John D. O'Toole
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003
Telephone (212) 460-2533

November 15, 1983

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Dear Mr. Varga:

This letter and attachment provide the additional information on NUREG-0737 Item II.B.3 "Post Accident Sampling System" as requested by your letter dated June 30, 1982. This submittal includes a response to each criterion as delineated in your June, 1982 letter and our current interim procedure for assessment of core damage following a postulated accident.

Should you or your staff have any questions please contact us.

Very truly yours,

John D. O'Toole

attach.

cc: Mr. Thomas Foley, Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

8311290123 831115
PDR ADOCK 05000247
P PDR

A046
1/1

50- 247

POST ACCICDENT SAMPLING SYSTEM IN RESPONS
TO LTR DTD 6/30/82

— NOTICE —

THE ATTACHED FILES ARE OFFICIAL RECORDS OF THE DIVISION OF DOCUMENT CONTROL. THEY HAVE BEEN CHARGED TO YOU FOR A LIMITED TIME PERIOD AND MUST BE RETURNED TO THE RECORDS FACILITY BRANCH 016. PLEASE DO NOT SEND DOCUMENTS CHARGED OUT THROUGH THE MAIL. REMOVAL OF ANY PAGE(S) FROM DOCUMENT FOR REPRODUCTION MUST BE REFERRED TO FILE PERSONNEL. 50-247

11/15/83

DEADLINE RETURN DATE 8311290 123

RECORDS FACILITY BRANCH

Attachment A

Post Accident Sampling System
Response to NRC letter dated June 30, 1982

Note: The information contained in Attachment A is organized as follows:

- Criterion from NUREG-0737, item II.B.3
- Clarification of criterion provided by NRC (letter of June 30, 1982)
- Con Edison Response

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).

CON EDISON RESPONSE

The design of the Indian Point Unit No. 2 Post Accident Sampling System (PASS) incorporates in-line sampling and analyses of the Reactor Coolant System and Containment atmosphere samples. The PASS contains all equipment necessary for in-line sample acquisition and chemical and radiochemical analyses for the Indian Point Unit No. 2 (except for containment atmosphere noble gases, Iodine and particulate analysis) as required by NUREG-0737, Item II.B.3. The equipment includes:

1. a Canberra In-line Isotopic Analyzer;
2. Sentry High Radiation Sampling System's (HRSS) Liquid Sampling Panel (LSP), Chemical Analysis Panel (CAP) and Chemical Monitoring Panel (CMP);
3. Sentry Modified DigiChem Boron Analyzer; and
4. a Comsip-Delphi Hydrogen/Oxygen Monitor.

All of these systems are located on the 80' level of the Primary Auxiliary Building (PAB), in either the Waste Gas Compressor room, the High Radiation Sampling System (HRSS) room or the Containment Atmosphere Sampling room. The Indian Point Unit No. 2 PASS is capable of sampling various streams, including RCS hot legs Nos. 21 and 23, recirculation pump discharge, RHR loop, and containment atmosphere.

The PASS has the capability to purge lines before sampling to ensure that representative samples can be obtained. Sample lines for the LSP, CAP and Containment Atmosphere Sample System can be flushed after the sampling operations are complete to reduce residual radioactivity.

To satisfy NUREG-0737, Item II.B.3 Criterion (8), (i.e., to provide further back-up reliability) and to provide analysis for containment atmosphere noble gases, particulates, and Iodine, monitoring can be completed by taking grab samples and taking them to the on-site Radio-Chemistry/Counting Room for analysis. Gaseous containment grab samples can be obtained from a manifold at the Comsip-Delphi $H_2/0_2$ monitors through quick disconnect couplings. Liquid RCS samples for back-up in-house analysis (or stored for subsequent off-site analysis) can be obtained from the LSP during accident conditions. The types of samples which can be obtained include:

a. Undiluted liquid samples from the reactor coolant

b. Diluted (1 to 1000) liquid samples from the reactor coolant

c. Diluted (1 to 15,000) stripped gas from the reactor coolant pressurized liquid sample.

Grab samples from the sampling panels are transported in a sample ladle approximately 255 feet to the Radio Chemistry Lab located on the 53' level of the Nuclear Service Building #1.

Using current plant procedures, the time to collect a sample (including purging of lines), transport it and perform an analysis was calculated to address Criterion (6) below. The results of this assessment demonstrate that after a decision has been made to take a sample, the total time for obtaining the sample and performing the necessary analysis for RCS samples and Containment atmosphere samples is well within the three hour time frame prescribed by NUREG-0737 Item II.B.3.

The power supply for the PASS is from a vital class 1E bus. In the event of a loss of off-site power, back-up power supply is provided from the plant diesel generators to allow in-line sampling and analyses to be completed within the 3 hour time period.

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 in-line 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 Xe-133, I-131, Cs-137 Cs-134, Kr-85, Ba-140, and Kr-88 (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).

CON EDISON RESPONSE

2(a) The Indian Point Unit No. 2 PASS has an in-line system which utilizes an undiluted sample for isotopic analysis. The in-line isotopic analyses are performed by a Canberra Isotopic Analyzer utilizing a Germanium detector connected to a computer based multichannel analyzer which identifies and quantifies the radionuclides present based on counts and energy levels. In addition, the PASS has the capability of obtaining grab samples of the containment atmosphere and reactor coolant system. Isotopic analysis of grab samples is performed in the Chemistry Lab/Counting Room utilizing a Canberra Isotopic analyzer to monitor short and long lived volatile and non-volatile radionuclides.

The PASS design utilizes time, distance, shielding and sample dilution (grab sampling) as the primary means of reducing personnel exposure.

Con Edison has developed an interim procedure to assess the extent of core damage following a postulated accident. The procedure, which is intended to be used by corporate support personnel to characterize core damage in terms of clad failure, fuel over temperature or fuel melt, utilizes, as primary inputs, post accident sampling results. The Westinghouse Owners Group (WOG) is currently developing a comprehensive methodology utilizing post accident sampling data and other applicable plant indicators to estimate the extent and type of core damage. Con Edison is a member of WOG and will utilize final WOG guidance to implement its final core damage assessment methodology.

2(b) Hydrogen levels in the containment atmosphere are monitored by utilizing equipment installed to satisfy NUREG-0737 Item II.F.1.6. Redundant Comsip Delphi Hydrogen/Oxygen monitors obtain containment atmosphere samples from two independent heat traced lines. These systems are provided with a local meter readout in the Motor Control Center of the PAB (El 98') as well as a recorder/display in the Central Control Room.

2(c) The capability to sample and analyze for dissolved gases, chloride, boron concentration, and accident sample species is discussed in the following response to Criterion 2(d). The IP2 PASS is capable of analyzing species listed in NUREG-0737 as well as Reg. Guide 1.97, Rev. 2.

2(d) The reactor coolant analyses can be done with various in-line instrumentation. There is also in-line capability to monitor hydrogen and oxygen concentration in the containment atmosphere. The NUS Corporation, our PASS system consultant, has performed in-depth testing of the RCS analyzers to demonstrate their applicabilty for this service. The methods and instruments used for performing the above analyses were selected based on: 1) reliability, 2) accuracy of technique, 3) simplicity of operation, 4) application to accident conditions, 5) minimum operator interaction during analysis sequence, and 6) wide measurement range.

A summary of the methods and instrumentation used is presented below:

Boron -	Titration method using a Sentry Boron Analyzer (Modified Ionics Digichem 3000)
Chloride -	Chloride content of the reactor coolant is determined using a modified Dionex Model 10 ion chromatograph.
Dissolved - Hydrogen	Dissolved hydrogen is measured by gas chromatography using a Baseline model 1030A analyzer.
Dissolved - Oxygen	Dissolved Oxygen concentration is determined by the polarigraphic method using a Yellow Springs Instrumentation (YSI) and Rexnord analyzer.
Conductivity -	To determine conductivity a Beckman conductivity probe is used with a cell constant of 0.1 cm^{-1}.
pH -	pH analysis is performed in-line using a Cole-Parmer combination probe.
Containment - H_2/O_2	H_2/O_2 Concentration is determined by thermal conductivity measurement of gases using a Comsip-Delphi Gas Analyzer

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.

CON EDISON RESPONSE

Reactor coolant and containment atmosphere sampling and recirculation during post accident conditions are not dependent on the availability of any auxiliary system which may be isolated during accident conditions. The IP2 PASS is shown in the system schematics provided by Figures 1 and 2.

The reactor coolant sampling system (Figure 1) would normally receive a pressurized sample from the hot leg loop 21 or 23. A sample booster pump is installed on this sample line to provide a RCS sample should the post accident sample pressure prove inadequate.

