



LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

P.O. BOX 604, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

July 22, 1981

SNRC-602

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SHOREHAM NUCLEAR POWER STATION - Unit 1
Docket No. 50-322

Dear Mr. Denton:

Enclosed herewith are sixty (60) copies of LILCO responses to specific NRC concerns which were previously identified as requiring additional information to complete NRC review. Attachment A provides a list of the specific responses included.

If you require additional information or clarification, please do not hesitate to contact this office.

Very truly yours,

B. R. McCaffrey
B. R. McCaffrey
Manager, Project Engineering
Shoreham Nuclear Power Station

Enclosures

cc: J. Higgins

B001
3/1

SNRC-602, July 22, 1981

Attachment A

Additional information is provided for the following items:

- 1) NUREG-0737 Item II.B.2 - Plant Shielding
- 2) NUREG-0737 Item II.B.4 - Training for Mitigating Core Damage
- 3) NUREG-0737 Item II.F.1 - Accident Monitoring Instrumentation
(Attachments 1, 2, 3, and 5)
- 4) NUREG-0737 Item III.D.3.3 - Radiation Monitoring
- 5) SER Open Item No. 37 - Secondary Containment Bypass Leakage
- 6) SER Section 6.3.1 - HPCI/RCIC 50% Plugged Strainer Test

II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation

NRC Position

With the assumption of postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. These actions should be taken pending conclusions resulting in the long term degraded core rulemaking, which may result in a need to consider additional sources.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station and sample analysis area must be included among those areas where access is considered vital after an accident. (See Item III.A.1.2 for discussion of the TSC and emergency operations facility). The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation release control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security

center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

In order to assure that personnel can perform necessary postaccident operations in the vital areas, the following guidance is to be used by licensees to evaluate the adequacy of radiation protection to the operators:

(1) Source Term

The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, and 1.7 and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., you may assume the radioactive decay that occurs before fission products can be transported to various systems).

- (a) Liquid-Containing Systems: 100 percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal (RHR), high-pressure coolant injection (HPCI), and low-pressure coolant injection (LPCI), or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, you may assume that the water contains no noble gases.
- (b) Gas-Containing Systems: 100 percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor-containing lines connected to the primary system (e.g., BWR steam lines), the concentration of radioactivity shall be determined assuming the activity is contained in the vapor space in the primary coolant system.

(2) Systems Containing the Source

Systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation should include, but not be limited to, containment, residual heat removal system, safety injection systems, chemical and volume control system (CVCS), containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not

be considered for this analysis. Leakage measurement and reduction is treated under Item III.D.1.1 "Integrity of Systems Outside Containment Likely To Contain Radioactive Material for PWRs and BWRs". Liquid waste system need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Item III.D.1.4, "Radwaste System Design Features to Aid in Accident Recovery and Decontamination".

(3) Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rate than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case bases. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are not to be used to limit access in the event of an accident.

- (a) Areas Requiring Continuous Occupancy: <15 mrem/hr (averaged over 30 days). These areas will require full-time occupancy during the course of the accident. The control room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.
- (b) Areas Requiring Infrequent Access: GDC 19. These areas may require access on an irregular basis, not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant radiochemical/chemical analysis laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples of sites where occupancy may be needed often, but not continuously.

(4) Radiation Qualification of Safety-Related Equipment

The review of safety-related equipment which may be unduly degraded by radiation during postaccident operation of this equipment relates to equipment inside and outside of the primary containment. Radiation source terms calculated to determine

environmental qualification of safety-related equipment consider the following:

- (a) LOCA events which completely depressurize the primary system should consider releases of the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to the containment atmosphere.
- (b) LOCA events in which the primary system may not depressurize should consider the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulate) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10 percent noble gases, 10 percent iodines, and 0 percent particulate as a source term.

The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/Particulate)
Outside	% (100/50/1) in RCS	% (10/10/0) in RCS
Inside	<u>Larger of</u> (100/50/1) in containment	(10/10/0) in RCS
	<u>or</u> (100/50/1) in RCS	

LILCO Position

Areas where access is vital after an accident have been analyzed for post accident personnel access. The Shoreham position is that access is only needed to the control room, the Technical Support Center (TSC), and the Post Accident Sampling and Analysis Facility (PASF). The other areas suggested as vital post accident in NUREG-0737 either do not apply for Shoreham or are not needed. The hydrogen recombiner system is controlled remotely from the control room. The containment isolation reset control area is also located in the main control room. Shoreham has no manual ECCS alignment area; all vital ECCS valves are automatic and operable remotely from the control room. Motor control centers do not need to be accessed, nor do the instrument panels located outside of the control room. No operability access is needed for emergency power supplies since they are

remotely operable from the control room. The radwaste control panels are not needed for accident mitigation since they control no safety related functions.

