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THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

October 20, 1981

Mr. Ronald C. Haynes
Director
United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA. 19406

Dear Mr. Haynes:

Your letter of September 3, 1981 identified commitments made to your staff as a result of an Emergency Preparedness Appraisal (Inspection Number 81-18) performed at the Nine Mile Point Nuclear Station. The purpose of this letter is to transmit written reports of evaluations requested by Items 3 and 4 of your letter.

Attachment I provides our re-evaluation of interim post-accident sampling equipment and procedures to determine maximum concentrations that could be handled and analyzed under accident conditions.

Attachment II provides a re-evaluation of our ability to rapidly and accurately detect and measure airborne radio-iodine concentrations under field conditions in the presence of radiation levels due to noble gases.

It is our understanding that the attached reports and their transmittal satisfies the commitment made to your staff on August 27, 1981.

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

PV/mtm
Attachments

ATTACHMENT I

EVALUATIONS OF THE NINE MILE POINT NUCLEAR STATION POST-ACCIDENT SAMPLING SYSTEMS

I. INTRODUCTION

A prompt and safe means of obtaining post-accident samples is necessary to provide plant personnel a basis for estimating the magnitude of the accident and for determining protective action recommendations to State and Local Authorities.

This study was undertaken because of NRC Emergency Preparedness Appraisal (Inspection Number 81-18) performed at the Nine Mile Point Nuclear Station. As a result of this appraisal, the NMPNS was requested to determine the maximum concentrations that could be handled and analyzed under accident conditions.

The study is broken down into two parts. Part A evaluates the NMPNS capability to sample reactor water, drywell air and plant effluents per the requirements of NUREG 0737. Part B discusses the NMPNS capability to analyze the samples collected per Part A.

II. REFERENCES

1. "TMI Lessons Learned Task Force Report (Short Term)," NUREG-0578, U.S. Nuclear Regulatory Commission, July 18, 1979.

2. "Clarification of TMI Action Plan Requirements", NUREG-0737, U.S. Nuclear Regulatory Commission, October 31, 1980.

3. Denton, H: "Discussion of Lessons Learned Short Term Requirements," U.S. Nuclear Regulatory Commission, October 30, 1979.

4. "Radiation Sources," G.E. Document No. 22A2703R, Rev. 5, MPL No. A62-4100, 1978.

5. Rockwell, Theodore, (Ed.): "Reactor Shielding Design Manual," 1st Ed., United States Atomic Energy Commission.

6. Hazard Summary Report for Nine Mile Point Nuclear Station - Unit #1, Appendix E.

7. Bowers, R. (Ed.): "Nuclear Power Station Shielding Manual, Volume I Gamma Shielding," Buffalo: Niagara Mohawk Power Corporation, 1965.

8. "Final Safety and Analysis Report," Nine Mile Point Nuclear Station, U.S Atomic Energy Commission Docket 50-220 Exhibit D-2, 1967.

9. Lederer, et. al., "Table of the Isotopes," 6th Edition, John Wiley & Sons, Inc., 1967.

10. Nuclear Containment Systems, Inc., Report on In-Place Testing of Nuclear Air Cleaning Systems for Nine Mile Point Nuclear Station Unit #1, 7/22/80.

II. REFERENCES (Cont'd.)

11. Nuclear Energy Services Inc., "Shielding Design Review of Nine Mile Point Nuclear Station Unit #1, Document No. 81A0636, Rev. 1, 1980.

12. Stack Sampling, NMP-1, NMPNS Chemistry and Radiation Protection Procedure N1-SP-7, Rev. 2, October 1980.

13. Reactor Water Sampling - Suspected High Activity, NMPNS Chemistry and Radiation Protection Procedure N1-PSP-10, Rev. 0, March 1980.

14. High Activity Drywell Atmosphere Sampling and Analysis, NMPNS Chemistry and Radiation Protection Procedure N1-PSP-11, Rev. 0, March 1980.

15. Interim Procedure for High Range Stack Noble Gas Release Rate Monitoring, NMPNS Chemistry and Radiation Protection Procedure N1-PSP-12, Rev. 0, October, 1980.