Containment atmosphere samples (Figure 2) are obtained from the five containment recirculation fan discharges through a common manifold and two separate heat traced lines. A sample is drawn into two separate class 1-E Comsip-Delphi Hydrogen and Oxygen Monitors by diaphragm pumps and discharged back to containment.

SOVs 5018, 5019, 5020, 5021, 5022, 5023, 5024 and 5025 are located on a rack, Elev. 67'-6", outside the containment wall, where sample penetrations come through the containment. These SOVs are normally closed with the Comsip-Delphi in the standby position. They are opened remotely when taking a sample by operating switches on the Comsip-Delphi Remote Panels in the MCC room, Elev 98' P.A.B.

Subsequent to post accident sampling, radioactive gas can be sent back into containment by displacing with nitrogen gas introduced at the above solenoid rack and controlled from valve 5120 in the MCC room, Elev 98' PAB.

The activity of the containment gas sample can be currently obtained from a bypass line with a septum installed across valves 5112 and 5113. Closing valve 5137 permits withdrawing a small syringe sample for analysis in the Radio Chemical Lab for gas, particulates and Iodine.

An alternative method for analysis could be accomplished by passing the containment atmosphere sample to a Canberra in-line isotopic analyzer by opening valves 5171 and 5173 and closing valve 5172. This method would require initially purging out the RCS liquid filled side of the analytical cell with primary water. The primary water supply is received from the Boric Acid Evaporator Tank and is considerably less active than the RCS sample flowing through the

Canberra in-line isotopic analyzer. Once a background level is obtained for fresh water, the containment atmosphere sample could be admitted to the gas side of the Canberra analytical cell. The accomplishment of this alternative method has not yet been demonstrated.

Remote - operated valves in the PASS are environmentally qualified for the postulated harsh environment in which they must operate. These valves were reviewed under the equipment qualification program and separately addressed in previously submitted environmental qualification reports.

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 hydrogen gas in reactor coolant samples is considered adequate. Measuring the oxygen 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 ALARA, direct monitoring for dissolved oxygen is recommended.

CON EDISON RESPONSE

The IP2 PASS is equipped with in-line instrumentation to perform both dissolved hydrogen and oxygen analyses of reactor coolant. Hydrogen analyses are performed by stripping the gases from a pressurized reactor coolant sample and measuring the hydrogen concentration using a gas chromatograph. Dissolved hydrogen concentration can be determined in the range of 2-2000 cc/kg. The range of the dissolved oxygen analyzer is 0.1 to 20 ppm.

These methods measure the total dissolved hydrogen and oxygen concentrations which are directly related to reactor coolant concentrations. Therefore, if the chloride concentration exceeds 0.15 ppm, the dissolved oxygen can be verified to be less than 0.1 ppm by either method.

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

BWRs 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. Additionaly, 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.

CON EDISON RESPONSE

The Sentry High Radiation Sampling System Chemical Analysis Panel and Chemical Monitor Panel provides the capability to remotely perform in-line analysis of a reactor coolant sample for chlorides within 24 hours. The Sentry Liquid Sampling Panel also provides the capability to collect an undiluted reactor coolant sample for back-up analysis.

The LSP will be used to collect 15 ml of an undiluted sample. The

CAP is equipped with a septum port and valve for manual injection of a sample for chloride analysis. A syringe will be used to withdraw a sample from the undiluted sample bottle and inject 10cc into the sample loop from the rear of the panel. The dose rate from the 10cc sample is estimated to be approximately 4R/hr at 1 foot, 30 days after the accident. Three inches of leadshot shielding would reduce the dose rate to 100 mr/hr at 1 foot.

To date, no chloride analysis procedure is available with sufficient sensitivity to permit quantitative analysis of chlorides indicating $<$ 10ppm on a 1-1000 diluted sample. The ion chromatographic technique used in the Sentry system provides the capability to detect chloride concentrations of less than 100 ppb (0.1 ppm) on a 0.2 ml undiluted sample which is injected in-line.

The range for I.C. Chloride analyses is 0.1 ppm-20ppm. The upper range value can be extended by calibrating in this region and varying the micromho setting on the conductivity meter. The Sentry system is capable of determining chloride concentrations in the range of 0.1-20.0 ppm.

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.

CON EDISON RESPONSE

The PASS in-line monitoring system is designed to maintain personnel exposure ALARA and within the limits of GDC 19. The PASS utilizes remote operation, time, distance, shielding and dilution to minimize personnel exposure. The maximum dose rate would occur if the grab sample were used.

A time/motion study in conjunction with dose rate calculations was performed to predict the doses to the whole body and to the extremities of an individual involved with reactor coolant and containment atmosphere post-accident sampling activities. The dose rates used in the calculations are based on NUREG-0578 source terms which are consistent with Regulatory Guide 1.4 source terms and have been modified for IP#2 containment and reactor coolant volumes. Also, dose rates from the source have been taken for t = 1.0 hrs.

Based on the performed analysis, the predicted exposure to a single individual is 500 mR for reactor coolant analysis and 1.5R for containment gaseous monitoring. Both cases are within the GDC-19 Criterion requirements.

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

PWRs need to perform boron analysis. The guidelines for BWRs are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

CON EDISON RESPONSE

The IP#2 PASS provides the capability to perform in-line boron analysis. In addition, the PASS provides the capability of obtaining diluted and undiluted grab samples for back-up boron analysis.

CRITERION (8)

If inline monitoring is used for any sampling and analytical capability specified herein, the licensees shall provide backup sampling through grab samples, and 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.

CON EDISON RESPONSE

The IP#2 Post Accident Sampling System provides undiluted and diluted reactor coolant samples and diluted reactor coolant stripped-gas samples in addition to its in-line monitoring capabilities. The diluted reactor coolant sample is collected within a shielded cart/cask assembly which interfaces with the Liquid Sampling Panel. The 1-1000 diluted liquid sample will be transported to the on-site Radio-Chem Laboratory for chemical and isotopic analyses.

Dissolved gases from the reactor coolant are stripped from a 30 ml pressurized liquid sample which is collected in-line behind the LSP shield wall. The stripped gases are routed from a gas expansion vessel through a diluter valve prior to being routed to an in-line gas chromatograph within the Chemical Analysis Panel. The diluter valve is used to perform a gas dilution of approximately 1-15,000. The diluted gas is collected within a 10cc glass bottle held in place by an off-gas grip tongs. The bottle is removed and transported to the on-site Radio-Chem Laboratory for isotopic analysis. The concentration of hydrogen in the diluted (1-15,000) stripped gas from the RCS sample is too small for for accurate analysis by existing equipment in the Radio-Chemical laboratory. In the event that analysis cannot be performed by the in-line Baseline Gas Chromatograph, the backup capability is to repair the inline instrumentation with backup spare parts. An alternative method using the CAP is being evaluated with the equipment vendor, Sentry Equipment Corporation, that would permit obtaining a more concentrated hydrogen sample for subsequent lab analysis. Gaseous containment grab samples can be obtained from a manifold at the Comsip-Delphi Hydrogen Monitors.

In-line sample and analysis lines can be flushed after the sampling operation are completed to reduce residual radioactivity and to facilitate access for repair.

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

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 effect will be on a sample being counted to assure an accuracy within a factor of 2.

CON EDISON RESPONSE

9(a) The IP2 PASS incorporates inline isotopic analysis (Canberra Isotopic Analyzer) with a capability of measuring isotopic activity in the range of 1 uCi/g to 10 Ci/g. The predicted activity in the RCS liquid (undiluted and undegassed) based on source terms given in Regulatory Guide 1.4 and 1.7 is 5.49 Ci/g. For back up sampling and analysis an isotopic analyzer similar to the in-line instrument is provided in the counting room.

9(b) The background radiation levels in the area of the inline isotopic analyzer are not expected to impact its measurement capability. Sufficient shielding is provided to assure that sample analysis results have acceptably small errors. In addition, the analyzer is located in an area served by the existing PAB ventilation system which contains HEPA and charcoal filters.

The back-up analysis is done in the counting room which is separate from the Radio-Chem Lab and from the sampling location. The Counting Room is located in an area where no post accident high radiation lines pass through and affect instrument sensitivity.

The grab samples to be counted are collected at the LSP in a 60 ml septum bottle and diluted sufficiently so that the sample activity is comparable to normal sampling conditions. The radiation fields from the samples decrease rapidly with distance and are shielded from the counting area so that there is no impact on the radionuclide counting equipment.

In the case where radiation levels become sufficiently high due to the airborne activity and preclude back up analysis, several alternatives will be evaluated to obviate this condition. These alternatives include taking the sample to the New York Power Authority Indian Point Unit No. 3 laboratory facilities for analysis, use of portable counters, or modifications to the ventilation system to control the presence of airborne activity.