Access to the Security Center is not needed to gain access to the control room, TSC or PASF. Since the control room is always manned, entrance could be gained manually at any time, should automatic security systems fail. Access to the PASF can be gained from any of the Emergency Operations facilities without passage through a building containing high radiation areas. Keys are available to authorized personnel to manually open the doors to the TSC or PASF should the automatic security systems fail.

The control room and the permanent TSC are both habitable (less than 15 mrem/hr) (30 day avg) in a post-accident environment. The PASF is designed for limited habitability to the extent necessary to obtain and analyze samples per the specific requirements of Item II.B.3. The PASF habitability is primarily affected by the LOCA cloud around the facility which is drawn in through the filtered intake. The worst-case gamma dose rate for this case will occur at about $t=8$ hours and will be less than 100 mrem/hr within the manned area of the facility.

The worst case accident which impacts the normal radiochemistry laboratory (in the turbine building) habitability is the Control Rod Drop or the failure of the air ejector lines. Both of these increase the airborne hazard in the laboratory to higher levels than a LOCA. They will result in gamma dose rates of approximately 200 mrem/hr at the worst part of the accident.

Detailed radiation calculations were performed to ensure adequate environmental qualification of safety related equipment within the harsh, post accident environment of the reactor building.

SOURCE TERM

Radioactive source release and distribution assumptions for Shoreham are as follows:

Radioactive Source Release

1. The percentages of core inventory radioactive fission products assumed to be released from the fuel rods are:

Noble gases (Kr, Xe)	100%
Iodine	50%
Others	1%

2. This entire release is assumed to occur instantaneously at the start of the accident.

Radioactive Source Distribution

In order to envelope the full spectrum of break sizes and depressurization rates, two bounding events and source distributions were considered.

1. LOCA (both pressurized and depressurized events)

The following fission products are considered to be uniformly mixed in the following volumes:

a. Suppression Pool

Noble gases	0%
Iodine	50%
Others	1%

b. Combined Drywell/Wetwell Air Space

Noble gases	100%
Iodine	25%
Others	1%

c. Reactor Coolant System Steam Space

The following distribution is used for determining reactor building pipe shine doses due to HPCI, RCIC and MSIV-LCS operation.

Noble gases	100%
Iodine	25%
Others	0%

Using the above distribution, time history radiation zones were established within the primary containment and within the secondary containment as follows:

- i. Primary Containment - Previously described total integrated dose accident radiation levels (Table 3.11.2-1) adequately bound all LOCA events for equipment within the primary containment and no new analyses were required.
- ii. Secondary Containment - Time history radiation zones were established for the secondary containment using the above sources distributed in the steam and liquid piping in the following fluid systems which were conservatively assumed to operate concurrently:
 1. HPCI
 2. RCIC
 3. RHR (all essential modes)
 4. Core Spray
 5. RBSVS
 6. MSIV-LCS
 7. PCAC

In addition to radiation shine from the above system piping and components, the primary containment was assumed to leak at technical specification limits

resulting in an airborne source term which was included in the radiation zoning. As provided by NUREG-0737, no additional leakage was assumed.

iii. Excluded Systems

- a. All piping which could potentially carry undiluted reactor coolant into the secondary containment is isolated and is nonessential (e.g., RWCU, shutdown cooling mode of RHR). Accordingly, the undiluted reactor coolant liquid source discussed above was excluded. Adequate means are provided using the post accident sampling system to ensure safe coolant activity levels exist prior to use of any of these nonessential systems.
- b. The post accident sampling lines were excluded from specific evaluation due to their size (typically 3/8 in tubing) and because they are flushed after each use. Enough conservatism in integrated dose calculations exists to bound any effects of these small sources.
- c. The gaseous radwaste system lines were also excluded because they were not sources for any safety related equipment. They are also located outside the reactor building and are isolated.

Pipe break inside containment and nonpipe break events are bounded by the LOCA event above.

2. Pipe Break in Secondary Containment

As specified in NUREG-0737, secondary containment airborne time history dose and dose rates were established for this event using the following fission products uniformly mixed in the primary coolant system:

Noble gases	10%
Iodine	10%
Others	0%

Reactor building radiation zones and time history dose and dose rate data from the above analyses are shown on Figs. II.B.2-1 through 17.

The above radiation conditions are being used in conjunction with other environmental conditions (pressure, temperature, and humidity) for the equipment qualification program. Safety related equipment is being qualified in accordance with NUREG-0588.