III. METHODOLOGY

A. Part A - Post-Accident Sampling Evaluation

In order to appropriately determine NMPNS post-accident sampling capabilities, it was necessary to determine the total doses to be received by individuals obtaining and transporting the samples. As described in NUREG-0578 and 0737 (See References 1 & 2), individuals during post-accident sampling are limited to 3 Rem whole body and 18.75 Rem to the extremities.

The assumptions and/or references used during this evaluation are listed below segregated into two areas: Source Term Calculations and Dose/Dose Rate Calculations.

1. Source Term Assumptions

(1) For this evaluation, the General Electric Isotopic inventory for UO_2 at 30 minutes after shutdown was used (See Reference 4).

(2) The release fractions assumed in this report are based on the NUREG-0578 and 0737 (See References 1 and 2) release fractions for a Loss of Coolant Accident. This release fraction consisted of 100% Noble Gases, 50% Halogens and 1% core solids being released from the core inventory to the reactor plants liquid system.

(3) The inventory released was distributed into seven energy groups to obtain the mev/sec release rate. These seven groups consisted of 0.8, 1.3, 1.7, 2.5, 4.0, 5.0 and 6.4 mev. respectively.

(4) Since emergency core cooling system was assumed functioning, the inventory released was equally distributed throughout the Reactor and Torus Water Volume.

(5) Of the inventory released, it was assumed that 100% of the noble gases and 25% of the halogens were released to the drywell atmosphere (See references 1 and 2). This drywell air inventory was equally distributed throughout the free spaces of the drywell and torus.

(6) Source terms related to stack sampling considered three stack release scenarios:

a) LOCA with leakage postulated to the Reactor Building 30 minutes after the accident; Emergency Ventilation System running and maintaining a negative pressure in Reactor Building; Emergency Ventilation System operating at an Iodine removal efficiency of 99.99% (see Reference 10); No other exhaust fans available for dilution.

b) Same as (a), but at 24 hours after the accident has occurred.

c) Same as (a), but with a drywell purge through the Emergency Ventilation System at 30 minutes following the accident; Iodine removal efficiency same as (a).

(7) The predominate sources assumed in the Stack and Marinelli sampling flask were mainly noble gases. Iodines were a factor of 10^4 less than noble gases due to the emergency ventilation system removal efficiency.

(8) The predominate source assumed in the glass fiber and charcoal cartridge used for stack sample are Iodines. Their collection efficiency per manufacturer is 99%.

2. Dose/Dose Rate Calculations Assumptions

(1) Equations used to perform dose calculations were obtained from T. Rockwell's Shielding Design Manual (see Reference 5).

(2) All piping at sample locations except Containment Spray lines and Emergency Condenser lines were considered as line sources.

(3) Containment Spray lines and Emergency Condenser lines were considered to be cylindrical sources due to their large diameters (10 and 12 inch) and because of their proximity to the dose point location.

(4) Reactor Water and Drywell air sample vials were considered to be point sources.

(5) The stack was considered a cylindrical source 30 feet high, 21.5 feet in diameter and containing a 1.25 feet concrete wall thickness.

(6) The dose point considered for the stack was located 5 feet from the floor and 1 foot from the stack wall.

(7) The contributions to the total dose received from the stack below the floor was negligible due to the additional attenuation provided by the concrete floor. In addition, the dose contribution from the stack above the 30 foot level being considered was also negligible due to its angular orientation to the dose point location.

(8) The Marinelli sample flask was considered to be a cylindrical source.

(9) The stack glass fiber filter and charcoal cartridge were assumed to be point sources.

B. Part B - Post-Accident Sample Analysis Evaluation

Each applicable site procedure was scrutinized to determine whether indicated processes could be accomplished under the limitations imposed by the Part A sample concentrations and dose rates. In addition, a supplemental laboratory report was conducted to determine the maximum activities permitting isotopic analysis (MAPIA). The laboratory report is contained as Attachment 1 to Part B and provides the basis for the results recommendation made for Part B.

IV. RESULTS/CONCLUSIONS

A. Part A - Post-Accident Sampling Evaluation

Table 1 summarizes concentrations calculated under the three stack release assumptions, as well as the reactor water and drywell air initial activities at 30 minutes. These are the maximum concentrations expected under a LOCA condition.