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 ±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 ±50 ppm). 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 concentrations 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 assess 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

Constituent	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs+	250	Cesium Nitrate
Ba+2	10	Barium Nitrate
La+2	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	Absorbed 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 sampling being tested.

2) For PWRs, procedure 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.

CON EDISON RESPONSE

The IP2 Post Accident Sampling System was developed to provide in-line analyses and in-line dilutions for grab sample analyses which would meet the requirements of NUREG-0578, Section 2.1.8a, NUREG-0737 Item II B.3, and Regulatory Guide 1.97 Revision 2. Undiluted coolants and containment air samples can be captured in shielded containers for on-site or off-site analyses. The Sentry HRRS equipment can also be used for routine sampling and analyses.

The Cyrus Wm. Rice Division of NUS Corporation developed and tested the methods selected for post-accident boron, chloride, dissolved hydrogen and oxygen, pH and conductivity analyses. The analytical requirements, selected method capabilities, and accuracies achievable based on testing are summarized in Table 10-1. The NUS development and test program considered the following criteria in reviewing the analytical methods available to meet the post-accident requirement.

1. Chemical effect of the post-accident coolant matrix
2. Analyses time and radiological dose limitations
3. Radiation effect on method and/or equipment
4. Application to both normal and post-accident conditions
5. Compliance with sensitivity and range requirements

6. Applicability to in-line or grab sample analysis
7. Sample size requirements

The methods evaluated, testing performed, and reasons for selection of the appropriate method are summarized for each analysis in the following discussion. Complete details of the NUS development and testing program are presented in NUS Report R-27-11-1-3, Final Verification Testing of HRSS Chemical Analysis Panel, NUS Report R-27-4-1-1, Development of Procedures and Analysis Methods for Post Accident Reactor Coolant Samples and NSAC/46, An Evaluation of On-line Boron Analyzers.

GROSS ACTIVITY

For gamma spectroscopy, accuracy of radiological analysis with the Canberra Isotopic Analyzer (MCA) is well within a factor of two.

BORON

NUS evaluated the overall advantages of using the Digi-Chem analyzer for boron determinations during accident condition, which are as follows:

(1) Most operations can be performed remotely. The exposure involved in determining boron concentration would be minimal.

(2) Sample volume requirements are on the order of 1 ml per analysis, thus shielding requirements would be minimal.

(3) Analyses results can be achieved within 10 minutes after the sample line is purged to obtain a representative sample.

(4) The analyzer's Titration Unit (TU) receives a degassed sample from the Sentry CAP. Also the TU is connected to the Sentry HRSS HEPA/Charcoal Filter exhaust system to eliminate the release of any potential gaseous activity to the environment.

The analyses results for matrix solutions containing simulated fission product species and caustic solutions indicate that the concentrations of fission product species expected following an accident will not interfere with boron analyses results. The data also indicate that boron analyses results will not be affected by the caustic added to the primary coolant when the containment sprays are activated during a LOCA event. Only limited testing was performed concerning the effect of lithium additions alone on boron analyses results. The addition of lithium had no apparent effect on accuracy or precision.

The data concerning pH indicate that high radiation levels (10^6 R/hr) will decrease the indicated pH by about 0.1 pH units for an internal reference probe. Indicated pH will increase by about 0.1 or 0.2 pH units for external reference probes in a high radiation field. An initial effect is noted at 10^3 R/hr. The increase in pH is immediate. The effect is fully reversible when the radiation source is removed. The DigiChem system has an external reference pH probe.

The shift in pH resulting from radiation should have a slight effect on accuracy of analyses with the DigiChem analyzer; however, the effect will not be significant in meeting post-accident requirements. During normal operating conditions, the system will be titrating from pH 5.5 to 8.5 to determine boron concentration. Under high radiation conditions the system will still titrate from an indicated pH 5.5 to pH of 5.3 to 8.3 because of the radiation induced shift in pH.

The Sentry design includes a driven shield which is designed to prevent radiation induced leakage in the cable shield to the pH electrode. This driven shield is a standard feature in all DigiChem analyzers. It should be emphasized, however, that the system tested without driven shield operated satisfactorily at radiation levels anticipated under post-accident conditions. Increased reliability can be anticipated with the addition of the driven shield.

CHLORIDE

NUS evaluated several chloride analysis methods and selected ion chromatography as the method best suited for post-accident use. This method was selected primarily because of the need to analyze undiluted samples to meet the sensitivity required, and the subsequent radiological impact and chemical interferences associated with other methods.

The results of the evaluation and testing performed to qualify the ion chromatographic method are summarized below. The following methods were included in the evaulation:

Ion Chromatography

Selective Ion Electrode

Mercuric Nitrate Titration

Spectrophotometric Methods

Ion chromatography is a tool for the quantitative analysis of low ionic concentration solutions. The method, applicable to both anions and cations, is based on the elution selectivity of ion exchange media for different ionic species. Sample size, physical size of the ion exchange column, ionic capacity of the resin, eluent concentration, and sample flowrate can all be held constant. The time after sample injection that a conductivity peak appears can be assigned to a specific ionic species. Ion chromatographs can be automated and can be operated remotely, thus optimizing their use for in-line chloride analyses under post-accident conditions.

NUS testing was performed on a Dionex Model 10 ion chromatograph - the same model installed in the Sentry HRSS Chemical Analysis Panel. Testing was performed using post-accident matrix solutions containing various concentrations of chloride from 0.1 ppm to 20 ppm. The range of analyses can be extended with additional calibration curves and varying attenuation settings. The test results indicate the absolute mean bias for the ion chromatographic analysis of post-accident matrix samples is 7.6 percent for chlorides in the range of 0.1 to 1 ppm and 5.8 percent for chlorides in the range of 1 to 20 ppm. The test results indicate the ion chromatographic method meets the range and accuracy requirements of the NRC post-implementation evaluation criteria guidelines.

Additional testing of the ion chromatographic method was conducted by NUS to determine if other contaminants, which could exist in the coolant through mixing with sump waters, would affect the chloride analysis. Contaminants tested included morpholine, hydrazine, ammonia, and natural and synthetic oils. No effects of morpholine, hydrazine, and ammonia were observed. Oil contamination increased the chloride results by about 10-30 percent. This result could be due to soluble chlorides present in the oils; however, this was not verified. Fouling of the resin columns was observed by progressive reduction of the time required for the appearance of the chloride peak. Approximately 10 to 15 oil contaminated samples could be analyzed under post-accident conditions without significant resin fouling.

DISSOLVED HYDROGEN

NUS reviewed various methods of measuring dissolved hydrogen concentration in reactor coolant and concluded in-line gas chromatographic techniques were best suited for post-accident applications. Hydrogen concentration in reactor coolant can be determined rapidly (20 to 30 minutes) and accurately by in-line gas chromatographic analyses using the Baseline 1030A which is installed in the HRSS chemistry analysis panel. The analytical section of the gas chromatograph, installed in the rear of the Chemical Analysis Panel, employs a dual column, thermal conductivity detector system to separate and measure hydrogen. The controller for the gas chromatograph is installed in the Chemical Monitor Panel.

Hydrogen gas is initially stripped from a 30 ml coolant sample captured in an in-line sampling flask in the Liquid Sample Panel. After collection of the liquid sample, hydrogen and other gases are stripped into a 300 cc evacuated flask. The stripped gas is routed to the Chemical Analysis Panel and to four 1 cc sampling loops of the gas chromatograph. Gas in the sampling loops is injected into the gas chromatograph by continuous flow of argon carrier gas. Residual water vapor is retained on the first column until the sample gases have eluted and then is back-flushed from the system.

Initial testing to establish optimum conditions was performed by NUS using a Fisher Model 1200 gas chromatograph. Both the Fisher and Baseline gas chromatograph have similar operating characteristics. Final calibration and identification testing was performed using a Baseline Model 1030A gas chromatograph as installed in the Sentry HRSS. The absolute mean deviation of results is 7.1 percent in a hydrogen range of 3.1 to 28 cc/kg at STP. The absolute mean deviation of results in the 21.7 to 186 cc/kg at STP was 8.8 percent. For the hydrogen range of 186 to 930 cc/kg at STP the absolute mean deviation was 7.6 percent. The absolute mean deviation was 2.8 percent in the hydrogen range of 930 to 1860 cc/kg at STP.

Based on the data, the Baseline gas chromatograph installed in the HRSS may be used to determine dissolved hydrogen concentration in the range of 3 to 2000 cc/kg at STP with an accuracy within the requirements of the NRC post-implementation evaluation criteria guidelines. The accuracy statements apply to determinations made using separate calibration curves for each sample loop and for each of the four attenuation settings available. The operator must relate his determined peak height units to the calibration curve corresponding to the loop and attenuation scale used during the analysis.

DISSOLVED OXYGEN

Since accurate dissolved oxygen results required analysis of undiluted samples with minimal exposure to air, in-line analyses methods must be performed.