FIGURES FOR ITEM II.B.2

REMAIN UNCHANGED.

REFER TO SNRC-563

DATED 5/15/81 FOR FIGURES.

II.B.4 - Training For Mitigating Core Damage - Additional Information

NRC Position: Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

LILCO Response: The program for Training For Mitigating Core Damage, as outlined in our previous response to this issue (See SNRC-579 dated May 29, 1981), will be in compliance with the GUIDELINES FOR TRAINING TO RECOGNIZE AND MITIGATE THE CONSEQUENCES OF CORE DAMAGE from The Institute of Nuclear Power Operations, Document Number STG-01, Rev. 1, dated January 15, 1981. The detailed course description is currently under preparation and will be submitted for NRC Staff review by October 1, 1981.

The Shoreham Shift Technical Advisors and operating personnel, from the Plant Manager through the operations chain down to the licensed operator level will receive all of the training indicated in Enclosure 3 to Mr. H. R. Canton's March 28, 1980 clarification letter.

Supervisory personnel and technicians in the Instrumentation and Control (I&C), health physics, and chemistry sections will receive the necessary training commensurate with their responsibilities.

II.F.1 Additional Accident Monitoring Instrumentation

Introduction

Item II.F.1 contains the following subparts:

1. Noble gas effluent radiological monitor;
2. Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities (this requirement was inadvertently omitted from NUREG-0660; see Attachment 2 that follows, for position);
3. Containment high-range radiation monitor;
4. Containment pressure monitor;
5. Containment water level monitor; and
6. Containment hydrogen concentration monitor.

Attachments 1 through 6 present the NRC position on these matters.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

1. the use of this information by an operator during both normal and abnormal plant conditions,
2. integration into emergency procedures,
3. integration into operator training, and
4. other alarms during emergency and need for prioritization of alarms.

ATTACHMENT 1 NOBLE GAS EFFLUENT MONITORNRC Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

1. Noble gas effluent monitors with an upper range capacity of $10^5 \mu\text{Ci/cc}$ (Xe-133) are considered to be practical and should be installed in all operating plants.
2. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of $10^5 \mu\text{Ci/cc}$ (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

Licensees shall provide continuous monitoring of high-level, postaccident releases of radioactive noble gases from the plant. Gaseous effluent monitors shall meet the requirements specified in the attach Table II.F.1-1. Typical plant effluent pathways to be monitored are also given in the table.

The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions. The design description shall include the following information.

1. System description, including:
 - a. instrumentation to be used, including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;
 - b. monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

- c. location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;
 - d. assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and
 - e. the source of power to be used.
2. Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

LILCO Position

The effluent monitor categories in Table II.F.1-1, which apply to the Shoreham Nuclear Power Station are: (a) "BWR reactor building exhaust air", (b) "other release points", and (c) "buildings with systems containing primary coolant or gases". See Fig. II.F.1-1 for a simplified diagram of Shoreham's gaseous effluent layout.

The maximum anticipated primary containment leakage rate is 0.005 volumes per day (volume of primary containment is 1.93×10^5 cu ft) into the secondary containment which has a volume of 2×10^6 cu ft. The primary containment leakage is highly diluted in the secondary containment atmosphere. This mixture will be discharged after passing through high efficiency particulate absolute filters and charcoal absorber banks via the reactor building standby ventilation system (RBSVS) discharge pipe, at the top of the station vent exhaust. Two Class 1E radiation monitors (RE-021 and 022) serve this system downstream of the filters and adsorbers along with a post accident Class 1E monitor (RE-134), which is added to the system for higher ranges.

The RBSVS monitors are supplied with power from vital instrument buses. These monitors read out in the control room and are located in the control building to permit access during an accident for collection of their radioiodine and particulate sample media for laboratory analysis.

The criteria in Table II.F.1-1 for other release points and buildings with systems containing primary coolant or gases are applicable to the station vent exhaust monitor (RE-042) and the station vent post accident high range monitor (RE-126). Normal ventilation discharges from the reactor building, the turbine building, and the radwaste building are mixed, thereby providing dilution prior to being exhausted through the station vent exhaust. When RBSVS is operating, and the reactor building normal ventilation system (RBNVS) is isolated, the loss of normal reactor building ventilation flow is compensated by opening louvers at the station vent exhaust to permit 90,000 cu ft/min of

outside air for dilution and to maintain a constant air velocity through the station vent. This single discharge point for the combined ventilation flow from all potentially contaminated buildings is monitored by a noble gas radiation monitor (RE-042) and post accident high range effluent radiation monitor (RE-126). The monitor (RE-042) is supplemented by in-line RE-069 with a high upper range. In addition, the individual building ventilation flows to the station vent exhaust are each analyzed by a high range in-line radiation monitor (RE-066, 067, and 068). All these monitors, except RE-042 and 126, are powered from a vital instrument bus.