Table 2 summarizes Drywell Air Sampling doses, for a sample drawn at 30 minutes. The conclusion to be drawn from Table 2 is that the drywell sample can be drawn, even at TMI-postulated conditons, if justified. Two teams can be used - one to set up the equipment and one to draw the sample. This allotment of tasks would lead to approximately 2 Rem exposure to each individual. Alternatively, the current procedures may be considered adequate for obtaining a drywell sample at 50% of TMI postulated activity.

Procedure revisions identified in this review include:

1) Provisions for communication should be made with this procedure.

2) Provisions for the utilization of 5R and 50R dosimeters should also be made to the procedure.

3) Cross reference to appropriate Emergency Plan Implementing Procedure should be included.

Table 3 summarizes doses and dose rates to be encountered at the Reactor Water Sample Sink, 30 minutes after shutdown.

Table 4 summarizes dose rates which could be encountered in the Emergency Condenser Isolation Valve Room in the process of opening manual sample line isolation valves.

The results of Tables 3 and 4 indicate the following:

1) The portion of the sampling procedure performed at the Reactor Water Sample Sink on El. 261' 0" presents no exposure problem.

2) The portion of the sampling procedure involving entry into the Emergency Condenser Isolation Valve Room presents a large exposure problem, especially since there exist also two emergency

condenser steam lines which will contain accident sources. Essentially, the dose rate from all the lines could be considered to be three times that of the containment spray line. However, if the accident releases were $5x10^{-3}$ to $1x10^{-4}$ times less (based on the same release fraction), i.e., between 20 to 40 μCi/ml, then the overall exposures in the ECIV room would be within the above-stated limits.

3) Remote operation of the ECIV room and Drywell Isolation Valves would alleviate potential exposure problems related to these samples. Remote operation of the ECIV room and Drywell Isolation Valves is currently scheduled to be completed by 3/82, dependent on receiving remaining electrical parts.

4) Provision for the use of 5R and 50R dosimeters, communications and asbestos gloves for operation of the valves in the ECIV should be made within the procedure.

5) Procedure needs to cross-reference appropriate Emergency Plan Implementing procedures.

Table 5 summarizes stack sampling dose rates encountered when stack sampling scenarios 1 and 2 are postulated.

Table 6 summarizes stack sampling capability during drywell purge.

The results of Table 5 indicate that there should be no exposure problem for stack sampling and monitoring during a LOCA if no drywell purge is required. If a drywell purge is required at 30 minutes, Table 6 indicates that no samples will be drawn during this period. During these circumstances, the interim high range stack monitor procedure will be utilized for gross estimates of release rates.

PART A

TABLE 1

ACTIVITY CONCENTRATION OF SAMPLES

ITEM	µCi/ml
Reactor Water at 30 minutes	2×10^{5}
Drywell Air at 30 minutes	4×10^{4} (1.4×10^{4} Iodines)
Stack Gases, LOCA at 30 minutes, post-accident	4.1×10^{-1}
Charcoal & Glass Filter, LOCA at 30 minutes, post-accident	2.0×10^{-5}
Stack Gases, LOCA at 24 hours, post-accident	4.1
Charcoal & Glass Filter, LOCA at 24 hours, post-accident	1.4×10^{-4}
Stack Gases, Drywell Purge at 30 minutes, post-accident	2.6×10^{4}
Charcoal & Glass Filter, Drywell Purge at 30 minutes, post-accident	1.2

PART A

TABLE 2

DRYWELL AIR SAMPLING AT H_2-O_2 MONITORING PANEL

ITEM	DOSE RATE R/hr	DURATION MINUTES	INTEGRATED DOSE Rem
Drywell Air Sample Lines (outside panel)	9.83(supply lines only)	4	0.655
	19.66(supply & return only)	8	2.621
15 ml Sample vial, unshielded	0.57	2	.019
15 ml Sample vial, shielded	0.037	10	.006
TOTAL DOSE	----	----	3.301

NOTES:

1. Dose rates are calculated at 2' from the sample lines and from the sample vial.

2. Eight minute sample time is not expected to be spent in the highest dose rate area. Actual dose rate will also include some contributions from the H_2-O_2 sample cabinet.