NUS reviewed the capabilities of various in-line oxygen analyzers and concluded that two analyzers, a Yellow Springs Instrument analyzer and a Rexnord analyzer, were required to meet the objectives of providing monitoring capabilities under both normal plant operating conditions and post-accident conditions.

The Yellow Springs Instrument (YSI) Analyzer, while not as sensitive as the Rexnord analyzer, meets post-accident criteria. The YSI was selected for post-accident use because the response time of the probe was relatively short (i.e., 5 minutes) and the smaller probe holder volume would lower associated radiation levels. The Rexnord analyzer provides the sensitivity required during normal operating conditions. Preliminary testing to determine the effect of variable flow and high dissolved gases was performed. The conclusions drawn from this testing were:

1. The YSI Model 54 oxygen analyzer is relatively insensitive to flow rates from 58 ml/min to 547 ml/min.

2. Hydrogen in solution and microscopic bubbles of hydrogen in the water have no effect on oxygen readouts.

3. The YSI oxygen analyzer has adequate accuracy at 0.10 ppm oxygen.

Final verification testing was performed on both the YSI and Rexnord dissolved oxygen analysis systems as installed in the HRSS. Testing results in the YSI probe lead to the specified flow time of five minutes at 200 ml/min for obtaining probe equilibrium. Testing with the Rexnord probe indicated that 15 minutes was required to reach equilibrium.

Testing was performed using both a PWR and BWR post-accident matrix and comparing the dissolved oxygen concentration indicated by the probes with results obtained using the modified Winkler method. The absolute mean bias for YSI probe results on samples varying from 0.38 to 8.0 ppm dissolved oxygen was 8.2 percent. The mean absolute bias for the Rexnord probe was 5.7 percent on samples varying from 0.1 to 10 ppm oxygen.

Based on the final verification testing, the YSI probe can be used during post-accident conditions to determine dissolved oxygen concentrations with an accuracy of at least ±10 percent in the test range of 0.1 to 10 ppm. The Rexnord probe data supports the value of ±10 percent accuracy for dissolved oxygen in the test range of 0.1 to 20 ppm. Testing at oxygen concentrations up to 20 ppm was not performed; however, the linear response obtained indicates measurements at the higher concentration are achievable. This data indicate the requirements established by the NRC in the post implementation evaluation criteria guidelines can be met.

PH

pH analyses are performed in-line using a Cole-Parmer combination pH probe mounted within a probe holder. Calibration of the probe is conducted in-place using pH 7 and pH 4 or 10 buffer solutions. Readout indication is provided by a Beckman pH monitor mounted within the Chemical Monitor Panel.

During final verification testing, two calibration tests were performed. Initially a pH 7 buffer was flowed through the probe holder and the pH monitor adjusted to read 7.0. The probe was removed from the probe holder and immersed in a pH 10 buffer; the reading obtained was 9.8. The probe was rinsed and immersed in a pH 4 buffer and a reading of 3.8 obtained. The second calibration involved flowing calibration buffers of pH 7 and 10 from the calibration tanks. Readings of 9.7 were obtained for the pH buffer.

Testing of the pH probe included flowing of PWR and BWR matrices through the Chemical Analyzer Panel, recording readings and comparing these readings with open grab sample measurements of the stream effluent. The data obtained indicate an absolute mean bias of ±0.3 pH units over a pH range of 5 to 8.

RADIATION EFFECTS

A study made to determine the effect of radiation on equipment in the NUS analysis system indicates that the radiation levels anticipated will have no effect on accuracy of measurement and very little, if any, effect on operating lifetime of the individual components with the system. The study included a survey of personnel who have performed similar analysis under high radiation levels in hot cell conditions and a review of the literature concerning irradiation effects on materials of construction. In addition, laboratory experiments were performed to determine the effect of concentrations of hydrogen peroxide on the conductivity of the reference solution. No effect was noted. The laboratory work was performed because there could be trace concentrations of peroxide in the coolant under zero or near zero hydrogen concentrations in the primary coolant.

In performing the survey, personnel were contacted at Battelle-Northwest and Oak Ridge Laboratories. Operation experience exists within the two laboratories concerning all phases of the analyses performed by the subject system. Results from these laboratories are consistent as both indicated the analytical measurements in question can be made at much higher radiation levels than will be experienced under post-accident conditions in the automated analysis system. One general conclusion that can be made from this survey, is that analytical instruments can be operated in a hot cell without loss in accuracy of results. There will be some reduction in operating lifetime of the system where continuing exposure at very high radiation levels is involved. Results of this survey as it applies to the individual analytical components within the system are discussed below:

a. Boron Determination

The Ionics Digi-Chem, as modified by Sentry, performed properly at radiation levels of 8.64×10^4 R/hr. Maximum radiation levels anticipated under credible accident conditions are on the order of 10^4 R/hr.

The Sentry boron analyzer operated at an integrated dose of 2.7×10^7 rads. This corresponds to about three months of operation at maximum dose rates anticipated under accident conditions. Sample volume requirements are on the order of 1 ml per analysis, thus shielding requirements are minimal.

b. Chloride Determination

Radiation levels for the chloride analysis equipment (ion chromatograph) will be lower than is the case for the other equipment in the analytical system. The reasons for this are as follows: (1) the maximum primary coolant volume contained within the ion chromatograph will be on the order of 1 ml, and (2) the system will be located further away from the radiation source than are other components in the system. Components in the ion chromatograph receiving relatively high radiation exposure include the following:

- Cation resin
- Anion resin
- Conductivity cell
- Transistors
- Capacitors
- Wire-wound potentiometers
- Circuit board

No radiation damage is anticipated with resins based on experience developed at Battelle. Resins are conventionally used to separate various isotopes at Battelle without problems. Damage occurs after extended exposure; however, the degradation process is slow. Anion resin will start to degrade at about 10^8 rads and cation resins at 5×10^8 rads. This is several orders of magnitude higher than will be encounted for the resins in the ion chromatograph. It is anticipated that electronic components of the type included in the ion chromatograph will be resistant to cumulative exposure well above 10^5 rads.

In the absence of dissolved hydrogen, it is possible that the primary coolant will contain low ppm concentrations of hydrogen peroxide from radiolysis of water. The effect of this peroxide on readout of the conductivity cell in the ion chromatograph was investigated by adding peroxide to a solution containing 2 ppm HCl and measuring conductivity before and after the addition of peroxide. The justification for using a 2 ppm HCl solution is noted below. Essentially no effect was observed on conductivity within the limits of the accuracy of the instrumentation.

Chloride determinations on an ion chromatograph are obtained by processing the water through resin columns and monitoring the conductivity of the effluent. The chloride (or other anions) will pass through the resin column in wave form at known time intervals after injection of the solution to be analyzed. Cations are removed by the suppressor column resulting in the formation of the corresponding acids from the anions in solution. Consequently, there will be a series of acids such as HCl, H_2SO_4, etc., passing through the conductivity cell in wave form, each at a different though known time interval after the injection of the solution to be analyzed. Conductivity of the solution at a specific time interval is then related to acid concentration for the anionic species which passes through at that time.

Information developed in testing performed by NUS indicates that 5-10 ppm chloride in the sample size used will result in a maximum HCl concentration of about 2 ppm after it is separated in peaks as it passes through ion exchange columns in the ion chromatograph. The 2 ppm HCl concentration used in the peroxide test is based on the reasonable assumption that chloride level in the primary coolant will generally be below 10 ppm.

c. Dissolved Hydrogen

A review and evaluation of the components of the gas chromatograph i.e., molecular sieves, thermal conductivity cell, etc., indicates they are relatively insensitive to radiation damage. There are no radiation effects anticipated.

d. Dissolved Oxygen Determination

Readout equipment for this system is located remotely. The equipment exposed to radiation contained in the oxygen and temperature detection probe includes the following:

- Gold cathode and silver anode
- Membrane made of 2 mil Teflon
- Electrolyte - saturated KCl
- Two metal oxide thermistors
- Connecting wires to anodes
- Synthetic rubber O-ring seals
- Stainless steel probe holder
- PVC block and Hysol Epoxy cement

The most sensitive component to radiation damage in the above list is the two-mil thick Teflon membrane. Unpublished data from Oak Ridge indicates that Teflon is resistant to radiation damage to about 10^6 rads. This data is supported by irradiation tests performed in connection with the aircraft nuclear propulsion (ANP) program. Results of this work indicate that high pressure (1400 psi) Teflon hoses would show incipient leakage at 10^6 rads exposure levels. Cumulative exposure levels for the Teflon membrane during post-accident conditions would be less than 10,000 rads. Thus, there is a safety factor of at least 100x associated with this component. It should be noted that the probe or the membrane itself can be easily replaced with very minimal exposure after the system is flushed, drained and blown dry. The two anodes and stainless probe holder are not sensitive to radiation other than perhaps high-intensity neutron irradiation. There will be no neutrons present in the analytical system. The two metal-oxide thermistors are not expected to be affected by radiation; however, no problem would result if damage occurred. The temperature indication provided by the thermistors is not a critical measurement.