Where practical, initial calibration includes detector response for a minimum of three decades using standard sources of three different energies and intensities. These calibration curves are initially generated using both gaseous and solid sources, where practical. Routine calibration of these monitors is in accordance with technical specifications provisions using solid sources related to the initial calibration. Calibration sources used are Sr-90, Cs-137, and Co-60 for low range monitors, and Cs-137 for high range monitors.

The conversion of the instrument readings to release rates are determined using the energy response of the detectors obtained during calibration. Accident release rates are then calculated based on anticipated radionuclide inventories following a design basis loss of coolant accident. Actual releases may be determined by analyzing a grab sample and correcting the release rate calculated. Continuous strip chart recording and CRT display are provided in the control room. Digital readout for the high range effluent monitors RE-126 and 134 will assure the availability of continuous reading in the control room during or after an accident.

The effect of background radiation on readings of the RBSVS noble gas monitors (RE-021, 022, and 134) will be minimized during an accident, due to their location in the control building and the detector's location in a 4" lead shield. For the station vent exhaust monitor (RE-042 and 126), background radiation in the vicinity of the monitor within the secondary containment will have minimal effect on the noble gas detector, due to its location in a 4" lead shield and the fact that the detector is a thin beta scintillator. This type of detector is very inefficient for detecting gamma radiation which might penetrate the lead shield, while it is efficient for detecting the beta radiation associated with the sample stream's noble gases brought in close contact with the detector.

For a listing of the radiation monitors with the ranges provided, refer to Table II.F.1-4.

ATTACHMENT 2 SAMPLING AND ANALYSIS OF PLANT EFFLUENTSNRC Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

Licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates to meet the requirements of the enclosed Table II.F.1-2. Licensees shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techniques are used, then provisions shall be made to minimize noble gas interference.

The shielding design basis is given in Table II.F.1-2. The sampling system design shall be such that plant personnel could remove samples, replace sampling media and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC 19 of 5-rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ± 20 percent. Further departure from the isokinetic condition need not be considered in design. Corrections for non-isokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969 may be considered on an ad hoc basis.

Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

LILCO Position

The normal station vent exhaust monitor (RE-042) is not powered from a vital instrument bus, however, it is powered from a dependable backup power supply to normal ac. Due to its location in the secondary containment, it may be inaccessible during an accident. This would preclude obtaining the radioiodine and particulate sample media from the monitor for analysis. However, inability to obtain these samples is compensated for by the fact that the turbine building and radwaste building ventilation flows are each sampled for radioiodine and particulates by the equipment associated with the normal range noble gas monitors for these flows (RE-057 and 055). These monitors are both located in the turbine building permitting access for collection of the sample media during an accident in order that laboratory analysis may be performed. Adding the results obtained for radioiodine or particulates from the turbine building and radwaste building ventilation flows will give the radioiodine or particulate release at the station vent exhaust should the secondary containment be inaccessible. Under these circumstances, RBSVS is operating and there is no reactor building ventilation contribution to the station vent exhaust. As discussed above, the RBSVS release is monitored separately for noble gases and continuous collection of samples for particulates and radioiodine releases (RE-021, 022, and 134). These monitors are capable of representative monitoring and sampling for all accident conditions except for pipe break outside containment (refer to Appendix 3C). The monitors associated with the reactor, radwaste and turbine buildings ventilation systems are not powered from a vital bus. This is consistent with the design of the monitored systems. The station vent exhaust monitors (RE-042 and 126) radioiodine and particulate sample media can be obtained for analysis if the secondary containment is accessible.

The addition of the high range station ventilation exhaust monitor (RE-126) assures continuous sampling of radioiodine and particulates during accident conditions. Continuous sampling is achieved with isokinetic sampling during normal operation and sampling probes during accident conditions. Provisions have been made to comply with ANSI N13.1-1969 to the maximum extent practical to assure representative sampling. The sampling collector will initiate an alarm in the control room when it reaches a concentration of $10^2 \mu\text{Ci/cc}$ and 30 min collection time. At this time the microcomputer associated with RE-126 transfers the flow to the next particulate and iodine assembly, isolates the alarmed assembly, and indicates to the operator the need to replace the collector assembly and transfer it to the laboratory for analysis.

