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SUBJECT: Forwards schedule for providing info re NUREG-0737,Item II.B.3, "Post-Accident Sampling Capability," in response to NRC 820726 ltr.W/three oversize drawings.Aperture cards are available in PDR.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

December 30, 1982

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit #1
Docket No. 50-220
DPR-63

Dear Mr. Eisenhut:

Your letter of July 26, 1982 requested a schedule for providing information relative to NUREG 0737, Item II.B.3 Post Accident Sampling Capability. Our letter of August 13, 1982 as amended by our December 20, 1982 letter indicated the information would be provided by December 28, 1982. Contained herein is the requested information.

Very truly yours,

C. V. Mangan
Vice President - Nuclear Engineering
and Licensing

CVM/RJP:bd

A046

Response to Request for Information
Post Accident Sampling System

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

Response (1) and General Description of Niagara Mohawk's Post-Accident Sampling Capabilities

The newly installed Post Accident Sampling Sytsem permits remote sampling of reactor water and reactor water dissolved gas within a short period of time after a Loss of Coolant Accident condition and with relatively low resultant personnel radiation exposure. The basic system consists of: (1) a piping station located inside the Reactor Building at elevation 281, (2) a sampling station located outside the Reactor Building on Turbine Building elevation 277, (3) a control panel situated about 25 feet from the sampling station, (4) assorted transport equipment including shielded lead caves and (5) a ventilation system. Process Survey Procedure N1-PSP-13 details operation of the system. Attachments 1, 2 and 3 of this letter show pertinent piping and instrumentation diagrams.

Using the Post Accident Sampling System, samples are obtainable from two sources within the primary reactor vessel: (1) the liquid poison sparger and (2) number 11 recirculation loop. Valve actuation and sample delivery to the Turbine Building are controlled at the sample station control panel although primary system isolation valves are operated from the main control room. The system is powered by an emergency diesel backup power board. Therefore, a sample can be obtained during a loss of off-site power. However, as indicated in our letter of April 1, 1982, the emergency powered post-accident sampling ventilation system exhausts to the non-emergency powered Turbine Building ventilation system prior to discharge to the stack. A post accident sample can still be obtained without ventilation provided Emergency Director authorization is received beforehand.

Response (1) (cont'd)

At the sample station, the sample can be circulated through various liquid loops equipped with pressure, temperature, flow and specific conductivity sensing elements. Rapid assessment of both general area and sample dose rates during the sampling process can be achieved by reference to local radiation detector RE507 or in-line radiation detector RE665 respectively. Local indicators for the aforementioned devices are located at the sample station control panel. Flushing provisions are also provided at the sample station so that: (1) personnel exposure is minimized during sample station approach and (2) repairs may be performed at the sample station if a breakdown occurs during sampling.

Both diluted (0.1 ml reactor water) and undiluted (10.0 ml reactor water) liquid samples are obtainable at the sample station and are collected in small septum bottles. Sample handling time and dose (see Table 1 for estimates) are minimized by using a remotely-operated, shielded, mechanical arm which draws the septum bottles into specially constructed lead caves. A 5 inch lead lined, undiluted liquid sample cave is mounted on a four wheel dolly for transport to the 261 elevation laboratory complex approximately 250 feet away.

It is also possible to obtain a reactor water dissolved gas sample at the sample station by depressurizing an isolated liquid loop and collecting the resultant gas phase in an evacuated vial. Using a one foot long vial holding tool, the vial can be placed into a 1 1/2 inch thick, lead lined carrying cask in less than one minute and carried to the laboratory (see Table 1 for time/dose estimates).

In the event any radioactive gas leakage occurs during the sampling process, the sampling station is maintained under negative pressure by an independent ventilation system equipped with effluent pre-filters, charcoal and HEPA units.

Process Survey Procedure N1-PSP-11 describes a method available for sampling containment air. A previous Niagara Mohawk submittal (October 20, 1981, to Mr. Ronald C. Haynes, Director, USNRC, Region I) provided sampling time/dose estimates. Some data generated at that time is recopied on Table 1. Basically, the sampling process involves (1) connection of a portable sample rig to the drywell H_2/O_2 monitoring system on Turbine Building elevation 291, (2) obtaining a 15cc sample within eight minutes, (3) placing the sample (using a remote handling tool) in a 5 inch lead lined transport cave equipped with wheels and (4) transport to the high level laboratory approximately 280 feet away.