No radiation damage is anticipated to the PVC block below 10^6 rads based on data concerning the general resistance of organics to radiation. The Hysol epoxy cement is stable to radiation exposure levels to 10^7 rads. Concerning other components (O-ring seals, insulation), the cumulative exposure levels anticipated are far below where damage will occur.

e. pH Determination

Readout equipment for this system is located in the HRSS CAP. The equipment exposed to radiation includes the following:

- Glass pH probe - Gel type with internal reference cell
- Connecting wires to probe
- Synthetic rubber O-ring seals
- Stainless steel probe holder

Experimental work reported in the literature indicates that pH electrodes of the type used in this system are stable at cumulative dosages up to 10^7 - 10^8 rads. These results are in agreement with data presented in other references and experience reported by Battelle and the naval test facility in Idaho. Concerning selective ion electrodes, it was observed that the potential response of a fluoride electrode shifted with time at very high cumulative exposure; however, the affected electrode did not lose its effectiveness as long as calibrations were made periodically with standard solutions. A nitrate electrode began to give erratic readings as the cumulative dose approached 10^7 rad.

Cumulative exposure levels to the pH probe in the automated analysis panel during post-accident conditions will be less than 10,000 rads. Thus, there is a factor of at least 1000X below the point where incipient radiation damage can be anticipated. It is worthy of note that the data reported in the literature indicate that valid pH readings can be obtained even where a radiation effect was noted, provided the system is calibrated periodically. Operating instructions to be included with the automated analysis equipment will specify periodic calibration when the system is in regular use. Other components within the pH assembly will not be affected by the radiation exposure levels involved.

f. Conductivity Determination

Readout equipment for this system is located remotely and thus will not be exposed to significant radiation levels. The equipment exposed to radiation includes the following:

- Platinum electrodes enclosed in a glass holder
- Connecting wires to the electrodes
- Stainless steel probe holder
- Synthetic rubber O-ring seals

Operating experience at Oak Ridge indicates that conductivity measurements can be made in hot cell conditions under very high radiation levels without problem. Battelle-Northwest could not recall an experience in measuring conductivity under hot cell conditions. Oak Ridge has noted that ruthenium in solution can plate out and poison the electrodes. This is a very slow process occurring only in solutions having a much higher radiation level than would be present under post-accident conditions.

Incipient damage to the insulation on the connecting wires to the platinum electrodes could be anticipated at about 10^7 rads based on information presented in the literature. Total radiation exposure of this insulation during post-accident conditions would probably not exceed 1000 rads and would certainly be less than 10,000 rads. Thus, the insulation on the wire has a 1000X factor of safety.

The synthetic rubber O-ring seals could suffer incipient damage at about 10^6 rads based on data listed in the literature. Total radiation exposure for these O-ring seals during total post-accident conditions would be less than 10,000 rads resulting in a 100X safety factor. The glass and stainless steel would not be affected by the radiation. Glass will darken with time on exposure to radiation; however, this would not affect its performance in this application.

In the absence of dissoved hydrogen in the primary coolant, it is possible that there will be low ppm concentrations of peroxide present from radiolysis of the water. Testing was performed to determine if this peroxide would have an effect on the conductivity of boron-containing waters. No effect was noted on boric acid solutions with peroxide concentrations of 100 and 200 ppm. A 1 percent increase in conductivity was observed for basic boron solutions containing 100 and 200 ppm peroxide.

g. Pressure Indicator

A pressure transducer incorporating a strain gauge is used to sense pressure and transmit this pressure to the readout equipment which is located remotely. Only the transducer is exposed to radiation. This contains the following material:

- Diaphragm made of 17-4 pH stainless steel
- A strain gauge made of silicon "doped" with phosphorous
- Synthetic rubber O-ring seal
- Circuit board with resistors, capacitors and transistors
- Die-cast aluminum housing for electronic components
- Insulation on lead-in wire

The most sensitive components to radiation damage in the pressure transducer are probably the transistors. These should be resistant to cumulative exposure above 10^5 rads based on data presented in the literature. Maximum cumulative exposure for the transistors will be below 10,000 rads. There are no data available concerning radiation resistance of the strain gauge. The principle of operation for a strain gauge involves a change in resistivity that is proportional to the deflection in the gauge. It is very unlikely that resistivity of an element will be affected by radiation exposures that will be experienced in the pressure-transducer during a post-accident condition.

All other components in the pressure transducer can withstand cumulative exposure on the order of 10^6 rads or more.

Operation of the pressure indicator is not required for any of the post-accident analyses performed with the NUS system. The pressure indicator is included so that operating personnel can maintain a back-pressure 40-50 psi or more in the system by suitable adjustment of the throttling valve. Back-pressure is required to keep dissolved gases in solution. Adjustment of the throttling valve to control back-pressure will be made during calibration of the system performed during normal power operations. Flow adjustment will be required only infrequently since inlet pressure will be relatively constant at about 60 psi. In the event that there were a significant change in pressure and the transducer failed, the change in pressure would be manifest by a change in flow rate. There can be no buildup of excess pressure since the system is pressure-relief-valve-protected with valves in the Sentry system.

h. Valves

All valves use Teflon packing. The use of Teflon packing is specified because of its good sealing characteristics. Oak Ridge prefers Teflon-packed valves for handling radioactive solutions containing high levels of radioactive iodine. Some radiation damage to the Teflon may be anticipated at 10^6 rads cumulative exposure; however, it is unlikely these valves will ever have 10,000 rads exposure. It is recognized that there are packing materials available that are more resistant to radiation than is Teflon but these were not specified because the primary concern is to maintain leak-tightness.

TESTING AND TRAINING

The PASS instrumentation is expected to be tested every 6 months. Plant chemistry technicians and management personnel have successfully completed an initial training program on the operation of the Post Accident Sampling System.

The training program was developed and implemented by our consultant, NUS. The initial course is one week in duration and consists of classroom lectures followed by a walkthrough of all the PASS procedures. Present plans are to provide refresher training for selected individuals every six months consistent with plant operability.

TABLE 10-1

ANALYSIS	REQUIREMENT		RESPONSE			
	Range	Accuracy	Method Selected	Range	Accuracy	Testing Method
Boron	0-2000 (PWR)	±5% > 1000 ppm ±50 ppm < 1000 ppm	Diluted Sample Using Mannitol Titration Method	60-6000	±1% FS 60-2000 ppm(1) ±10% FS 2000-6000 ppm	matrix
Chloride	0-20 ppm	±10% 0.5-20 ppm ±0.05 ppm < 0.5 ppm	Ion Chromatography	0.1-20	7.6% 0.1-1 ppm(2) 5.8% 1-20 ppm	matrix
Hydrogen	0-2000 cm^3/kg	±10% 50-2000 cc/kg ±5 cc/kg < 50 cc/kg	Gas Chromatography	0.5-2000 cc/kg	7.1% 3.1-28 cc/kg 8.8% 21.7-186 cc/kg 7.6% 186-930 cc/kg 2.8% 930-1860 cc/kg	non-matrix
Oxygen	0-20 ppm	±10% > 0.5 ppm ±0.05 ppm < 0.5 ppm	YSI in-line probe Rexnord in-line probe(3)	0.1-20 ppm	8.2% 0.38-8.0 ppm 5.7% 0.1-10 ppm	matrix
pH	1-13	±0.3 pH > 5 < 9 ±0.5 pH < 5 > 9	in-line probe	1-13	± 0.3 pH 5.0 < pH < 8.0	matrix
Conductivity	---	----	in-line probe	0.1-500 mho/cm	±5.4% at 29.6 mho/cm	matrix

(1) Based on operating temperature being within 10°F of the calibration temperature.

(2) % error is the mean absolute bias

(3) The Rexnord probe for routine (non-accident use) is installed parallel to the YSI probe.

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

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.

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.

CON EDISON RESPONSE

11(a) The IP2 PASS sample lines and panels can be purged for several minutes at approximately 2 liters/minute to ensure representative samples will be obtained. The high velocities at this flow rate are expected to flush loosely adherent particulates from the sample lines. For RCS liquid, the purge flow can be directed to the Chemical Drain Tank No. 21 during normal operation and for a post accident condition, to the shielded HRSS Waste Collection Tank or to the Containment Sump. The PASS equipment location was primarily determined by considering ties to existing sample lines, available areas for equipment installation and maintaining personnel exposures ALARA. In consideration of the above, each sample line of the PASS was kept as short as possible to limit the volume of fluid needed to be taken from the various sample systems.

