UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In The Matter Of	)	
	)	Docket No. 50-322 (OL)
LONG ISLAND LIGHTING COMPANY	)	(Emergency Planning --
	)	Phase I)
(Shoreham Nuclear Power Station,	)	
Unit 1)	)	

TESTIMONY OF JOHN F. SCHMITT AND JOSEPH S. BARON FOR THE LONG ISLAND LIGHTING COMPANY ON PHASE I EMERGENCY PLANNING CONTENTION 10(C) -- ACCIDENT ASSESSMENT AND MONITORING

PURPOSE

The purpose of this testimony is to demonstrate that the equipment the Long Island Lighting Company (LILCO) will use to monitor plant effluent during a radiological emergency satisfies the applicable regulatory requirements and guidelines. Further, this monitoring equipment, in conjunction with the other accident assessment and monitoring procedures, will enable LILCO to initiate an adequate response to the relea of iodine to the environment in the event of such an emergency.

Attachments to this Testimony:

EP 10(C)-1	Resume of John F. Schmitt
EP 10(C)-2	Resume of Joseph S. Baron
EP 10(C)-3	LILCO Emergency Plan section 6.1.1, "Assessment Instrumentation"
EP 10(C)-4	LILCO Emergency Plan section 6.4.1, "Offsite Actions"

LILCO, October 12, 1982

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ACCIDENT ASSESSMENT AND MONITORING

Q1. Please state your name and business address.

A1. [Schmitt] My name is John F. Schmitt. My business address is Long Island Lighting Company, Shoreham Nuclear Power Station, Post Office Box 628, Wading River, New York 11792.

[Baron] My name is Joseph S. Baron. My business address is Stone and Webster Engineering Corporation, 245 Summer Street, Boston, Massachusetts 02107.

Q2. By whom and in what capacity are you employed?

A2. [Schmitt] I am employed by LILCO as the Radiochemistry Engineer fo the Shoreham Nuclear Power Station. My current resp nsibilities include maintaining the Radiation Monitoring System (RMS) operable, quantifying radioactive releases from the plant, and calculating offsite doses.

[Baron] I am employed by Stone & Webster Engineering Corporation as a Power Engineer on the Shoreham Nuclear Power Station. My current responsibilities include the procurement of a calibrated RMS at Shoreham. The iodine collection system for the station vent and the Reactor Building Standby Ventilation System (RBSVS) exhaust is a subsystem of the RMS.

Q3. Please state your professional qualifications and explain why you are knowledgeable about the issues raised in EP 10(C).

A3. [Schmitt] My resur e, describing my professional qualifications, appears as Attachment EP 10(C)-1. My knowledge of the issues raised in EP 10(C) stems from my current position. As Radiochemistry Engineer, I am responsible for assessing radioactive releases, including

iodine, from the plant during normal and accident conditions. This function requires thorough familiarity with the system used to monitor releases.

[Baron] My resume, describing my professional qualifications, is Attachment EP 10(C)-2. My knowledge of the issues raised in EP 10(C) stems from the fact that the post accident iodine sampling and collection equipment was purchased as part of the post accident RMS for which I am responsible.

Q4. Are you familiar with the text of EP 10(C)?

A4. [Schmitt & Baron] Yes. EP 10(C) states:

> The equipment intended for use by LILCO to monitor plant effluent does not provide timely and accurate information as to the actual value of the quantity of iodine released to the environment in the case of a radiological accident. In the absence of such timely and accurate information, LILCO is unable to initiate an adequate response to the release of iodine to the environment in the case of such an accident.

Q5. Does EP 10(C) cite any statutory or regulatory deficiencies with respect to the equipment LILCO intends to use to monitor plant effluents?

A5. [Schmitt & Baron] EP 10(C) does not cite any specific statutory or regulatory deficiencies. The preamble to EP 10, however, states that LILCO fails to comply with 10 C.F.R. §§ 50.47(b)(2), (4), (8), (9), and (10); 10 C.F.R. Part 50, Appendix E; and NUREG-0654, Items II(B), (D), (H), (I), and (J).

Q6. What do those provisions require?

A6. [Schmitt & Baron] The portions of 10 C.F.R. § 50.47 cited in the preamble to EP 10 require:

> (b) The onsite and offsite emergency response plans for nuclear power reactors must meet the following standards (footnote deleted):
>
> * * *
>
> (2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.
>
> * * *
>
> (4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

* * *

> (8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.
>
> (9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.
>
> (10) A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

10 C.F.R. Part 50, Appendix E is entitled "Emergency Planning and Preparedness for Production and Utilization Facilities." As the title indicates, the Appendix covers emergency planning and preparedness for the Shoreham Nuclear Power Station. EP 10(C), however, does not cite any specific parts of Appendix E with which LILCO does not comply.