The sampling media is paper with more than 90 percent collection efficiency for 0.3 micron particles and a charcoal cartridge with more than 90 percent collection efficiency for methyl iodide.

The radioiodine and particulate sampling media is analyzed in the counting room at Shoreham. Charcoal cartridges are purged with nitrogen or air to remove entrapped noble gases. A separate counting station is provided which serves as a backup for the counting facility in the radiochemistry laboratory. At least one of these locations will remain a low-contamination, low-background area for all postulated accident conditions. The above meets the requirements of Table II.F.1-2.

Further, procedures will be prepared for conducting all aspects of the measurement and analyses correctly and in a manner to minimize personnel exposure.

ATTACHMENT 3 CONTAINMENT HIGH-RANGE RADIATION MONITORNRC Position

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-3.

The specification of 10^8 rad/hr in the above position was based on a circulation of postaccident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments but gamma - sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10^7 R/hr.

The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

For BWR Mark III containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

The monitors are required to respond to gamma photons with energies as low as 60 KeV and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV, as specified in Table II.F.1-3. Monitors that use thick shielding to increase the upper range will under-estimate postaccident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

LILCO Position

Two physically separate monitors are located inside the drywell, 78'-7", for photon radiation. One is located adjacent to the equipment hatch and the other being adjacent to the personnel hatch, (180° separation). See Fig. II.F.1-2. These locations

have been selected to provide an unobstructed, large view of the containment volume, and to ensure ease of access for replacement, maintenance, and calibration. Calibration will be performed during routine refueling outages.

These monitors are each powered by a vital instrument bus, are seismic qualified, and are designed to withstand the temperatures, pressures, humidity, and total radiation in the drywell containment through an accident. Monitor readouts are displayed continuously and recorded on a Category I panel in the main control room. These monitors provide unshielded, unattenuated containment radiation readings during an accident and meet the requirements of Table II.F.1-3. For a listing of the radiation monitors with the ranges provided, refer to Table II.F.1-4.

ATTACHMENT 4 CONTAINMENT PRESSURE MONITORNRC Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

Design and qualification criteria are outlined in Appendix B to NUREG-0737.

Measurement and indication capability shall extend to 5 psia for subatmospheric containments.

Two or more instruments may be used to meet requirements. However, instruments that need to be switched from one scale to another scale to meet the range requirements are not acceptable.

Continuous display and recording of the containment pressure over the specified range in the control room is required.

The accuracy and response time specifications of the pressure monitor shall be provided and justified to be adequate for their intended function.

LILCO Position

Currently installed instrumentation provides continuous display and recording of containment pressure in the control room. Pressure transmitters and associated instrumentation have been replaced in order to provide the capability to measure three times the design pressure of the primary containment. The range of pressure instrumentation is from -5 to +150 psig. The pressure transmitters have an accuracy of ± 0.25 percent of span and at 100 F the response times are 0.2 sec (63 percent of the time). The components provided meet the design criteria outlined in Appendix B to NUREG-0737 to the maximum extent practical.

ATTACHMENT 5 CONTAINMENT WATER LEVEL MONITOR

NRC Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

The containment wide-range water level indication channels shall meet the design and qualification criteria as outlined in Appendix B to NUREG-0737. The narrow-range channel shall meet the requirements of Regulatory Guide 1.89.

The measurement capability of 600,000 gallons is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

Narrow-range water level monitors are required for all sizes of sumps but are not required in those plants that do not contain sumps inside the containment.

For BWR pressure-suppression containments, the emergency core cooling system (ECCS) suction line inlets may be used as a starting reference point for the narrow-range and wide-range water level monitors, instead of the bottom of the suppression pool.

The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

LILCO Position

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 Shoreham the lower limit is at the elevation of the center line of the ECCS suction lines containment penetrations. In order to provide suppression pool water level measurement with an upper limit of 5 ft above the normal water level (26'-6") the currently installed instrument taps have been relocated to 31'-6". The accuracy of the water level monitors are ± 0.2 percent of span.

II.F.1, Attachment 5 - (Cont'd) - Additional Information

The ECCS suction lines (including RCIC) currently penetrate the containment at El 24-0. The lines then turn downward via an elbow connection and are attached to a suction strainer. The elevation of the strainer connections are approximately as follows:

- RHR - EL 21-3
- CS - EL 21-1
- HPCI- EL 20-10
- RCIC- EL 22-3.5

Separate level monitoring instrument penetrations are provided in the suppression pool at El 24-0, and represent the approximate nominal elevation for adequate system performance. Operation below the suction line penetration would be operating under suction lift conditions; ie., if pump operation were stopped, a subsequent restart could not be guaranteed since a vacuum condition could exist with possible air in leakage. Also, operation near or at the minimum pipe suction elevation would be subject to vortex concerns and pump performance could again not be guaranteed.