PART A

TABLE 3

REACTOR WATER SAMPLING AT THE SAMPLE SINK

ITEM	DOSE RATE R/hr	DURATION MINUTES	INTEGRATED DOSE Rem
Sample Lines at 5'	0.91	12	0.182
1 ml Sample Vial at 5', unshielded	0.0385	4	0.0026
1 ml Sample Vial at 2', unshielded	0.0143	10	0.0024
TOTAL	-----	--	0.19

NOTE: Reactor water activity at 30 minutes totals 2×10^{5} µCi/ml.

PART A

TABLE 4

REACTOR WATER SAMPLING AT THE EMERGENCY CONDENSER ISOLATION VALVE ROOM

ITEM	DOSE RATE R/hr	DURATION MINUTES	INTEGRATED DOSE Rem
Containment*			
Spray Line at 2'	15,874	12	N/A
Sample Line at 2'	70	12	N/A

*Dose rate also representative of Emergency Condenser steam lines.

Procedures: N1-PSP-12
N1-SP-7
Locations: Turbine Bldg.
El. 261' 0"
Screenhouse,
El. 256' 0"

PART A

TABLE 5

STACK SAMPLING (LOCA)

ITEM	DOSE RATE* mr/hr	DURATION MINUTES	INTEGRATED DOSE mRem
Stack Sample Lines at 2'	0.41	20	0.137
Marinelli Beaker 4000 ml at 1.5'	1.56	10	0.260
Charcoal Cartridge and Glass Filter at 2'	0.05	8	0.007
Stack at 1'	1.57	25	0.654
TOTAL	----	--	1.058

*Dose rates correspond to sampling at 30 minutes. For sampling at 24 hrs., multiply by 10.

Procedures: N1-PSP-12
N1-SP-7
Locations: Turbine Bldg.
El. 261' 0"
Screenhouse
El. 256' 0"

PART A

TABLE 6

STACK SAMPLING (DRYWELL PURGE)

ITEM	DOSE RATE R/hr	DURATION MINUTES	INTEGRATED DOSE Rem
Stack Sample Lines at 2'	24.5	20	N/A
Marinelli Beaker 4000 ml at 1.5'	93.4	10	N/A
Charcoal Cartridge and Glass Filter at 2'	3.4	8	N/A
Stack at 1'	97.2	25	N/A

B. Part B - Post-Accident Sample Analysis Evaluation

1. Liquid Systems (i.e., Reactor Water, Torus)

Present methodology for sampling and diluting reactor coolant samples during a LOCA can be found in site procedure N1-PSP-10. Using the one-step dilution process found in the procedure, a 1 ml reactor water sample containing a .2 Ci/ml (the activity predicted during a LOCA assuming conditions specified in NUREGs 0578 and 0737) can be diluted by a factor of 10^5 within exposure limitations of 10 CFR 20. A 1 ml aliquot of this dilution has an activity well below the maximum activity permitting isotopic analysis (MAPIA) of 1.8 mCi for small sources at 50 cm from the GeLi detector (see Attachment 1 to Part B for supplemental laboratory report).

Despite the present capability to isotopically analyze reactor water samples during a LOCA, the following procedural amendments to N1-PSP-10 are now under investigation.

1) Dilution of the reactor water at the sampling location.

2) Adjust allowable sample configuration based on the values calculated in this report.

3) Calibration of the GeLi detector at greater source distances (i.e., >50 cm shelf) thereby enabling direct analysis of higher activity samples.

These procedural amendments, together with present capabilities for sampling and analysis, should provide a sufficient degree of safety during a LOCA event until the reactor water portion of the GE post-accident sampling system is operable in March 1982.

2. Gaseous Systems (i.e., Drywell Atmosphere and Stack Gas)

The interim procedure for sampling and analysis of the drywell atmosphere during a LOCA event can be found in N1-PSP-11. As written, the procedure does not allow for the capability to isotopically analyze drywell atmospheres with activities of .04 Ci/ml (the activity predicted during a LOCA per Part A of this report.) Gaseous dilution capabilities on the order of 300x are needed to meet the maximum activities permitting isotopic analysis (MAPIA) value for 50 cm distance from the detector (see Attachment 1 to Part B for supplemental laboratory report).