In an effort to minimize personnel exposure for containment air sampling a sample line from the H_2O_2 monitoring system is being connected to the post accident sampling station. After subsequent preoperational testing, procedural revisions and system training, the gas portion of the Turbine Building elevation 277 Post Accident Sampling System will be fully operational. This is expected by November 1983. Revised containment air sampling time/dose estimates will be made following pre-operational testing.

TABLE 1[1]

TIME AND DOSE PROJECTIONS FOR PAS SAMPLING, TRANSPORT AND ANALYSIS

TASK	APPLICABLE PROCEDURE	EST. SAMPLE CURIE CONTENT (CURIES)	EST. TIME FOR PERFORMANCE OF TASK[2] (MIN)	SAMPLE HANDLING TIME[3] (MIN)	MAX. WB DOSE RATES EXPOSED TO (mRem/hr)	PROJECTED WHOLE BODY DOSE[4] (mRem)	SENSITIVITY OF ANALYSIS[5]
Sample and Transport of undiluted Reactor Water Sample	N1-PSP-13	2.0	45	0	100	100[6]	--
Sample and Transport of diss. gas sample	N1-PSP-13	11.8[7]	45	1	1400	300[6]	--
Sample and Transport of containment air sample	N1-PSP-11	0.6	45	12	20000[8]	4000	--
Dilution of Undiluted Reactor Water Sample[9]	S-CAP-60	2.0	20	2	2400	100	--
Dilution of diss. gas sample	S-CAP-60	11.8	20	2	14000	520	--
Dilution of Containment Air Sample	S-CAP-60	0.6	20[10]	2	570	100	--
Isotopic Analysis of Diluted Sample	V.A.7N	7.3 mCi	10	0.5	100	100	20%[11]
Chloride Analysis of Undiluted Sample (25.0 ml)[12]	S-CAP-11	5.0	30	3	6250	310	0.020 ppm
Chloride Analysis of Diluted Sample (0.25 ml Undiluted Reactor Water)	S-CAP-11	0.5	30	3	100	100	2 ppm

TABLE 1[1] (cont'd)

TIME AND DOSE PROJECTIONS FOR PAS SAMPLING, TRANSPORT AND ANALYSIS

TASK	APPLICABLE PROCEDURE	EST. SAMPLE CURIE CONTENT (CURIES)	EST. TIME FOR PERFORMANCE OF TASK[2] (MIN)	SAMPLE HANDLING TIME[3] (MIN)	MAX. WB DOSE RATES EXPOSED TO (mRem/hr)	PROJECTED WHOLE BODY DOSE[4] (mRem)	SENSITIVITY OF ANALYSIS[5]
Boron Analysis of Undiluted Sample (20 ml)	S-CAP-9	0.4	60	2	500	100	0.050 ppm
Boron Analysis of Diluted Sample (0.02 ml)	S-CAP-8	4 mCi	60	2	100	100	5 ppm
pH Analysis	IV.A.16	1.0	5	2	1250	100	0.2 units
Oxygen Analysis of Dissolved gas (0.1cc)	IV.A.22[14]	0.08	40	1	1400	100	0.1 cc[13] (STP)/kg
Hydrogen Analysis	IV.A.22[14]	0.08	40	1	1400	100	3.4 cc[13] (STP)/kg
Oxygen Analysis of Containment Air Sample (0.1 cc)	IV.A.22[14]	4 mCi	40	1	100	100	0.2%[13]
Hydrogen Analysis of Containment Air (0.1 cc)	IV.A.22[14]	4 mCi	40	1	100	100	1.0%[13]

Notes for Table 1

1. See October 21, 1982 submittal to the NRC for source term assumptions, dose/dose rate calculation assumptions used in construction of this table.

2. The time for a preplanning meeting is not included.

3. "Sample handling time" includes time exposed to the unshielded source at a 2-3 foot distance.

4. Since it should never be necessary to hold a post accident sampling vial closer than 6 inches from extremities, extremity doses should never exceed six times whole body doses.

5. Stated sensitivities are at lower end of analytical range.

6. A "Shielding Review of Nine Mile Point Unit 1" predicted insignificant dose rates in the 277 elevation post accident sampling area. Thus, the expected source of personnel exposure during sampling would be from the sample itself and not from area background.

