Harmon & Weiss
1725 I Street, N.W.
Suite 506
Washington, DC 20006

Roy P. Lessy, Jr., Esquire
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Robert A. Backus, Esquire
116 Lowell Street
P.O. Box 516
Manchester, NH 03105

Philip Ahrens, Esquire
Assistant Attorney General
Department of the Attorney General
Augusta, ME 04333

Mr. John B. Tanzer
Designated Representative of
the Town of Hampton
5 Morningside Drive
Hampton, NH 03842

Roberta C. Pevear
Designated Representative of
the Town of Hampton Falls
Drinkwater Road
Hampton Falls, NH 03844

Mrs. Sandra Gavutis
Designated Representative of
the Town of Kensington
RFD 1
East Kingston, NH 03827

Edward J. McDermott, Esquire
Sanders and McDermott
Professional Association
408 Lafayette Road
Hampton, NH 03842

Jo Ann Shotwell, Esquire
Assistant Attorney General
Environmental Protection Bureau
Department of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Olive L. Tash
Designated Representative of
the Town of Brentwood
R.F.D. 1, Dalton Road
Brentwood, NH 03833

Edward F. Meany
Designated Representative of
the Town of Rye
155 Washington Road
Rye, NH 03870

Calvin A. Canney
City Manager
City Hall
126 Daniel Street
Portsmouth, NH 03801

Dana Bisbee, Esquire
Assistant Attorney General
Office of the Attorney General
208 State House Annex
Concord, NH 03842

Anne Verge, Chairperson
Board of Selectmen
Town Hall
South Hampton, NH 03842

Patrick J. McKeon
Selectmen's Office
10 Central Road
Rye, NH 03870

Ruthanne G. Miller, Esquire
Law Clerk to the Board
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Maury Tye, President
Sun Valley Association
209 Summer Street
Haverhill, MA 01830

Mr. Angie Machiros
Chairman of the Board of Selectmen
Town of Newbury
Newbury, MA 01950