The isotopic analysis of stack samples during a LOCA should be possible using site procedure N1-SP-7 with minor amendments, provided a drywell purge does not occur. Isotopic analysis of iodines and particulates at concentrations of 2.0×10^{-5} µCi/ml (the activity predicted during a LOCA per Part A of this report) can be accomplished by limiting sample volumes passed through collection media. Sample volumes up to 90,000 liters (corresponds to a 31 hr. collection period) could be counted at a distance of 50 cm from the GeLi detector without exceeding MAPIA values (see Attachment 1 to Part B for supplemental laboratory report).

2. Gaseous Systems (Cont'd)

Isotopic analysis of stack noble gases at a LOCA concentration of 0.41μCi/ml is possible using a Marinelli flask called for in procedure NI-SP-7. However, because this specific geometry has not been evaluated at 50 cm, there will be an error associated with this analysis (i.e., 2" diameter geometry vs 7.5" diameter at 50 cm.). The specific geometry is now under investigation in an attempt to evaluate and reduce the error associated with this analysis.

As evident in Part A of this report, dose rates near the stack during a LOCA drywell purge are prohibitive. For this reason, isotopic analysis of stack gases under these conditions is not possible. However, the installation of the SAI Stack Gas Analyzer, currently scheduled for completion by January 1983, will allow for dilution and analysis of noble gas concentrations up to 10^5 μCi/ml.

In an effort to further enhance isotopic analysis of gaseous systems during a LOCA, the following measures are now under consideration or planned for implementation:

1) The installation of the GE Post-Accident Sampling System to enable us to sample and dilute the drywell atmosphere.

2) Current sampling techniques are adequate for short term sampling (i.e., 30 minutes after accident). However, use of a smaller volume flask will be investigated for long term samples at later periods of an accident (i.e., 24 hours or greater).

3) The development of gaseous dilution techniques.

4) Calibration of the GeLi detector at greater source distances and with different geometries.

PART B

ATTACHMENT 1

SUPPLEMENTAL LABORATORY REPORT

ESTIMATED MAXIMUM ACTIVITIES PERMITTING ISOTOPIC ANALYSIS (MAPIA)

Methodology

The radioactivity from a 15 ml sample of off-gas was counted for 10 minutes with a GeLi detector - MCA combination at a distance of 10 cm from the crystal. Isotopic Analysis of the data was accomplished with a Hewlett-Packard computer equipped with APT peak search software and interfaced with a MCA. Percent dead time of the detector - MCA combination was observed prior to counting at sample distances of 0 cm, 3 cm and 10 cm. The Maximum Activities Permitting Isotopic Analysis at various distances from the detector were found by application of the inverse square law to the above data. The applicability of the inverse square law was verified by (1) ratioing detector efficiencies found at source distances of 3 cm and 50 cm and (2) showing that the efficiency ratios (efficiency at 3 cm/Efficiency at 50 cm) either exceeded or approximated efficiency ratios estimated by the inverse square law.

Results

Table #1 shows the isotopic analysis (decay uncorrected) of the 15 ml off-gas sample. The total activity of the sample was 6.6μCi. Also shown is the percent dead time data as read on the MCA before counting commenced. Typically, sources resulting in $\leq$ 20% dead time are considered identifiable, although some instrument gain adjustment may be necessary.

Table #3 and Graph #1 shows the results of the efficiency experiments. The ratio of the efficiencies (i.e., Efficiency at 3 cm/ efficiency at 50 cm) exceeded or approximated the efficiency ratio predicted by the inverse square law.

Conclusions

By assuming that GeLi detector efficiencies change with the inverse of the square of the distance from the detector, a conservative estimate of the maximum activity permitting isotopic analysis (MAPIA) can be made. For eg, knowing that the 6.6μCi off-gas sample resulted in 18% detector dead time at 3 cm, it can be estimated that at 50 cm distance, a $(50^2/3^2) \times 6.6\mu Ci$ = 1.8 μCi small source would result in less than 18% dead time. Using similar methodology, the following MAPIA's for small sources can be estimated:

SLR-TABLE 1

d(cm)	MAPIA (millicuries)
50	1.80
30	.66
10	.07
3	6.6E-3

SLR-TABLE 2

Isotope	Activity µCi/ml	Energy* (Kev)	% Relative Error
Xe-133	.007	81.08	9.1
Kr-88	.012	196.40	4.6
Xe-135	.022	250.26	1.4
Xe-138	.229	258.76	0.6
Kr-87	.019	403.14	2.1
Xe-135m	.152	526.84	1.1

$\lesssim$ = .441 µCi/ml x 15 ml = 6.6 µCi

Off-Gas Sample Distance From The Detector	% Dead Time
0 cm	45%
3 cm	18%
10 cm	5%

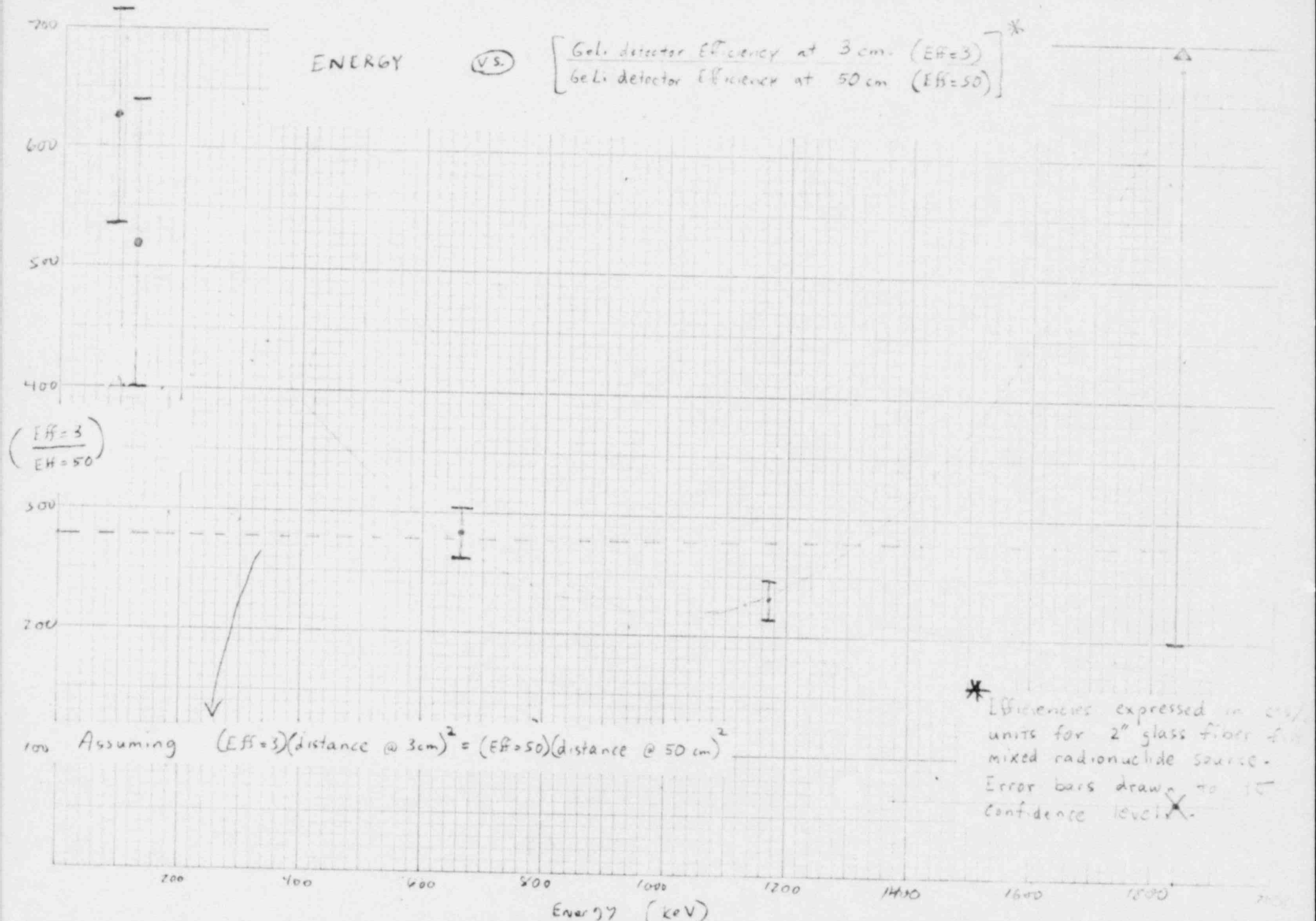
* For isotopes with multiple peaks, only the peak giving the lowest % relative error was considered.