As discussed above, the level monitoring instrument penetrations are appropriately located at El 24-0. In addition, it should be noted that there are no remaining spare penetrations below that elevation. It is not feasible to provide additional penetrations at this time. The Shoreham containment construction is complete and drilling through the liner and concrete would have a significant schedule impact. Also, it is not desirable to provide connections below the minimum water level from the standpoint of containment integrity.

Shoreham has also considered alternative instrumentation for level measurement of the suppression pool. We have in the past evaluated instruments which could be mounted inside the primary containment such as those presently utilized in Pressurized Water Reactors. However, due to the structural loads impressed by Mark II Containment suppression pool swell, it was judged impractical since the level instrumentation was of the float type and the manufacturers had no test data to evaluate the effects of such an event. Non-mechanical ultrasonic type level instrumentation was evaluated and found unsuitable for use inside the primary containment due to a history of poor reliability and a lack of any manufacturer who could provide such an instrument for Category I service. We believe that the use of proven, reliable instrumentation mounted outside the primary containment seismically and environmentally qualified, and accessible for periodic calibration and test is the most acceptable arrangement.

In summary, continuous, repetitive and reliable system operation would not be expected below the existing penetration elevation and should not be a system design requirement.

ATTACHMENT 6 CONTAINMENT HYDROGEN MONITOR

NRC Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Design and qualification criteria are outlined in Appendix B of NUREG-0737.

The continuous indication of hydrogen concentration is not required during normal operation. If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

The accuracy and placement of the hydrogen monitors shall be provided and justified to be adequate for their intended function.

LILCO Position

The hydrogen concentration in the primary containment atmosphere is continuously monitored by the hydrogen analysis system. This system consists of two redundant subsystems, each including two hydrogen analyzers to sample the drywell and the suppression chamber atmosphere (see Figure 6.2.5-1). Each analyzer is provided with dedicated instrument penetrations to ensure continuous monitoring. The range of the analyzer is from 0 to 10 percent hydrogen concentration by volume over a pressure range of -2 to +60 psig. The accuracy of the hydrogen monitors are 2 percent of full scale. Containment hydrogen concentration measurement channels meet the design and qualification criteria as outlined in Appendix B to NUREG-0737 to the maximum extent practical.

TABLE II.F.1-1HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

- REQUIREMENT - Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.
- PURPOSE - To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

DESIGN BASIS MAXIMUM RANGE

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter of air at standard temperature and pressure (STP) for monitors employing beta radiation detector (Note: 1R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gases changes) should be taken into account.

- $10^5 \mu\text{Ci/cc}$ - Undiluted containment exhaust gases (e.g., PWR reactor building purge, PWR drywell purge through the standby gas treatment system).
- Undiluted PWR condenser air removal system exhaust.
- $10^4 \mu\text{Ci/cc}$ - Diluted containment exhaust gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).
- BWR reactor building (secondary containment) exhaust air.
- PWR secondary containment exhaust air.
- $10^3 \mu\text{Ci/cc}$ - Buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).
- PWR steam safety valve discharge, atmospheric steam dump valve discharge.
- $10^2 \mu\text{Ci/cc}$ - Other release points (e.g., radwaste buildings, fuel handling/storage buildings).

SNPS-1 FSAR

TABLE II.F.1-1 (CONT'D)

REDUNDANCY	-	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	-	(None) Sampling design criteria per ANSI N13.1.
POWER SUPPLY	-	Vital instrument bus or dependable backup power supply to normal ac.
CALIBRATION	-	Calibrate monitors using gamma detectors to Xe-133 equivalent (1 R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.
DISPLAY	-	Continuous and recording as equivalent Xe-133 concentrations or $\mu\text{Ci/cc}$ of actual noble gases.
QUALIFICATION	-	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	-	<p>Offline monitoring is acceptable for all ranges of noble gas concentrations.</p> <p>Inline (induct) sensors are acceptable for 10^2 $\mu\text{Ci/cc}$ to 10^5 $\mu\text{Ci/cc}$ noble gases. For less than 10^2 $\mu\text{Ci/cc}$, offline monitoring is recommended.</p> <p>Upstream filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident.</p> <p>For external mounted monitors (e.g., PWR main steam line), the thickness of the pipe should be taken into account in accounting for low-energy gamma radiation.</p>

TABLE II.F.1-2SAMPLING AND ANALYSIS OR MEASUREMENT OF HIGH-RANGE RADIOIODINE AND PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

- EQUIPMENT - Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for PWR secondary main steam safety valve and dump valve discharge lines.
- PURPOSE - To determine quantitative release of radioiodines and particulates for dose calculation and assessment.
- DESIGN BASIS - 10^2 $\mu\text{Ci/cc}$ of gaseous radioiodine and particulates deposited on sampling media; 30 minutes sampling time, average gamma energy (E) of 0.5 MeV.
- SHIELDING
- ENVELOPE

SAMPLING MEDIA

- Iodine > 90% effective adsorption for all forms of gaseous iodine.
- Particulates > 90% effective retention for 0.3 micron (μ) diameter particles.