Additional RCS sample connections are available for more flexibility in selecting sampling points. For post accident conditions, the Reactor Coolant liquid sample may be taken from the RCS hot legs nos. 21 or 23, the recirculation pump discharge or the RHR loop. Sampling from the latter two systems provides a means to sample the RCS should the hot legs become vapor bound.

RCS lines have passive flow restrictions and redundant isolation valves outside the reactor containment which trip closed upon generation of the containment isolation signal. The isolation valves are environmentally qualified to meet the anticipated harsh condition during and following an accident.

Containment atmosphere is sampled from the combined containment recirculation fans discharge. These multiple sample sources from the five (5) fans will provide a representative sample, even in the event of a blockage at one of the sampling points. The sample lines are routed from the point inside containment to the Comsip-Delphi H_2/O_2 monitors and back to the containment. The sample lines are heat traced from where the lines penetrate containment up to the H_2/O_2 monitors. The heat trace can maintain a temperature of $285^{\circ}F$ on the sample lines. Heat tracing reduces plateout of Iodine and prevents condensation in the sample lines.

Redundant atmosphere sample lines are provided. This back-up capability ensures sampling of the containment atmosphere in the event one of the lines is rendered non functional due to blockage or damage.

11(b) Both the RCS sampling station at the Sentry HRSS equipment and the containment atmosphere sampled at the Comsip Delphi Hydrogen and Oxygen Monitors have HEPA and Charcoal absober ventilation exhaust systems.

REVISIONS

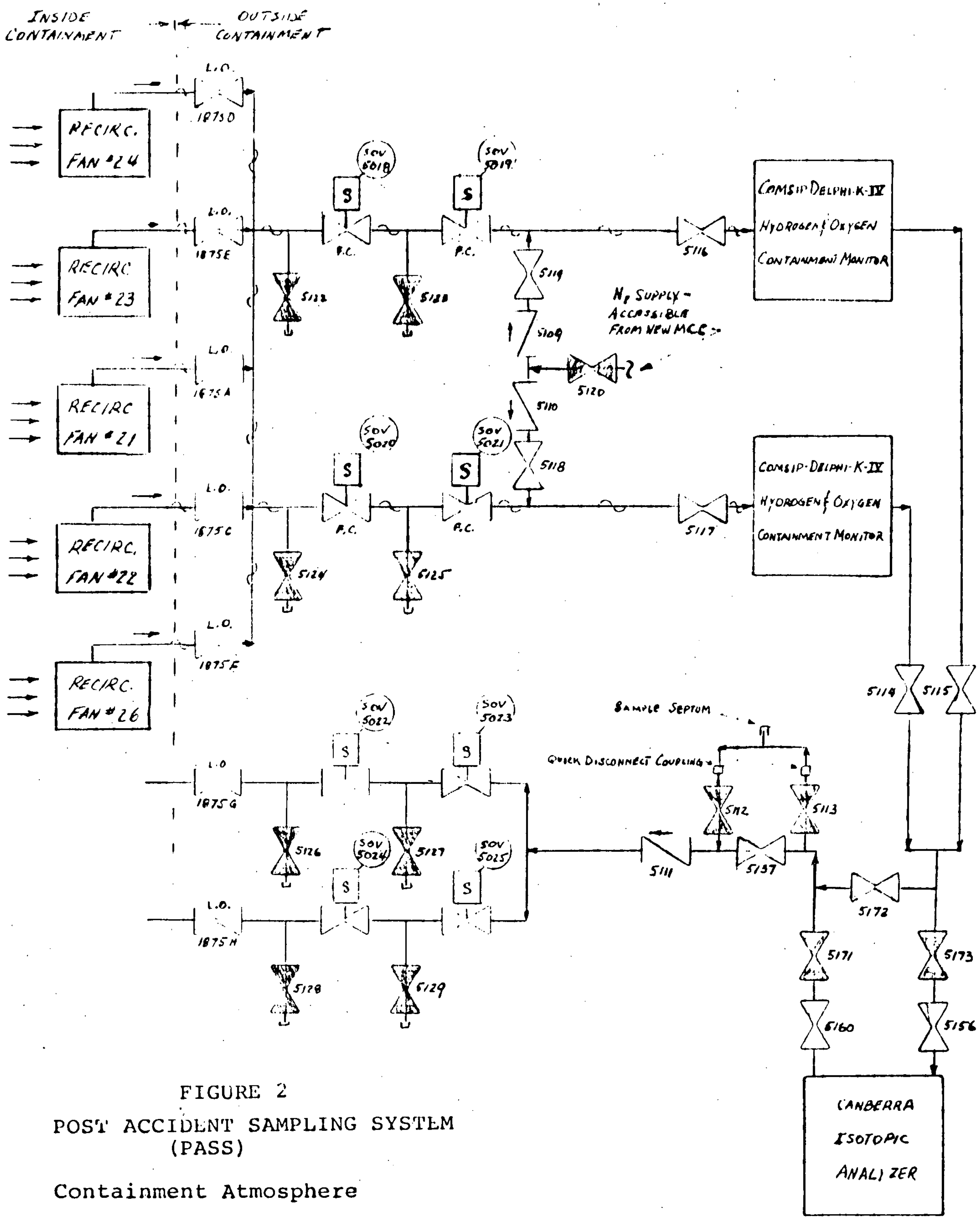


FIGURE 2

POST ACCIDENT SAMPLING SYSTEM
(PASS)

Containment Atmosphere

STATION-

DRAWN BY

SCALE ESM/ESR No.

APPROVALS

SUB-SECT. ENGR.

ENG.

DATE

Con Edison

ENGINEERING
SKETCH No.

Attachment B

Indian Point Unit No. 2

Interim Procedure for Assessment of Core Damage Following a Postulated Accident

Procedure No. NEP-1

Table of Contents

1.0 PURPOSE

To provide a methodology to determine the extent of core damage following a postulated accident based on radionuclide concentrations in reactor coolant (RCS) and containment atmosphere (VC) samples as well as other plant indications and parameters.

2.0 DATA PREREQUISITES

The information described in the following paragraphs will be utilized in assessing the core damage condition. Where information (other than measured concentrations) is not fully available and cannot be obtained in a timely manner, attempts should be made to conservatively estimate those values, with the use of such estimates noted in all discussions and evaluations of results.

2.1 RCS and VC samples have been taken and analyzed (with appropriate isotopic breakdown) in accordance with established post accident sampling procedures. The isotopes shown in the upper section of Table 2 should be given primary consideration in this assessment. The appropriate source of the liquid sample will depend on the particular accident scenario. The normal post accident sample path is from the Recirculation Pump discharge line. Additional sampling paths are available from the Residual Heat Removal Pump suction line and the RCS Loops 21 and 23 hot legs.

2.2 The following additional information has been obtained:

2.2.1 The reactor coolant temperature at the time the sample is taken.

2.2.2 Temperature of the RCS sample.

2.2.3 Pressure and temperature of both the containment atmosphere (at time of sampling) and the corresponding sample.

2.2.4 The volume of emergency core cooling water injected into the primary system.

2.2.5 The amount of dilution performed by the operator during the sampling process.

2.2.6 The elapsed time from reactor shutdown to sample analysis.

2.2.7 Power history data for current cycle and Effective Full Power Days (EFPD) for two previous cycles.

3.0 QUALITATIVE ASSESSMENT OF CORE DAMAGE

Where plant parameters indicate an abnormal plant shutdown has occurred with core cooling jeopardized or interrupted and possible core damage, a preliminary determination of the extent of core damage can be made by qualitatively reviewing the analyzed sample and evaluating it together with other plant parameters and operating data. Based on the following indicators, a general core damage assessment can be made:

3.1 Cladding Damage Indicators

3.1.1 The appearance of noble gases (xenons, kryptons), iodines and possibly small amounts of cesium in the reactor coolant without the presence of other fission products is a fair indication that damage is limited to clad failure and possibly, a limited amount of fuel overheating.

3.1.2 Additional indicators of clad damage include:

3.1.2.1 Core exit thermocouple temperature readings higher than 650°F (For LOCA with normal Engineered Safeguards response, the maximum expected core exit temperatures with natural circulation should be about 620 - 650°F).

NOTE: Thermocouple readings should be used only as confirmatory information for core conditions beyond clad damage since coolant conditions (i.e. steaming) can significantly affect thermocouple accuracy.