EP 10 also cites NUREG-0654, Items II(B), (D), (H), (I), and (J). Each of these Items contains a Planning Standard which is identical to one of the above-cited subsections of 10 C.F.R. § 50.47. Each Planning

Standard is followed by a number of evaluation criteria to be used in measuring compliance with the Planning Standard. EP 10(C), however, does not cite specific evaluation criteria with which LILCO does not comply.

Q7. Does LILCO meet the regulatory requirements and guidelines that the County cites in EP 10(C)?

A7. [Schmitt and Baron] Yes. The regulatory requirements and guidelines call for adequate methods, systems, and equipment for monitoring potential iodine releases. LILCO's iodine monitoring system, in conjunction with other procedures and equipment, provides this capability.

Q8. Are there any other regulatory guidelines, not cited by the County, that cover post accident monitoring equipment?

A8. [Schmitt & Baron] Yes. Regulatory Guide 1.97, Revision 2 and NUREG-0737 are applicable. LILCO's iodine monitoring system satisfies the guidelines in both those documents.

Q9. Simply put, what is the gist of 10(C)?

A9. [Schmitt & Baron] It appears that Suffolk County's position is that LILCO is unable to assess, in a timely and accurate fashion, the releases of iodine from Shoreham during an emergency. This alleged inability, according to the County, prevents LILCO from initiating an adequate emergency response.

Q10. Please explain how the iodine monitoring system at Shoreham works.

A10. [Schmitt & Baron] The two potential release paths for iodine during an accident are the station vent exhaust and the Reactor Building Standby Ventilation System (RBSVS) exhaust. The method for determining the quantity of iodine released is identical for each pathway.

Shoreham's design includes radiation monitors in these two pathways. These monitors work by continuously withdrawing a sample of the exhaust stream via a special sampling probe and passing it through both a charcoal filter, that traps almost all of the iodine in the stream, and a noble gas detector. The monitors' filtration logic is such that after a filter assembly receives a predetermined level of activity, a valve is automatically closed and the next filter assembly is valved into

service. The amount of iodine emitted via that effluent stream is then determined by removing the charcoal filter and analyzing it in the onsite laboratory facilities.

Q11. What is the RBSVS exhaust?

A11. [Schmitt & Baron] During normal operation, the reactor building discharges through the Reactor Building Normal Ventilation System (RBNVS) exhaust. The RBNVS, along with the exhaust from the turbine and radwaste buildings, normally discharge through the station vent.

In most accident situations, however, the RBNVS will be secured and the RBSVS, with its own radiation monitors, will be in operation. The RBSVS exhaust is the low volume effluent gas stream filtered through High Efficiency Particulate Absolute and charcoal filters that is required to aid in maintaining the reactor building at negative pressure during certain abnormal operating conditions. Thus, during most accidents the reactor building discharges only through the RBSVS exhaust, while the turbine and radwaste buildings continue to discharge through the station vent (see LILCO Emergency Plan section 6.1.1, Attachment EP 10(C)-3).

Q12. Does the iodine monitoring equipment LILCO will use affect LILCO's initial recommendation for an adequate response to the potential release of iodine in case of a radiological accident?

A12. [Schmitt & Baron] No.

Q13. How is the radiological monitoring equipment used in initiating an adequate response to a potential release of iodine?

A13. [Schmitt & Baron] Should a radiological accident occur, the effluent monitors, with their sampling probes in the release paths, immediately measure the noble gases being released. Based on this measurement of noble gases, and making the conservative assumption that the iodines released are proportionate to the amount of noble gases released, a conservative figure for the quantity of iodine released is computed. It is this computed iodine figure that is one of the inputs to the dose assessment model used to recommend an adequate response (see LILCO's testimony, including attachments, on EP 4 and 14). The County's intimation that an actual measurement from the iodine monitoring system is the sole figure used to initiate an adequate response, and that such a

figure is necessary for that purpose, is simply incorrect (see LILCO Emergency Plan, section 6.4.1, Attachment EP 10(C)-4). Recognizing, however, that the initial computed iodine figure overestimates the amount of iodine actually being released, the iodine monitoring system as well as the field monitoring teams are used to refine that initial value (see LILCO's testimony, including attachments, on EP 10(B)).