7. In the calculation of the curie concentration for post accident sampling dissolved gas samples, it was assumed that half of the activity contained in the 117.8 ml reactor water sample loop is stripped-out and collected during depressurization.

8. Source of the maximum dose rate exposed to are the sample lines, not the sample.

9. In lieu of performing this task, a diluted (100:1) reactor water sample can be obtained directly from the 277 elevation Post Accident Sampling System.

10. First step dilution only considered. Exposure and time factors associated with secondary dilutions would be significantly less. Also, note that this dilution is unnecessary unless an isotopic analysis of dissolved gas is desired.

11. Primary system isolation/negligible airborne activity in the laboratory area are assumed.

12. Since the 277 elevation Post Accident Sampling System delivers a 10 ml undiluted sample, performance of this analysis would require three liquid samples to be taken.

13. Stated sensitivities are calculated on the basis of a 0.1 ml sample volume. Use of a 1.0 ml volume should increase sensitivities tenfold.

14. Procedure in draft to be completed prior to Matrix testing.

Response (1) (cont'd)

Table 1 provides estimated handling times, analytical times and doses which would be received during dilution and analysis of post accident samples with activities commensurate with Regulatory Guide 1.3 Source Terms. It is evident from Table 1 that sampling, transport, dilution and analysis of parameters stated in Criterion (2) below can be achieved in a three hour time frame provided: (1) two sampling/analysis teams are used and (2) Chloride and Boron analyses are performed on diluted reactor water samples.

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

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

(b) hydrogen levels in the containment atmosphere;

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

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

CLARIFICATION (2): (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation 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.

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

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

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

Response (2)

Niagara Mohawk's October 20, 1981, submittal to Mr. Ronald Haynes listed estimated maximum activities permitting isotopic analysis (MAPIA) for several source-to-detector distances. Subsequent to the reference submittal, the following analytical/counting improvements for isotopic analysis of any post accident sample have been made:

- Dilution procedure S-CAP-60 has been instituted which allows for dilution of liquid or gaseous samples with specific activities higher than assumed Loss of Coolant Accident levels. Using source term assumptions found in our October 20, 1982 submittal to the Nuclear Regulatory Commission, a dissolved gas sample obtained using the 277 elevation Post Accident Sampling System could have an activity approaching 12 curies. Any required number of dilutions can be performed on post accident samples thereby eliminating any gamma spectroscopy dead time problems.

- Dilution equipment, including micro-pressure-lock syringes, a lead window, and a small sample holding cave have been purchased to reduce background radiation and personnel exposure during performance of analytical/dilution procedures.

- The laboratory gamma spectroscopy system has been calibrated at 100 cm distance from the detector which should permit direct analysis of small geometry sources with activities as high as 7.3 mCi.

A modification has been completed on the laboratory area ventilation system which should limit high airborne activity/background levels in the lab counting room during an accident. Details of the modification were provided in our December 31, 1982 letter to Mr. Darrell G. Eisenhut, Director Division of Licensing. As indicated in that submittal the laboratory area ventilation system discharges to the turbine building ventilation which is not powered from emergency power sources.

Niagara Mohawk does not presently have a formal procedure for relating radionuclide concentrations to core damage. However, we are participating in a generic review of this item by the Boiling Water Reactor Owner's Group. The current schedule calls for completion of the review by December 30, 1982. Additional information will be provided following receipt and review of that study.

As indicated in our December 31, 1981 letter to Mr. Eisenhut, on-line monitoring will not be used at Nine Mile Point 1 for post accident sampling. An exception is the H_2/O_2 monitoring system which provides the Control Room with a continuous readout of drywell H_2/O_2 concentrations. Sensitivities are on the order of 0.2 percent. Reliability of the instrument is maintained and verified quarterly by an in-house Instrument Calibration Procedure.

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

Response (3)

As stated in our December 31, 1981 letter to Mr. Eisenhut, isolated auxiliary systems are not required for post accident sampling of reactor water or containment air at Nine Mile Point 1.

Provisions for (1) sample recirculation back to the reactor vessel, or (2) "in-vessel" recirculation are not included in our Post Accident Sampling System. Adequate flushing of Post Accident Sampling System sample lines is accomplished by directing sample flow through the entire system and then to the torus.