SAMPLING CONSIDERATIONS

- Representative sampling per ANSI N13.1-1969.
- Entrained moisture in effluent stream should not degrade adsorber.
- Continuous collection required whenever exhaust flow occurs.
- Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

ANALYSIS

- Design of analytical facilities and preparation of analytical procedures shall consider the design basis sample.
- Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

SNPS-1 FSAR

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT	-	The capability to detect and measure the radiation level within the reactor containment during and following an accident
RANGE	-	10^4 rad/hr to 10^6 rads/hr (beta and gamma) or alternatively 1 R/hr to 10^3 R/hr (gamma only).
RESPONSE	-	60 keV to 3 MeV photons, with linear energy response $\pm 20\%$ for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.
REDUNDANT	-	A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).
DESIGN AND QUALIFICATION	-	Category I instruments as described in Appendix B to NUREG-0737, except as listed below.
SPECIAL CALIBRATION	-	In situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.
SPECIAL ENVIRONMENTAL QUALIFICATIONS	-	Calibrate and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to 10^6 R/hr. Prior to initial use, certify calibration of each detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr.

SNPS-1 FSAR

TABLE II.F.1-4

RADIOACTIVITY CONCENTRATION RANGES FOR SHOREHAM
GASEOUS EFFLUENT RADIATION MONITORS.

<u>GASEOUS EFFLUENT MONITOR</u>	<u>RANGE</u> <u>(μCl/cc)</u>
Reactor Building Standby Ventilation RE-021, RE-022*	1×10^{-6} to $1 \times 10^{+2}$
Post Accident Reactor Building Standby Ventilation RE-134*	3×10^{-3} to $2 \times 10^{+5}$
Reactor Building Normal Ventilation RE-029*	1×10^{-6} to 1×10^{-1}
Turbine Building Ventilation RE-057*	1×10^{-6} to 1×10^{-1}
Radwaste Building Ventilation RE-055*	1×10^{-6} to 1×10^{-2}
Station Vent Exhaust RE-042*	1×10^{-6} to 1×10^{-1}
Post Accident Stativ. Vent Exhaust RE-126*	3×10^{-3} to $2 \times 10^{+5}$
Reactor Building Normal* Ventilation RE-068	1×10^{-2} to $1 \times 10^{+3}$
Turbine Building Ventilation RE-067*	1×10^{-2} to $1 \times 10^{+3}$
Radwaste Building Ventilation RE-066*	1×10^{-2} to $1 \times 10^{+3}$
Station Vent Exhaust RE-069*	1×10^{-2} to $1 \times 10^{+3}$
Drywell Monitors RE-085A, B	1×10^0 to 1×10^7 R/hr

*Ranges shown for these radiation monitors are for the noble gas portion of the monitor.

SHOREHAM NUCLEAR POWER STATION - UNIT 1
FIRE HAZARD ANALYSIS REPORT

III.D.3.3 Inplant Radiation Monitoring

NPC Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

- a. The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.
- b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.
- c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.
- d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

LILCO Position

The inplant iodine concentration will be determined by using either portable, semi-portable, or fixed air samplers to draw a known quantity of air through either a charcoal filter or silver zeolite cartridge. Fixed samplers are located in ventilation streams and have charcoal filter collection capability for radioiodine.

There shall be at least three (3) semi-portable continuous air monitors capable of detecting particulate, iodine, and noble gas concentrations, (Eberline Instrument Corporation PING-3 or equivalent). Also included shall be at least four (4) portable air samplers with the capability of using either a charcoal filter or silver zeolite cartridge for collection of radioiodine (Eberline Instrument Corporation RAS-1/ICH-1 or equivalent). If the presence of interfering noble gas activity is confirmed, silver zeolite shall be used where feasible for iodine sampling.