In sum, the iodine monitoring system is not necessary to initiate an adequate response to a potential iodine release. The computed iodine figure, available almost immediately because of the instantaneous noble gas measurements, provides a conservative iodine release figure that is refined by the figures obtained from the iodine monitoring system and the field monitoring teams.

Q14. Is the iodine monitoring system, in conjunction with the field monitoring teams, effective in refining the initial iodine figure?

A14. [Schmitt & Baron] Yes.

Q15. Are the systems and procedures described above for iodine monitoring similar to those used at other operating nuclear plants?

A15. [Schmitt & Baron] Yes.

Q16. Mr. Schmitt and Dr. Baron, will you please summarize your conclusions regarding this contention?

A16. [Schmitt & Baron] The post accident iodine monitoring equipment at Shoreham meets the guidelines specified in Regulatory Guide 1.97, Revision 2; NUREG-0737; as well as the regulatory requirements and guidelines cited in EP 10. This equipment, in conjunction with accident dose assessment models and field monitoring teams, permits Shoreham to initiate adequate responses in the event of a radiological emergency and to monitor and assess iodine releases into the environment.

PROFESSIONAL QUALIFICATIONS

JOHN F. SCHMITT

Radiochemistry Engineer

LONG ISLAND LIGHTING COMPANY

My name is John F. Schmitt. I am the Radiochemistry Engineer of the Shoreham Nuclear Power Station, a position I have held since January 1975. As such, I am responsible for developing and implementing the chemistry, radiochemistry and effluent monitoring program for Shoreham. This includes, among other things, directing all work related to conducting the chemical and radiochemical analyses and treatments of plant process systems; detecting and controlling environmental releases; implementing the ALARA policy for these releases; and preparing records and reports of chemical surveys.

I graduated from Manhattan College in 1966 with a Bachelor of Science degree in chemistry and received a Master of Science degree in Environmental Health Science, specializing in Radiological Health (Health Physics), from the University of Michigan in 1974 and became a Certified Health Physicist in 1982. I completed the General Electric Boiling Water Reactor Chemistry Course in November 1975. I have also completed many industry seminars and training programs, including:

a. Radiation Protection - LILCO Evening Institute

b. Radiation Protection Workshops - General Electric Company

c. BWR Chemistry Training - General Electric Company

d. Health Physics Review - Rockwell International

e. Accelerated Health Physics Instruction - NUS

f. Accelerated Nuclear Plant Chemistry Instruction - NUS

g. Health Physics Review - Brookhaven National Labs

h. Environmental Radiation Surveillance - Harvard School of Public Health

i. Radioactive Waste Management for Nuclear Power Reactors - ASME/University of Virginia

j. Post Accident Sampling Workshops - Sentry Equipment, EPRI

k. Control of Plant Radiation Fields - EPRI, General Electric Company

l. Atomic Absorption/Atomic Emission Spectrometry - Instrumentation Labs

m. Gamma Spectrometer Operation - Canberra Industries

I started work for the Long Island Lighting Company in 1966 as an Assistant Engineer at the Far Rockaway Power Station. I took a military leave of absence from 1967-1972 to serve as an officer in the U.S. Air Force. Returning to LILCO in 1972, I was an Associate Engineer at the Glenwood Power Station. From 1973 until assuming my present position in 1975, I was assigned to the staff of the Shoreham Nuclear Power Station as an Associate Engineer and Plant Engineer. During this time, I studied health physics at the University of

Michigan and received training at the AEC's Savannah River Plant and Commonwealth Edison's Dresden Nuclear Power Station.

I am a member of the Health Physics Society, New York Chapter of the Health Physics Society, Power Reactor Health Physicists, and the Long Island Chapter of the American Nuclear Society.

PROFESSIONAL QUALIFICATIONS

JOSEPH S. BARON

Power Engineer, Nuclear Engineering Group

STONE & WEBSTER ENGINEERING CORPORATION

My name is Joseph Baron. My business address is 245 Summer Street, Boston, Massachusetts 02107. I am employed by Stone & Webster Engineering Corporation (SWEC) as a Power Engineer and have held this position since January 1973. In this capacity I am currently responsible for the radiation monitoring system for the Shoreham Nuclear Power Station - Unit 1 Project.