Environmental suitability of Post Accident Sampling System piping/valving supplied by our equipment vendor is currently being investigated by the Boiling Water Reactor Owners Group. A report on the subject is expected by December 30, 1982. Additional information will be provided following receipt and review of that report.

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 sample is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.'

CLARIFICATION (4): 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 <0.1 ppm by measurement of a dissolved hydrogen residual of $\geq$ 10 cc/kg is acceptable for up to 30 days after the personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Response (4)

Laboratory Instrument Procedure IV.A.22, "Operation and Calibration of the Carle Instrument Analytical Gas Chromatograph" describes the methodology for analyzing H_2 and O_2 in reactor water dissolved gas samples and calculating reactor coolant system H_2/O_2 concentrations. By injecting 0.1 ml of dissolved gas into the gas chromatograph, 0.3 ppm (3.4 cc (STP)/kg) H_2 and 0.2 ppm (0.1 cc (STP)/kg) O_2 should be detectable. Increasing sample size to 1.0 ml could be used to increase instrument response tenfold, if required. Additionally, the purchase of an amplifying device, which would increase instrument reponse another order of magnitude, is now under consideration.

Niagara Mohawk has no plans at present to install any on-line dissolved oxygen monitors in the Post Accident Sampling System.

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

Response (5)

Using the colorimetric method found in Laboratory Instrument Procedure S-CAP-11, reactor water chloride concentrations as low as 2 ppm can be detected by analysis of a 100:1 sample dilution (see our December 31, 1981 letter to Mr. D. Eisenhut). Undiluted samples can be analyzed for chloride at concentrations as low as 20 ppb. Worse case exposure estimates for the analysis of both undiluted and diluted samples are shown on Table 1 and are well within GDC 19 Criterion. Appropriate shielding can be used during performance of S-CAP-11 to minimize personnel exposure.

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).

CLARIFICATION (6): Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Response (6)

The exposure estimates of Table 1 are not as yet based on actual person-motion studies. The current schedule calls for performance of these studies by May 1983. Revised exposure estimates should be available shortly thereafter.

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 (7): PWRs 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)

Chemical Analytical Procedure S-CAP-8 provides methodology for colorimetric determination of boron. This method is acceptable for use in post accident sampling boron analysis. Necessary procedural revisions to include reference to S-CAP-8 will be completed by March 1983.

CRITERION (8): If in-line 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 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 (8): A capability to obtain both diluted and undilted backup samples is required. Provisions to flush in-line monitors to facilitate access for repair is desirable. If an offsite 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)

As indicated in Response (1), the Turbine Building elevation 277 Post Accident Sampling System provides for grab sampling of both diluted and undiluted liquid samples. Furthermore, the system has sample line flush capabilities to facilitate access for repair after an accident.

Niagara Mohawk does not rely on an outside laboratory for grab sample analysis. However, the adjacent James A. Fitzpatrick Nuclear Power Plant (Power Authority of the State of New York) can assist in post accident sampling analysis if necessary. Additionally, their system is similar to Niagara Mohawk's so that spare parts can be rapidly obtained. Niagara Mohawk participates in the Pooled Inventory Management program which may be able to provide a post accident sample cask and transportation for offsite analysis of samples.

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 1u Ci/g to 10 Ci/g.

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

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

(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)

Our letter of October 20, 1981 to Mr. Ronald Hayes, Item (2) and Table 1 of this letter, provided the requested information. Dilution per S-CAP-60 should: (1) preclude any sample counting radiation effect (i.e., dead time) problems arising from high sample activities, and (2) permit sample counting within a normal (i.e., 20 percent) range barring high counting room airborne conditions.

Response (9) (cont'd)

Assuming (1) containment isolation, and (2) proper operation of the emergency laboratory ventilation system, predicted laboratory background levels during a Loss of Coolant Accident would be approximately 2 mr/hr based on a shielding review study. In the laboratory facility's 4 inch lead lined GeLi detector cave, background levels would be approximately 0.02 mr/hr; a level too low to significantly affect sample counting.

In the event containment isolation is not maintained and high airborne activity exists in the counting room, background levels higher than 3 mr/hr would not produce unmanageable detector dead time problems or greater than a 50 percent error in sample analyses. These facts are demonstrated in the attached laboratory report (Attachment 4).