SLR-TABLE 3

Peak Energy (kev)	Detector Efficiency* at 50 cm (A) (cts/γ)	Detector Efficiency* at 3 cm (B) (cts/γ)	(B)/(A)
88	4.78 E-5	2.00 E-2	628
122	4.60 E-5	2.4 E-2	520
662	1.81 E-5	5.1 E-3	283
1173	1.28 E-5	3.0 E-3	231
1836	2.80 E-5	1.9 E-3	679

*GeLi #2 calibration data compiled 2/10/81 and 2/17/81 using Standard Radionuclide source 121A3-09, 2" glass fiber filter.

GRAPH 1


ENERGY (V.S.) [GeLi detector Efficiency at 3 cm. (Eff=3) / GeLi detector Efficiency at 50 cm (Eff=50)] *
700
600
500
400
(Eff=3 / Eff=50)
500
200
100
Assuming (Eff=3)(distance @ 3cm)² = (Eff=50)(distance @ 50 cm)²
200
400
600
800
1000
1200
1400
1600
1800
Energy (keV)
* Efficiencies expressed in exp units for 2" glass fiber f mixed radionuclide source. Error bars drawn to 1σ confidence level.

ATTACHMENT II

Detection and Measurement of Airborne Iodine Under Field Conditions During Emergency Situations

I. INTRODUCTION

A rapid means of detecting airborne iodine activity during an emergency is necessary to expedite the identification of the plume centerline and the recommendation of protective actions to State and Local authorities. Valuable time would be consumed if samples taken out in the field had to be transported to a counting facility to determine airborne iodine activity.

During June of 1981, the NMPNS performed an evaluation of charcoal cartridges (SAI CP-100) face loaded with I-131 by Analytics, Inc. of Atlanta, Georgia. This evaluation proved to be inconclusive because it used I-131 as the only nuclide being counted.

As a result of the NRC Emergency Preparedness Appraisal (Inspection Number 81-18), NMPNS re-evaluated this earlier work using a mixed source of iodines in determining the iodine detection efficiency for environmental field samples.

The iodine detection efficiency determined as a result of this evaluation will be used in the field to provide an expeditious means of evaluating an airborne release of iodines during an emergency.

II. METHODOLOGY

To determine an iodine detection efficiency, iodines were chemically separated for NMPNS Reactor Water and then surface loaded on SAI CP-100 Radio-iodine Charcoal Cartridges. Prior to loading, the Reactor Water was analyzed on the Station GeLi to verify that only nuclides of iodine were present.

The cartridges were loaded with 1, 2, 3, and 4 ml of organic solution containing the mixed iodine activity. The cartridges were allowed to dry overnight and then counted on the station GeLi the next day to determine the deposited activity on each cartridge.

Subsequent to the GeLi analysis, each cartridge was counted using an Eberline RM-14 countrate meter and a HP-210 GM probe. Each cartridge was held approximately 1/2 inch from the GM probe and counted for a total time of 1 minute. Background for the determinations was performed using a clean CP-100 charcoal cartridge and counted in the same manner as the loaded cartridges. Each cartridge was counted three (3) times to ensure reproducibility and the data averaged for the detection efficiency determination.

II. METHODOLGY (continued)

The data retrieved from the GeLi and RM-14/HP-210 analyses was used to determine total activity and count rate respectively and inserted in the following equation to determine iodine detection efficiency:

$$\% \text{ efficiency} = \frac{\text{Total Count Rate (cpm) - Background Count Rate (cpm)}}{\text{Activity in dpm}} \times 100$$

III. RESULTS

The data collected is summarized in Table 1 and indicates that an efficiency of 5% would be more than conservative in estimating airborne iodine activity in the field immediately following a release. Based on this efficiency, Table 2 summarizes the Minimum Detectable Activities we would be able to detect in the field given a different set of variables. In all cases MDA is well above the 1×10^{-7} uc/cc detection capability specified in NUREG-0654.