I was awarded a Bachelor of Science degree in chemical engineering in 1966, dual Master of Science degrees in chemical and nuclear engineering in 1968, and a Ph.D. in nuclear engineering in 1971, all by Massachusetts Institute of Technology.

Prior to joining SWEC in August 1971, I worked as a part-time Assistant Process Engineer for Diamond Shamrock Company in Cleveland, Ohio. I was responsible for the evaluation of chemical kinetics data, development of a workable kinetics model for use in the design of a production chemical reactor and design of scrubbing towers. Later as a Research

Associate with Argonne National Laboratory, I established the setup and calibration of an analytical system for the determination of inpurities in sodium. Next, with Oak Ridge National Laboratory as a Research Associate (August 1967 - February 1968), I was responsible for the design of an accurate method of determining the thermal flux history of the irradiaton cavity of the high flux isotope reactor, for feasibility and kinetic studies in the use of amines as dehydrating agents in the microsphere production step of the Sol-Gel process; analysis of the electrical charge distribution in a metallic aerosol; and preparation of reactor physics data for use in an economic evaluation of a high temperature gas-cooled reactor (HTGR). From February 1968 - August 1971 I was involved in resident study toward my doctorate degree.

Upon joining SWEC in August 1971 as an Engineer in the Nuclear Division, I functioned as an Assistant Supervisor in charge of the design and development of light water reactor (LWR) radioactive waste systems as well as specialist in ion exchange. In this capacity, I interacted with technical staff members involved in other plant systems in an effort to minimize potential radioactive releases. I supervised the simulation group which developed computer models for the operation of radioactive waste systems and for plant effluent releases, both steady state and transient. On assignment to the Boston Edison Pilgrim Project, I participated in the conceptual development

of alternate radioactive waste processing capability. I was also involved in the evalation of the existing equipment, and systems to determine the long-term viability. Another activity concerned determination and development of various accident scenarios for the liquid metal fast breeder reactor (LMFBR) prototype project.

On the Wisconsin Utilities Project as Principal Nuclear Engineer (February 1978 - July 1979) I was responsible for all nuclear steam supply system (NSSS) interfaces and the design of systems in the reactor portion of the plant. I participated in the development of site specific potential accident sequences. On temporary assignment to Virginia Electric and Power Company's Surry project, I assisted in coordinating the proposed primary coolant hot magnetic filter retrofit, which was not installed.

As Principal Nuclear Engineer on the SWEC sponsored Reference Nuclear Power Plant (July 1979 - May 1980), I ensured that systems designs within the reactor portion of the plant met applicable interface criteria for the various pressurized water reactor (PWR) NSSS vendors and developed generic systems descriptions. I participated in the design and development of the concept of the Independent Fuel Storage Facility.

Later, as Lead Nuclear Process Engineer on the Nuclear Power Company, Ltd. (NPC), Project (April 1980 - May 1981), I was responsible for the development of the Civil Demonstration

Fast Reactor Cover Gas System design. Additionally, I coordinated design and structural activities for the NPC efforts within the London and Boston offices.

I was also responsible for developing an economical and efficient method of cleaning the reactor coolant of a boiling water reactor following an inadvertent injection of sodium pentoborate. A constraint was using existing plant equipment. This involved simulation of the various operations to determine the rate limiting step; the development and sequencing of the process to minimize the impact of this step was an integral part of the study for Toyo Engineering, Japan.

Additionally, I was engaged in development of the conceptual process design for a coal slurry dewatering and storage facility. Although a generic design was being developed, specific application was for the Nevada Power and Light Company.

Since assigned as a Power Engineer on the Shoreham Nuclear Power Station - Unit 1 (SNPS-1) Project (May, 1981), I am responsible for securing a workable and calibrated radiation monitoring system. This will be achieved through the support of experience in the design and construction of test apparatus, planning experiments and analyzing accumulated data.

I am a Registered Professional Engineer in Massachusetts and a member of the following technical societies: The American Institute of Chemical Engineers, the American Nuclear Society, The American Nuclear Society's

Standards Groups developing design criteria for Gaseous and Liquid Radioactive Waste Systems for Light Water Reactors, Sigma Xi - Honorary Research Society, Tau Beta Pi - Honorary Engineering Society and Phi Lambda Upsilon - Honorary Chemical Society.

Publications include "Upper-Bound Cost/Benefit Analysis under Appendix I for a Hypothetical Pressurized Water Reactor," J.S. Baron and R.M. Vanasse, presented at the ANS Toronto meeting in June 1976; and "Treatment of Liquid Wastes," Chapter 6, Nuclear Power Waste Technology, J.S. Baron and B. V. Coplan, ASME (1978).