Post accident sampling procedures do not presently address sample storage considerations. Procedural revisions will be made by June 1983 which specifically designate a sample storage area and methods of transport to the storage room. The storage location will be far enough away from the counting room so as to minimize background contributions to counting room equipment.

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 (10): 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 deradiation and corrosion potential of the coolant.

An accuracy of ± 10% is desirable between 50 and 2000 cc/kg but ± 20% can be acceptable. For concentrations 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 pH of 5 to 9, the reading should be accurate within ± 0.3 pH units. For all other ranges ± 0.5 phH units is acceptable.

CLARIFICATION (10): (cont'd)

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 Ioidide
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	Asborbed 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 were 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.

CLARIFICATION (10): (cont'd)

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

Response (10)

Post accident sampling boron analysis can be performed within the suggested analytical accuracy ranges using a diluted sample and Chemical Analytical Procedure S-CAP-8.

Post accident sampling chloride analysis can be performed on an undiluted reactor coolant sample within the suggested analytical/accuracy ranges using Chemical Analytical Procedure S-CAP-11.

Hydrogen in reactor water dissolved gas can be measured on the Gas Chromatograph using Laboratory Instrument Procedure IV.A.22 and a 0.1 ml sample within the suggested analytical/accuracy ranges.

Oxygen in reactor water dissolved gas can also be measured on the Gas Chromatograph using procedure IV.A.22 within the suggested analytical/accuracy ranges. However, a 1.0 ml sample injection (instead of a 0.1 ml) is necessary to achieve a $\pm$ 0.050 ppm accuracy at concentrations below approximately 1.0 ppm.

pH can be analyzed using a combination electrode and Laboratory Instrument Procedure IV.A.16 within the suggested accuracy ranges.

Niagara Mohawk has not yet verified that all post accident sampling analyses can be performed in the Standard Test Matrix. The present schedule calls for "Matrix Testing" to be conducted by March 30, 1983. Procedural revisions which may be necessary as a result of the "Matrix Testing" will be completed by the end of 1983.

Requirements for calibration of Post Accident Sampling System equipment (including conductivity equipment, radiation monitors, pressure, temperature and flow devices) are all included in Process Survey Procedure N1-PSP-13. Calibration of required laboratory instrumentation is done routinely or before each use.

To date, three site personnel have received thorough training on the 277 elevation Post Accident Sampling System. By June 30, 1983, two additional personnel will attend formal training sessions in all phases of post accident sampling, analysis and transport.