IV. CONCLUSION

During the initial days of an emergency, the shorter lived iodines (I-132, 133, 134, and 135) will predominate over I-131. Given this set of circumstances, the calculated 5% iodine detection efficiency should be more than adequate in evaluating the mixed iodine activity of a released plume. Subsequently, as the I-131 becomes the predominate nuclide the detection efficiency will decrease. From previous evaluations, it could appear that this efficiency would be a factor of 10 less than the mixed iodine efficiency. In order to compensate for this difference in efficiency, current emergency implementing procedures assume that all iodine activity measured in the field is due to I-131. This over-compensation during the initial moments of an emergency would ensure that appropriate protective actions are recommended for the general public. Subsequently, as the emergency condition continues, environmental samples would be expeditiously counted using the station GeLi to verify the isotopic mix in the sample.

With respect to noble gas interference, it is not believed that noble gases will interfere with our field determination because:

1. If noble gas interference is suspected (eg. high gamma exposure rate measurements) current emergency implementing procedures require the use of Silver Zeolite cartridge for the collection of iodine samples. Silver Zeolite has a reported Xenon retention efficiency of less than 5×10^{-6} %.

IV. CONCLUSION (continued)

2. If noble gases are suspected in the counting area, current emergency implementing procedures require survey teams to retreat to a low background area (Count rate <100 cpm) for counting of the air samples.

3. All field samples are expeditiously returned to station for quantitative analysis on GeLi to verify field results.

TABLE 1

A. Background/MDC Determination

20 cpm, MDC = 4.66 $\sqrt{B}$ = 20.8 ~ 21
51 cpm, MDC = 4.66 B = 32.95 ~ 33
60 cpm, MDC = 4.66 B = 36.09 ~ 36
70 cpm, MDC = 4.66 B = 38.9 ~ 39
100 cpm, MDC = 4.66 B = 46.6 ~ 47

B. Efficiency Determination[1]

Sample[2]	Activity (dpm)	Iodine Nuclides	Total cpm	Bkgd	Net cpm	Efficiency %[3]
1ml-A	3.732×10^3	133, 135	247	60	187	5.0
1ml-B	1.3873×10^3	133	233	60	178	12.5
2ml-A	5.4745×10^3	133, 135	347	70	287	5.3
2ml-B	4.36×10^3	133	393	60	333	11.3
3ml-A	5.625×10^3	133	467	60	407	7.3
3ml-B	7.99×10^3	133,132,135	430	60	370	4.7
4ml-A	8.8×10^3	131,133,135	567	50	517	5.9
4ml-B	8.9271×10^3	133,132,135	533	60	473	5.3

1 - Total cpm was found by averaging 3 separate measurements for each sample

2 - All samples contained Xe-135 from the decay of I-135

3 - Samples Imi-B, 2ml-B, 3ml-A were dropped from overall efficiency determination since they contained only I-133, Xe-135 and not the mixed nuclides of other samples. Therefore, an efficiency of 5.24% ~ 5% will be used in field determination for mixture of nuclides.

TABLE 2

Counting Statistics for use of RM-14/HP-210 to Detect Iodines on a charcoal/Silver Zeolite Cartridge in the field.

A. Given:

1. Air Sample Volume - $15ft^3$ and $20ft^3$
2. Cartridge retention efficiency for Iodines -
   Silver Zeolite - 95%
   Charcoal (CP-100) - 99%
3. Background - 100 cpm or less
4. Detection Efficiency for Mixed Iodines - 5%
5. $$MDA = \frac{MDC}{(6.28 \times 10^{10} \frac{dpm\text{-}cc}{uCi\text{-}ft^3})(ft^3)(\text{eff. of Det})(\text{eff. of cartridge retention})}$$

B. Data:

Count Time	Air Vol (ft^3)	Bkgd (cpm)	MDC ($4.66\sqrt{B}$)	MDA uCi/cc Silver Zeolite	MDA uCi/cc CP-100
1 min.	15	60	36	8.05×10^{-10}	7.72×10^{-10}
1 min.	20	60	36	6.03×10^{-10}	5.79×10^{-10}
1 min.	15	100	47	1.05×10^{-9}	1.01×10^{-9}
1 min.	20	100	47	7.88×10^{-10}	7.56×10^{-10}