6.0 EMERGENCY MEASURES

6.1 Assessment Actions

Accurate assessment of the situation is important in the initial stages to assure proper implementation of the Emergency Plan. Implementation of station procedure, Classification of Emergency Action Levels, is the guide to be used for both the initial classification of an incident and for continuing re-assessment for those emergencies of extended duration to assure proper classification of conditions.

For each of the four emergency classifications, extensive and continuing assessment actions will be done for the purpose of:

1. Identification and characterization of the incident.

2. Prediction of offsite doses resulting from the incident.

3. Notification of and verification of offsite authorities.

4. Determination of appropriate measures.

5. Indication of escalation, reduction, or termination of the emergency classification.

Figure 6-4 presents a detailed breakdown of the sequence of plant events and actions taken by plant personnel up to and including notification of State and County personnel of the emergency. Times presented are elapsed times from time zero and are based on a detailed analytical analysis of the loss of coolant incident, where possible, and on engineering judgment.

Time estimates are referenced to the Low Low Level Signal (i.e., approximately 180" above top of core) resulting from a loss of coolant. Thus, no credit is taken for plant personnel to initiate protective measures during the period following the incident and preceding the Low Low Level signal. Depending on the actual incident, this time period may be significant. Although a very low probability event, a loss of coolant accident is the only event which has the potential of requiring evacuation of persons living in the vicinity of the plant.

6.1.1 Assessment Instrumentation

The station instrumentation which is provided for accident assessment includes the following equipment:

1. Station Ventilation Exhaust Monitor

2. Turbine Building Normal Ventilation Exhaust Monitor

3. Reactor Building Normal Ventilation Exhaust Monitor

4. Radwaste Building Normal Ventilation Exhaust Monitor

5. Reactor Building Standby Ventilation Exhaust System Monitors

6. Post Accident Monitors

7. Radiation Monitoring System

8. Meteorological Tower Input (i.e., wind speed, direction, and temperature measurements)

9. Effluent Flow Rate Meters

10. Post Accident Reactor Building Standby Ventilation

11. Post Accident Station Ventilation Exhaust

Table 6-1 lists the monitors which would detect and monitor all accident releases for the accidents analyzed in Chapter 15 of the FSAR. The ranges of the instruments, which are described below, are also provided in the Table.

Additional instrumentation, alarms and annunciators used for accident assessment and classification are defined directly in the Emergency Action Levels in Section 4 of this volume. Further details of these systems may be found as follows:

• Process and Radiological Monitors	FSAR 11.4
• Radiation Protection Design Features	FSAR12.3
• Seismic Instrumentation	FSAR 3.7
• Hydrologic Instrumentation	System Des. 1020.135
• Meteorological Monitoring	System Des. 1020.659
• Portable Monitors	EPIP SP69.062.01
• Survey Equipment	EPIP SP69.020.01

The Station Ventilation Exhaust Duct Monitor and the Turbine, Reactor, and Radwaste Buildings Normal Ventilation Exhaust Monitors are described in Sections 11.4 and 12.3.4 of the FSAR. Their function is to detect normal releases which result from minor steam and liquid leakages within the plant. Each detector is provided with a limited multichannel analysis capability which will allow identification of selected gamma emitting isotopes. These units are capable of detecting gross concentrations of about 10^{-6} to 10^{-1} uCi/cc of gas and about 10^{-10} to 10^{-5} uCi/cc of particulates without exceeding the range of the instrument. These units will allow timely detection and identification of releases resulting from minor occurences and from most of the accidents analyzed in Chapter 15 of the FSAR.

The Reactor Building Standby Ventilation System Exhaust Monitors are described in Section 11.4 of the FSAR. These redundant units are seismically qualified and are built to Quality Assurance Category I standards. Their functions is to detect activity release from a loss of coolant accident, a fuel handling accident, an instrument line break, or any other releases which cause actuation of the Reactor Building Standby Ventilation System. Each unit is capable of

detecting gross concentrations of about 10^{-6} to 10^{+2} uCi/cc of gas without exceeding the range of the instrument. Each unit also has a multichannel analysis capability similar to that for the Station Ventilation Exhaust Duct Monitor. These units will allow assessment of activity release from the Reactor Building Standby Ventilation System and are provided with strip chart recorder and rate meters in the Control Room.