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

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

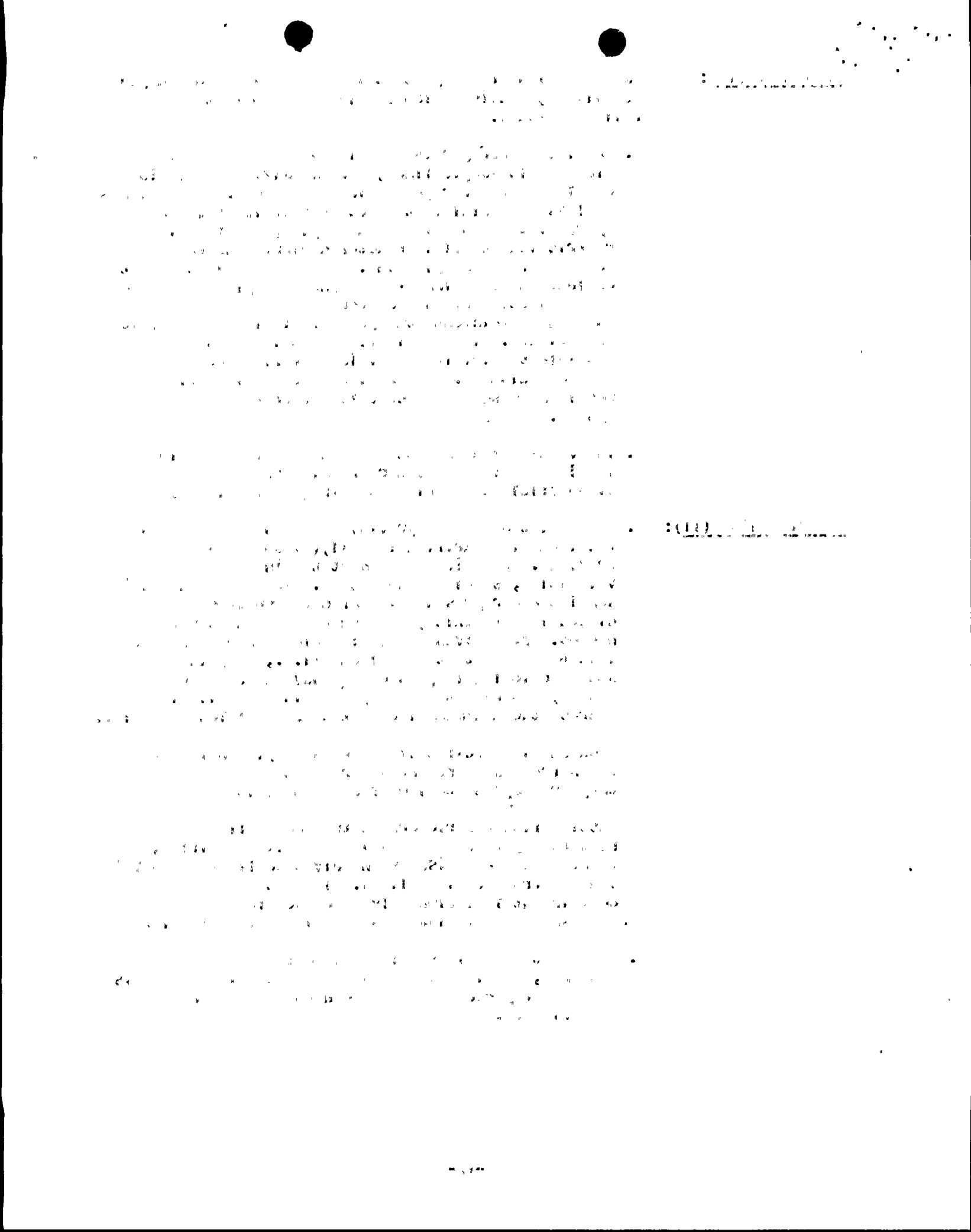
b. The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

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

BWRs 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.



Response (11)

Attachment 1 to this letter shows the Piping and Instrumentation diagram for the liquid post accident sampling sample lines. Sample pipe diameter used include 1 inch and 1/2 inch and total pipe run length is approximately 280 from the reactor to the elevation 277 Turbine Building sample station. All piping can be purged rapidly at near reactor pressure directly to a 3/4 inch line to the torus. In the event reactor pressure is low, two 3 gpm sample pumps are provided. Cumulatively, these provisions should effectively reduce plateout, minimize sample distortion, help prevent sample line blockage and preclude the necessity for heat tracing.

Sample line flow restrictors are unnecessary since the Post Accident Sampling System is equipped with Reactor Protection System activated isolation valves (see Attachments 1 and 3). The isolation valves are included in the ongoing Equipment Qualification effort which is applicable to safety related portions of this system. Environmental suitability of Post Accident Sampling System piping/valving supplied by our equipment vendor is currently being investigated by the Boiling Water Reactor Owners Group. A report on the subject is expected by December 30, 1982. Additional information will be provided following receipt and review of that report.

As stated in Response (1), the Post Accident Sampling System includes piping from two primary reactor vessel sources: (1) the liquid poison sparger and (2) number 11 recirculation loop. To ensure sample representativeness after an accident, sample flow can be established from each source to the post accident sampling station. At the sample station in-line radiation readings obtained on monitor RE 665 can be compared. If the readings do not correspond, mixing processes can be considered (see Reponse (2)).

The sample source redundancy described above should provide sufficient backup sampling capabilities. In the event that both of these sampling locations are nonfunctional, it may be possible to take grab samples from the reactor water sink (sample sources include number 11 recirculation loop, Reactor Clean-up System, or the Shutdown Cooling System - see Attachment 3) within the secondary containment.

A dedicated Charcoal and HEPA filtration system is provided. Effluent is routed to the stack via Turbine Building ventilation system at a point 53 feet upstream of the Turbine Building exhaust fans. See Response (1) and our December 31, 1981 submittal for additional details on the Post Accident Sampling Ventilation System.