3.1.2.2 A significant increase in Reactor Coolant System conductivity.

3.1.2.3 High Range Containment Monitors in the following ranges [where release to Containment has been verified by other parameters (i.e. pressure, temperature, humidity)]:

$10 \text{ mR/hr} \leq R10 \leq 330 \text{ mR/hr}$

$9.6 \text{ R/hr} \leq R25 \leq 3.9 \times 10^3 \text{ R/hr}$

$15 \text{ R/hr} \leq R26 \leq 4.5 \times 10^3 \text{ R/hr}$

3.2 Overtemperature Indicators

3.2.1 The presence of an appreciable cesium concentration will be indicative of at least a fuel overheat situation since no substantial quantity of cesiums should be found if core temperatures remain below 2370°F or if the core has not been at least partially uncovered for some extended period of time. In addition, strontium and barium may be present.

3.2.2 High Range Containment Monitors read:

R10 $>$ 330 mR/hr
R25 $> 3.9 \times 10^3$ R/hr
R26 $> 4.5 \times 10^3$ R/hr

3.2.3 Reactor Vessel Level Indicating System indicates core uncovery (Reactor Vessel Level $<$ 3.5 feet) for an extended period of time.

3.2.4 Significant releases (e.g. greater than 100 uCi/cc) of tellurium, ruthenium and more refractory materials will occur only if the temperature approaches the fuel melting point ($\approx$ 4500 - 5000°F). The presence of ruthenium and tellurium does not "prove" melting, but their absence is a good indicator that melt has not occurred.

3.3 Fuel Melt Indicators

3.3.1 Presence of cerium and lanthanum in fluid samples is generally indicative of fuel melt. Ruthenium and tellurium must also be present (although as stated in 3.2.4, their presence does not prove melt).

3.3.4 High Range Containment Monitor readings approach the following values:

R10: 74 R/hr
R25: 4.7×10^5 R/hr
R26: 5.5×10^5 R/hr

4.0 ASSESSMENT OF CORE DAMAGE FROM SAMPLE ANALYSIS

The following procedural steps address more detailed characterization of core damage based on results of sampling and analysis of RCS and VC samples. The calculations are based on characteristics of a reference sample corrected for actual sampling conditions. Containment atmosphere sampling is of importance only for "line-break" type accidents.

4.1 Application and Determination of Correction Factors

The "reference sample" assumes the following characteristics:

- 1135 effective full power days (EFPD) of operation (36,000 MWD/MTU)
- Dilution of the liquid source term in Reactor Coolant System volume only (non-line break accident)
- No additional dilution during the sampling and analysis process
- The RCS sample at RCS temperature and pressure (and therefore density)
- The VC sample at containment temperature and pressure
- Samples taken at time of shutdown

To correct for deviations from the above assumptions under specific sampling conditions, the following correction factors should be applied as appropriate to the measured activity. The correction factors should be used for both the RCS and VC samples unless otherwise indicated.

4.1.1 Operating Time Correction Factor (P)

This correction factor is applied to correct for operation for less than the 1135 days of effective full power operation assumed in the formulation of the reference "maximum concentration" (MX) values shown in Table 2. For long lived isotopes this correction factor may be significant. For short lived isotopes, this factor may be neglected if the reactor has been continuously operating prior to this shutdown for a period of seven or more half lives.

$$P = \frac{1-e^{-\lambda_i 1135}}{(1-e^{-\lambda_i T_o}) + 2/3[(e^{-\lambda_i t_1})(1-e^{-\lambda_i T_1})] + 1/3[(e^{-\lambda_i t_2})(1-e^{-\lambda_i T_2})]}$$

where: T_o = operating time for current cycle (EFPD)
t_1 = time since end of last cycle (days)
T_1 = operating time for last cycle (EFPD)
t_2 = time since end of cycle before last (days)
T_2 = operating time for cycle before last (EFPD)
λ_i = decay constant for isotope "i" (days^{-1})

If a plant shutdown has occurred during the current cycle, the first term in the denominator may have to be adjusted to account for this, treating the portion of the cycle prior to the shutdown as if it was a previous cycle. The first term then becomes:

$$\left\{(1-e^{-\lambda_i To})+[(e^{-\lambda_i to'})(1-e^{-\lambda_i To'})]\right\}$$

where: To = operating time since last restart (EFPD)
To' = time since last shutdown (days)
to' = operating time prior to shutdown during current cycle (EFPD)

4.1.2 Reduced Power Correction Factor (Y)

This factor is applied if the reactor had been operating during its last period of operation at a reduced power level.

$$Y = \frac{100}{\text{\% Full Power (@ time of shutdown)}}$$

4.1.3 Correction for Additional Dilution Provided by Emergency Core Cooling System (ECCS) Volume (ED)

For accidents involving injection of emergency core cooling water into the primary system, the assumed dilution volume for the RCS sample must be corrected. The volumes which may need to be added will depend on the specific accident scenario and may typically include the accumulators, boron injection tank and Refueling Water Storage Tank. The actual volumes can be determined from appropriate level and flow instrumentation.

$$ED = \frac{91{,}600 + \text{ECCS Volume Injected (gallons)}}{91{,}600}$$

4.1.4 Correction for Dilution During Sampling Process (SD)

The "reference sample" is an undiluted sample using the in-line isotopic analyzer or drawn from the High Radiation Sampling System or Containment Atmosphere sample point. A Sampling Dilution (SD) correction factor must be applied if the operator further dilutes the drawn sample.

SD = Dilution Ratio
(e.g. SD = 1000 for a 1000: 1 dilution)

4.1.5 Density Correction Factor (R)

This is a straightforward correction if the density of the sample differs from that of the RCS or Containment. The correction factor (R) is taken from Table 1 for the RCS sample.

For the Containment atmosphere sample this factor can be determined by:

$$R = \frac{(Pvc)\ Ts}{Ps\ (Tvc)}$$

where: Pvc = Containment pressure (psia)
Ps = Sample pressure (psia) during analysis
Tvc = Containment temperature (^{O}R)
Ts = Sample temperature (^{O}R) during analysis

4.1.6 Decay Correction Factor

This corrects the measured fission product concentration (MC) for decay from reactor shutdown to the time of analysis.

$$MC_o = MC\ (e^{-.0417\,\lambda t})$$

where: MC_o = decay corrected measured concentration
MC = uncorrected measured concentration
t = time (hrs) from shutdown to analysis
λ = decay constant (See Table 2) (days $^{-1}$)
.0417 = conversion factor (hours to days)

4.2 Determination of "Adjusted" Measured Concentration

Having determined and calculated all appropriate correction factors, the "adjusted" measured concentration AC is given by:

$$AC = [MC_o \cdot P \cdot Y \cdot ED \cdot SD \cdot R]_{RCS} + [MC_o \cdot P \cdot Y \cdot SD \cdot R]_{VC}\ \frac{(VC\ volume)}{(RCS\ volume)}$$

where: AC = Adjusted total measured concentration. This includes both RCS and VC concentration and will be compared to the "reference" sample concentrations (MX's)

$[\ \ ]_{RCS}$ = Adjusted Reactor Coolant System concentration

$[\ \ ]_{VC}$ = Adjusted Containment atmosphere concentration

$\frac{(VC\ volume)}{(RCS\ volume)}$ = This term adjusts the containment atmosphere concentration to allow it to be directly added to the RCS concentration.

This concentration should be determined for each of the isotopes in the upper section of Table 2 (or as many as possible). These adjusted concentrations should then be compared to the reference concentrations (MX values) given in Table 2 as follows:

4.3 Clad Damage Estimate

To estimate the fraction of clad failure damage:

4.3.1 Determine the maximum clad failure concentration, MX_{CFi} from Table 2, Column I for each selected isotope "i".

4.3.2 Calculate a Clad Failure Fraction (CFF):

$$CFF = \frac{ACi}{MX_{CFi}}$$

where AC_i = adjusted concentration for isotope "i"

4.3.3 Criterion for Cladding Failure

4.3.3(a) If $CFF < .01$ there is little or no cladding damage. Verify that MC is within normal concentration measurements.

4.3.4(b) If $.01 \leq CCF \leq 1.0$ there is some degree of clad damage. To estimate percentage of damage, multiply CCF by 100.

4.3.5(c) If $CCF > 1.0$ proceed to next paragraph to estimate damage.

4.4 Fuel Overtemperature Estimate

To estimate the degree of fuel overtemperature:

4.4.1 Determine the maximum fuel overtemperature concentration, MX_{FOi}, from Table 2, Column II for each selected isotope "i".

4.4.2 Calculate a Fuel Overtemperature Fraction (FOF):

$$FOF = \frac{ACi}{MX_{FOi}}$$

where AC_i = adjusted concentration for isotope "i"

4.4.3 To estimate the percentage of the core in an overtemperature condition multiply FOF by 100. If $FOF > 1$, proceed to the next paragraph to evaluate for core melt.