The Post Accident Monitors are high range radiation detectors which are provided on the Station Ventilation Exhaust Duct and on the Turbine Building, Reactor Building, and Radwaste Building Normal Ventilation Exhaust Ducts. The Post Accident Monitors are described in Section 11.4 of the FSAR. This function is to monitor releases from the Station Ventilation Exhaust Duct and from the ventilation exhaust from each of the three major buildings of the plant. Each unit is capable of detecting gross radioactivity concentrations of about 10^{-2} to 10^{+3} uCi/cc without exceeding the range of the instrument. Each unit is provided with multichannel analysis capability similar to that for the Station Ventilation Exhaust Monitor. These units will allow timely detection and assessment of any serious accident which occurs in any building of the plant. The Post Accident Monitors are provided with strip chart recorders and rate meters in the Control Room.

Signals from all of the above monitors are sent to the Radiation Monitoring System computer. This computer will evaluate the inputs from each monitor, along with air flow rates in each duct or pipe, and will calculate the current activity release rate and the integrated values for specific periods of time. The computer will calculate individual isotopic release rates based on a sample of the effluent, if available, or from the data obtained from the multichannel analysis conducted at the monitor level, or from the hypothetical nuclide composition described in the FSAR for the respective accident. The system will combine the release data with atmospheric dispersion information (X/Q values). This information is available through a direct tie from the meteorological instruments (wind speed, direction and temperature difference), or the conservative accident X/Q values from Chapter 15 of the FSAR will be used. These inputs will be used along with the dose evaluation methods of NRC Regulatory Guide 1.3 to calculate and display the following information:

1. X/Q versus direction and distance (if meteorological inputs are available).

2. Dose rate versus direction and distance.

3. Estimates of dose rates and total integrated doses for any distance, direction, and time, past or future.

4. In-plant dose rates due to airborne activity (if in-plant airborne monitors are operational).

Area radiation monitors as described in FSAR Section 12.3.4 will provide information on radiation levels for selected in-plant locations.

In the event that the instrumentation normally providing input to the Radiation Monitoring System computer is offscale or inoperable the necessary meteorological and radiological information can be obtained directly. Emergency plan implementing procedure Determination of Offsite Doses details the method for this calculation and assumptions for accident scenario and isotopic composition of release. If the radiation monitor reading cannot be used, provisions for input from the results of a grab sample (I-131 or Xe-133 dose equivalent) can be utilized to determine the release rate or the offsite dose.

As an aid in assessing potential core damage Figure 6-6 shows drywell high range monitor response as a function of time after reactor scram for the following LOCA scenarios:

- 100% primary coolant
- 100% gap activity release
- 100% fuel damage
- 10% fuel damage
- 1% fuel damage

For each of these scenarios the isotopic mix was varied for noble gases, halogens and solids. Other assumptions utilized were drywell leak rates, design basis activity values and contribution from deposited nuclides. Based on this graph an indication of the inventory mix and extent of core damage is provided for use in accident assessment. Source term determination can also be made by taking a containment air or reactor coolant sample from the post accident sampling facility.

The Post Accident Reactor building Standby Vent and the Post Accident Station Vent Exhaust are both Category I off-line gas monitors with ranges from 10^{-2} uCi/cc extended up to 10^{+4} uCi/cc to detect gross concentrations of gas. They shall also be tied into the computer system and strip chart recorders.

1. Post-Accident Sampling Capability

   The design basis of the post accident sampling system (PASS) and post accident sampling facility (PASF) is to give site personnel the capability of drawing and analyzing prompt samples (less than 3 hours) under accident conditions while insuring a radiation exposure of less than 5 rem to the whole body and 75 rem to the extremities. The PASS is also capable of providing at least one sample per day for seven days following the onset of an accident and at least one sample per week until the accident condition no longer exists.

System Description

The PASS, with the exception of containment and system isolation valves, is operated from a control panel located in the PASF (see Figs. II.B.3-1A and B of SNPS-1 FSAR). This system will provide the following capability:

(a) Sampling of reactor coolant, suppression pool liquids and containment atmosphere.

(b) Prompt quantification (less than 3 hours) of certain radionuclides that are indicators of the degree of core damage.

(c) Chemical analyses within 3 hours of commencement of the post accident sampling operation.

(d) On-line gross gamma activity level monitoring.

(e) Dilution of liquid, reactor coolant gases or containment atmosphere samples by either volumetric or feed and bleed methods for off-line gamma spectrum analysis.