CONCLUSIONS (continued)

Also plotted on Figure #1 is the data point generated from counting the 2.37 μCi Cs-137 source under a Cs-137 induced background. % accuracy at the indicated % dead time is lower than what would have been achieved under a corresponding Eu-152 induced dead time. The reason for this result is unclear. One possibility is that since the 2.37 μCi Cs-137 source was located just below the 12.4 μCi Cs-137 background source, it acted like a shield, thereby reducing the total Cs-137 count.

TABLE SLR #1

DETERMINED ACTIVITY OF 2.37 μCi Cs-137 SOURCE UNDER VARIOUS INDUCED BACKGROUND CONDITIONS

Count Description [1]	Net Counts Near 662KeV Peak (60 sec.)	Baseline Counts Near 662 KeV Peak (60 sec.)	Peak Width (Channels)	Determined Peak Energy (Kev)	Distance (cm) From Background Source to the Surface of the Detector	% Dead Time	Determined Activity of 2.37 μCi Cs-137 Source (μCi)	% Accuracy
2.37 μCi Cs-137 Source at 3cm from the detector	16083	173	10	66.170	---	2	2.37	100
2.37 μCi Cs-137 source at 3cm from detector with 3 mr/hr Eu-152 induced background [2]	15764	2270	11	661.39	6.8	10	2.32	98
2.37 μCi Cs-137 source at 3 cm from detector with 5 mr/hr Eu-152 induced background [2]	14588	5590	13	661.21	3.4	18	2.15	91
2.37 μCi Cs-137 source at 3cm with a 10 mr/hr Eu-152 induced background [2]	12270	10524	13	660.55	0.0	30	1.80	76
2.37 μCi Cs-137 source at 3cm with a 4 mr/hr 12.4 μCi Cs-137 induced background [3]	64921 (12530)	2006 (360)	11	661.37	3.4	12	9.55 (1.84)	78
Background count 12.4 μCi Cs-137 count at 3.4 cm from detector	52391	1646	12	661.33	3.4	12	___	___

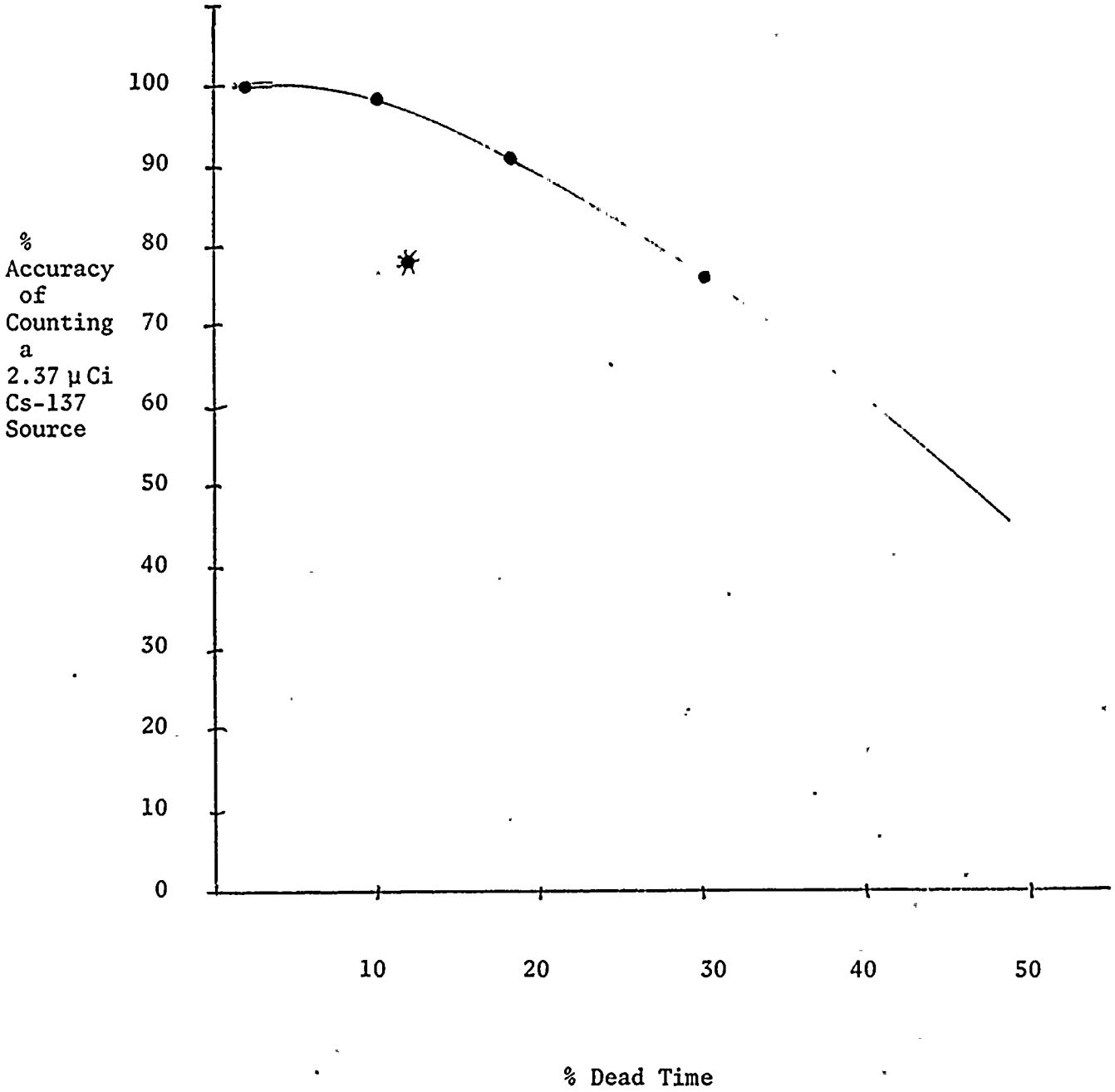
1. Stated dose rates are surface detector dose rates as measured with an Eberline E-520 G-M instrument. The midpoint of the probe was parallel to the detector surface.