4.5 Fuel Melt Estimate

To estimate the fraction of the core which may have experienced core melt:

4.5.1 Determine the maximum fuel melt concentration, MX_{FMi} from Table 2, Column III for each selected isotope "i".

4.5.2 Calculate a Fuel Melt Fraction (FMF):

$$FMF = \frac{ACi}{MX_{FMi}}$$

where AC_i = adjusted concentration for isotope "i"

4.5.3 To estimate the percentage of fuel melt, multiply FMF by 100. Note the presence of ruthenium and tellurium as per 3.3.1.

5.0 CAUTIONS AND RECOMMENDATIONS

If conflicting data exists based on criteria guidance given in Sections 3 and 4, reanalyze all indications. For example; the analysis yields results which indicate core melt. The core thermocouple readings show the core is not hot enough to have melted. The sample analysis should therefore be re-examined to determine if the isotopics chosen are appropriate.

It should also be noted that the above evaluations address an assumed uniform distribution of core damage. Since the degree of damage is likely to vary within the core, calculations for CFF, FOF and FMF should all be performed to provide a better understanding and perspective with regard to the potential existence of a mixture of fuel damage conditions. Variations in core exit thermocouple readings may supply additional supporting information.

Finally it should also be noted that the results determined using this procedure are subject to inaccuracies related to physical processes occurring in the RCS and VC in the post accident condition. These processes, including among others, adsorption, sedimentation and plateout, may remove a significant amount of source term from both the atmosphere and fluid from which the sample is drawn. Although inclusion of these processes is beyond the scope of this evaluation, their potential effect should be borne in mind.

6.0 REFERENCES

5.1 Reactor Safety Study (WASH-1400)

5.2 Three Mile Island - Report to the Commissioners and to the Public ("Rogovin Report")

5.3 W Mitigating Core Damage Training Manual, Sections 6 & 8

5.4 Westinghouse Radiation Analysis Design Manual

5.5 ORIGEN (Isotope generation and depletion code)

TABLE 1

DENSITY CORRECTION FACTORS (R)

RCS Temperature at time of Sample (°F)	RCS Sample Temperature (°F)			
	70	80	90	100
100	.995	.996	.998	1.0
150	.982	.983	.985	.987
200	.965	.966	.968	.970
250	.944	.945	.947	.949
300	.920	.921	.923	.924
350	.892	.894	.895	.897
400	.861	.862	.864	.865
450	.826	.827	.828	.830
500	.787	.788	.789	.791
550	.737	.738	.739	.742
560	.726	.727	.728	.730
570	.716	.717	.718	.720
580	.704	.705	.706	.707
590	.692	.693	.694	.695
600	.680	.681	.682	.683
620	.650	.651	.652	.653
640	.617	.618	.619	.620
660	.579	.580	.582	.582
680	.528	.529	.530	.531
700	.438	.438	.439	.441

TABLE 2

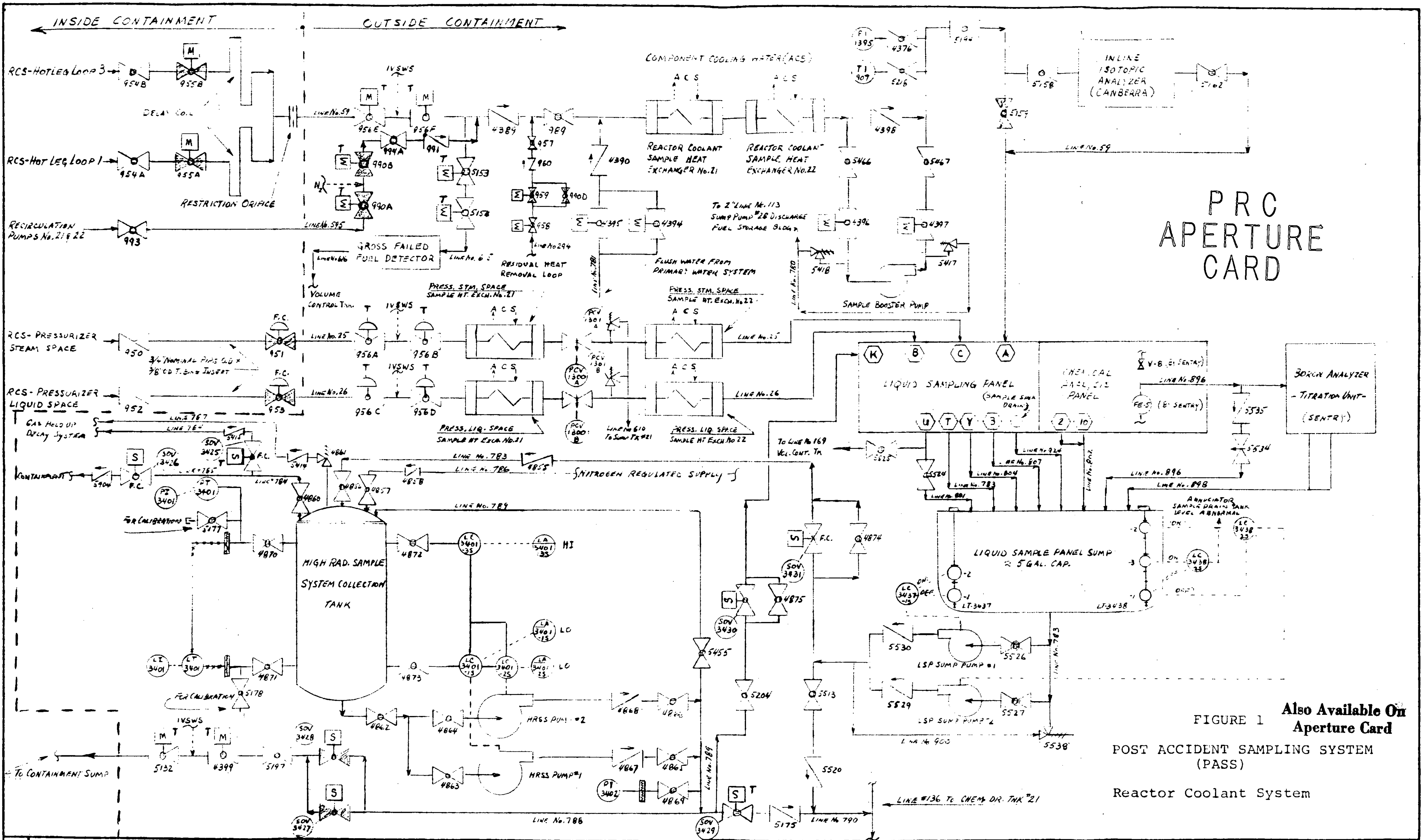
Isotope	Half-Life[a]	I MX_{CF} (Ci/cc)	II MX_{FO} (Ci/cc)	III MX_{FM} (Ci/cc)
* Isotopes For Primary Consideration *				
I-131	8.05d	4.29E-3	1.26E-1	2.26E-1
I-133	21h	7.81E-3	2.30E-1	4.14E-1
Kr-85	10.76y	9.25E-5	1.54E-3	2.77E-3
Xe-133	5.27d	1.38E-2	2.30E-1	4.14E-1
Cs-134	2.05y	4.16E-3	3.33E-2	6.67E-2
Cs-137	30y	1.60E-3	1.27E-2	2.55E-2
Te-132	78h	NA	1.73E-2	5.19E-2
Ba-140	12.8d	NA	8.22E-3	4.11E-2
Mo-99	66h	NA	4.38E-3	1.31E-2
Ce-141	32.5d	NA	NA	1.16E-3
* Additional Isotopes *				
Sr-89	52.7d	NA	3.63E-3	1.82E-2
Sr-90	27.7y	NA	4.22E-4	2.11E-3
Te-134	72m	NA	2.30E-2	6.89E-2
La-140	40.22h	NA	NA	1.27E-3
Ce-144	284d	NA	NA	9.25E-4
Ru-103	39.6d	NA	3.70E-5	1.11E-3
Ru-105	4.44h	NA	3.03E-5	9.08E-4
Ru-106	367d	NA	1.85E-5	5.54E-4
Ru-107	4.2m	NA	1.90E-5	5.69E-4
Ru-108	4.5m	NA	1.33E-5	3.99E-4

(a)
$\lambda = .693/T^{1/2}$
where:
λ = decay constant (days^{-1})
$T^{1/2}$ = half life (days)

NA : Not Applicable

STATION

ENG. SK. No.



PRC APERTURE CARD

FIGURE 1

POST ACCIDENT SAMPLING SYSTEM (PASS)

Reactor Coolant System

Also Available On Aperture Card

POOR ORIGINAL

REVISIONS

STATION-

DRAWN BY

SCALE

ESM/ESR No.

APPROVALS

SUB-SECT. ENGR.

ENG.

DATE

Con Edison

ENGINEERING SKETCH No.

BRUNING 40-53 185

8811290128-01