(f) On-line chemical analysis of chloride, boron, and total gas concentrations, as well as pH and conductivity performed on full strength reactor coolant and suppression pool water.

(g) The ability to perform all analysis assuming a highly radioactive initial sample (Reg. Guide 1.3 source term).

Continuous monitoring of containment hydrogen and oxygen levels is provided as part of the primary containment atmospheric control system. Both diluted and undiluted grab samples are provided. The diluted grab samples are used for radioactive spectium analysis of liquids and gases. Provisions are being developed for backup analysis for the on-line equipment. Certain chemical analyses can only be accurately performed on undiluted coolant samples off-site. Provisions have been made in the design of the sampling facility to obtain such undiluted coolant samples at the required frequency. An agreement and plan to perform such analyses at an offsite facility is being actively pursued, including design of a mutually acceptable sample container, to provide for backup chemical analyses. This backup capability will be implemented at the earliest practical date, but not necessarily prior to fuel load.

Sample Building (PASF)

The PSAF is located on the south side of the reactor secondary containment at zero azimuth directly adjacent to the truck lock. The PASF is a two-story structure designed to house the PASS. The layout of both floors are shown on Figs. II.B.3-2A and B of SNPS-1 FSAR. This building will be accessible following an

accident (reactor building entry not necessary) to obtain and analyze required post accident samples.

The facility is divided into four distinct sections:

(a) Sample enclosure

(b) Sample collection room

(c) Support (HVAC) room

(d) Habitable (Control) area

The sample enclosure is an integral part of the reactor building secondary containment and utilizes the reactor building standby ventilation system (RBSVS) to filter and monitor any potential leakage release from the PASS. The remainder of the facility is served by a 1000 scfm charcoal/HEPA filter that is placed in service post accident to provide a habitable environment for personnel and analytical equipment.

2. Additional Accident Monitoring Instrumentation

Noble Gas Effluent Monitor

The effluent monitor categories in Table II.F.1-1 of SNPS-1 FSAR which apply to the Shoreham Nuclear Power Station are:

(a) BWR reactor building exhaust air.

(b) Other release points.

(c) Buildings with systems containing primary coolant or gases.

The following monitors have been provided to satisfy the requirements of Table II.F.1-1:

(a) Reactor Building Standby Ventilation (RBSVS) RE-021 and 022.

(b) Post Accident High Range RBSV - RE-134.

(c) Station Vent Exhaust - RE-042

(d) Post Accident High Range Station Vent Exhaust - RE-126.

Details on placement, calibration, ranges, power supply, instrument capability, overlap and calculation methods can be found in SNPS-1 FSAR pages II.F.1-1 and -4 and Figure II.F.1-1.

In-Plant Iodine Monitors

The station vent exhaust monitor (RE-042) obtains radioiodine and particulate sample media for analysis in the onsite

radiochemistry laboratory when the secondary containment is accessible. When the secondary containment building is inaccessible, monitors RE-055 (Radwaste Building ventilation flow) and RE-057 (Turbine Building ventilation flow) are accessible and are each sampled for radioiodine and particulates. Details on the above can be found in SNPS-1 FSAR item II.F.1.

Containment Monitoring

(a) High Range Radiation Monitor

Two physically separate monitors are located inside the drywell for photon radiation. For details on the above, refer to SNPS-1 FSAR pages II.F.1-7 and 8 and Table II.F.1-3 and 4 and Figure II.F.1-1.

(b) Pressure Monitor

Installed instrumentation provides continuous display and recording of containment pressure in the control room to measure three times the design pressure of the primary containment. For details, refer to SNPS-1 FSAR page II.F.1-9 and Table II.F.1-3.

(c) Water Level Monitor

Containment wide-range water level indication channels meet the design and qualification criteria as outlined in Appendix B to NUREG-0737 to the maximum extent practical. For details refer to SNPS-1 FSAR page II.F.1-10.

(d) Hydrogen Monitor

The hydrogen concentration in the primary containment atmosphere is continuously monitored. The system consists of two subsystems each including two hydrogen analyzers to sample the drywell and the suppression chamber atmosphere. For details refer to SNPS-1 FSAR page II.F.1-11.

3. In-Plant Radiation Monitoring

The in-plant iodine concentration will be determined by using either portable (at least 4), semi-portable (at least 3) or fixed (located in ventilation streams) air samplers to draw a known quantity of air through either a charcoal filter or silver zeolite cartridge. For details refer to SNPS-1 FSAR pages III.D.3.3-1 and 2.