2. No peaks were identified near 662 KeV with the Eu-152 source present alone. Therefore, Eu-152 background data was omitted.

3. Values in parentheses were obtained after background subtraction.

FIGURE #1

% Accuracy of Gamma Counting Versus Detector Dead Time



● data point generated under Eu-152 background conditions

✱ data point generated under Cs-137 background conditions

ATTACHMENT 4

THE ACCURACY OF GAMMA SPECTROSCOPY UNDER HIGH BACKGROUND CONDITIONS

METHODOLOGY

The accuracy of the Nine Mile Point #1 GeLi Gamma Spectroscopy System under high background conditions was investigated by (1) inducing various GeLi detector background conditions with Eu-152 and Cs-137 sources (29.4 μCi and 12.4 μCi respectively), (2) simultaneously counting a 2.37 μCi Cs-137 source at a distance of 3 cm from the detector and (3) comparing net counts found from 662 KeV Cs-137 peaks. % Accuracy was calculated by ratioing net counts found from the 662 KeV peak under each background condition with net counts from the 662 KeV peak found in the control (ie., without background source present) count.

The GeLi #1 Gamma Spectroscopy System used included an 8100 Canberra MCA (not equipped with a pile-up rejector) a 1413 Canberra amplifier and a 7229 Canberra GeLi detector. Spectrum analysis was performed using a 9825B Hewlett Packard Computer equipped with APT Peak Search Software.

RESULTS

The following results are apparent from Table SLR #1: (1) Net counts from the 662 KeV Cs-137 peak, determined activity of the 2.37 μCi Cs-137 source and % accuracy of sample counting all decreased as detector background was increased. (2) Baseline counts under the 662 KeV Cs-137 peaks, Cs-137 peak widths and detector dead time all increased as detector background was increased. (3) % accuracy of sample counting was lower in a background field induced by Cs-137 than in background fields of similar intensity induced by Eu-152.

Note in Table SLR #1 that net count data reported in parentheses for samples counted in the Cs-137 induced background field has background subtracted. This was not necessary for samples counted under Eu-152 background conditions since no peaks were found in the 662 KeV region when the 2.37 μCi Cs-137 source was removed.

CONCLUSIONS

Accuracy of the Nine Mile Point GeLi #1 Gamma Spectroscopy System decreases as detector background increases. This is probably attributable to the masking of the 662 KeV Cs-137 peak by compton scattering. Figure #1 shows a graph of % accuracy of sample count versus % dead time. Extrapolation of the data generated under Eu-152 background conditions would predict a 50% accuracy in counting at about 40-50% dead time. This corresponds to a detector surface dose rate well above 10 mr/hr or a detector midpoint dose rate above 3 mr/hr.