6.1.2 Radiological Environmental Monitoring

To supplement the plant instrumentation data, radiological survey teams will be dispatched as necessary to perform site and offsite surveys. These teams will be equipped with radiation survey

Additionally, EPIP describes subsequent and/or supplemental corrective actions for the scope of potential situations within each of the emergency classifications. These EPIPs are designed to guide the actions of personnel to correct or mitigate a condition as early and as near to the source of the problem as feasible. Specific actions are described, for example, which may prevent or significantly reduce a potential release of radioactive material, provide for prompt fire control and ensure timely damage control and repair. These procedures are also utilized as emergency training media and are the basis for periodic emergency drills.

6.4 Protective Actions

6.4.1 Offsite Actions

The EPIP gives the details of which offsite authorities will be notified for each emergency class, information to be provided in accordance with the New York State Notification Fact Sheet and Dose Assessment Fact Sheet and verification practices to be used.

LILCO will make a protective action recommendation to Suffolk County and New York State authorities for the population at risk. The various protective action options available are detailed in the New York State and Suffolk County emergency response plans. The protective action recommendation is based upon dose projection calculations, field monitoring data, EPA protective action guidelines, sheltering factors offered by local dwellings and evacuation time estimates for ambient conditions. The emergency plan procedure, "General Emergency" immediate implementing actions, contains protective actions to be recommended during events that are deteriorating rapidly based upon conditions in accordance with NUREG 0654, Appendix 1. The details of this decision process are contained in the EPIPs. Regarding the protective actions taken on behalf of the general public, notification will be made of an emergency situation via the use of the Prompt Notification System set up throughout the ten (10) mile Emergency Planning Zone (EPZ).

This notification system, installed by LILCO, will be operationally tested and functional prior to fuel load and consistent with the criteria set forth in Appendix 3 to NUREG-0654.

Although the utilization of this system is the responsibility of Suffolk County (individual operating and administrative responsibilities for this system are described fully in the County's Emergency Response Plan Procedures), the system shall be maintained by LILCO. This system, made up of sirens for general population coverage and tone activated radios for special facilities (i.e., hospitals, nursing homes, nursery schools, etc.), shall alert the public within the 10 mile EPZ of a possible nuclear incident.

Upon notification of an emergency to the general public via the Prompt Notification System, the public shall be directed by previously disseminated information to tune to a specific radio station and await

informative instruction on what protective actions such as sheltering or evacuation, if any, should be taken for their respective Emergency Response Planning Area.

Informative pamphlets shall be located in strategic locations such as gas stations, motels and resorts for the purpose of supplying the transient population with emergency information. Public notification and education are reviewed in great detail in Section 8.4.

Evacuation routes are defined in the Suffolk County Emergency Plan; however, maps of the EPZ and population distribution, in a sector format, are located on Figures 6-2 and 6-3, respectively.

As stated above, notification to the public as a whole will be made via the siren warning system. Incorporated into this system for the purpose of notifying those organizations with a large number of personnel, such as large businesses, hospitals, etc., are separately operated, tone-activated, alert radios which would be in accordance with the appropriate County procedures. At the same time, the population would be notified of the need for evacuation, buses would be dispatched to evacuate schools and special institutions, and road blocks would be set up for the purpose of restricting in-coming traffic in accordance with the Suffolk County Radiological Emergency Response Plan.

The basis for the choice of recommended protective actions from the plume exposure pathway is shown in the EPIP. Time estimates for the evacuation of the 10 mile EPZ are as delineated as in the attachment to LILCO's submittal to the NRC in SNRC-488, dated August 7, 1980 and as amended by the information found in Appendix C.

6.4.2 Plant Site Action

Protective action within the plant site will be initiated by actual or imminent radiological conditions or other habitability hazards such as toxic gas or fire. Upon assessment by the Emergency Director that a situation exists that requires evacuation of areas of the plant, an evacuation signal will be activated simultaneously with an announcement of the emergency condition over the party page system indicating the areas to be evacuated. Evacuated personnel will report to designated assembly areas consistent with implementing procedures.

When personnel have assembled, personnel accountability will then proceed following the guidance of the personnel accountability procedures. Accountability for onsite personnel will be accomplished within 60 minutes.

In the event of a site evacuation, Figure 6-1 details the onsite assembly areas with primary and secondary evacuation routes leading to the LILCO main access road. Transportation for onsite personnel shall be by personal vehical as well as car pooling where conditions warrant.