Prior to analysis, the charcoal filters will be purged with bottled nitrogen or clean air to remove entrapped noble gases; this is not necessary for the silver zeolite sample. The sample will be counted according to normal operating health physics procedures using instrumentation capable of accurately measuring iodine concentrations. Instrumentations used for this analysis will be located in both the radiochemistry counting room and the alternate on-site counting room. At least one of these locations will remain a low-contamination, low-background area for all postulated accident conditions.

Item # 37 - Secondary Containment Bypass Leakage

- I. An evaluation was conducted of all fluid systems to examine the lines or penetrations that pass through the primary containment and extend, without being vented to the secondary containment, outside the secondary containment boundary. A list of those lines and the results of our review are as follows:

A. Feedwater System

Primary containment isolation is provided by a check valve inside and a testable check valve outside of primary containment. In addition to the containment isolation valves, there is a motor operated stop check valve located outside the primary containment. The two check valves located outside of primary containment are equipped with positive closure measures which ensure valve closure and seating following an accident. After isolation of the HPCI and RCIC systems, there are three valves in series between the inside of primary containment and the environment.

B. HPCI and RCIC Suctions

The HPCI and RCIC systems draw suction initially from the condensate storage tank and then from the suppression pool. The condensate storage tank, during HPCI and RCIC operation, is isolated from suppression pool water by means of a check valve and motor-operated gate valve in each system's suction line. After operation of the HPCI and RCIC systems is terminated, a second, closed, motor-operated gate valve provides additional isolation.

C. Core Spray Suctions

A locked-closed globe valve in each suction line of the core spray system isolates suppression pool water from the condensate storage tank.

D. HPCI and RCIC Test Return Lines

Two motor operated, normally closed gate valves on the test return lines isolate the condensate storage tank from either the HPCI or RCIC systems. After HPCI or RCIC operation is terminated, further isolation is provided by closure of another motor operated gate valve.

E. Condensate Fill Connections to the HPCI, RCIC, Core Spray and RHR Systems

The condensate transfer system is used as the alternate fill source to the RHR, HPCI, Core Spray and RCIC systems. The condensate connections are isolated by means of normally closed globe valves as well as check valves in the lines.

F. RWCU Connection to the Condenser

During an accident the RWCU System is isolated from the RPV and RWCU is not expected to contain highly radioactive water. The condenser is further isolated from the RPV by two closed gate valves in the blowdown line.

G. Suction and Recirculation Lines for the RCIC Loop Level Pump from the Condensate Storage Tank

Three (3) 1" spring-loaded check valves are used to isolate the RCIC Loop level fill system from the RCIC system. Further isolation is provided when operation of RCIC is terminated.

H. Service Water System & Ultimate Cooling Connection to the RHR System

The ultimate cooling water connection to the RHR system is protected against leakage in either direction by dual isolation valves with a drain-off connection between the two valves.

I. CRD System

The design configuration of this system is discussed in FSAR Section 6.2.4.3.2 (Containment Isolation System)

J. RBCLCW System

This system including the provisions for leakage detection is discussed in FSAR Section 9.2.2.

K. Instrument & Service Air System

This system is discussed in FSAR Section 9.3.1 and TMI Item Response II.K.3.28.

The service air portion of the system is not hard piped to the containment penetration. As shown on Figure 9.3.1-1D, a base connection with a quick disconnect type fitting is removed during plant operation. This arrangement effectively vents any leakage past the isolation valves to the secondary containment.

- II. None of the lines or penetrations identified in Section I need be considered as potential bypass leak paths of containment atmosphere.

Lines A through H, by system design, functions, and location, will not expose the internals of the primary containment isolation valves to containment atmosphere. Water exists on both sides of the containment isolation valves thereby preventing contact with containment atmosphere and acting to seal the valves against atmosphere leakage. The isolation valves will not be exposed to containment atmosphere for at least 30 days.

Within the primary containment Systems I, J and K are closed and they are not exposed to containment atmosphere. Systems I and J are used during normal operation, and potential system degradation would be identified during normal maintenance. Systems I and J are water filled which further act as a seal against containment atmospheric leakage. In addition, System J is vented to the secondary containment via the RBCLCW head tank.

The Instrument Air Subsystem of System K is at all times at a system pressure greater than the containment peak pressure ensuring that any air leakage would be into containment.

III. N/A

IV. N/A

- V. Closed Systems I & K meet all of the requirements of Position 9, BTP CSB 6-3.

SER Section 6.3.1 - Additional Information

HPCI/RCIC - 50% PLUGGED STRAINER TEST

LILCO Response

LILCO will perform tests to verify the flow capability of the HPCI pump and the RCIC pump, with their respective strainers 50% plugged, during the Preoperation Startup Test Program.

These tests will be performed using auxiliary boiler steam. The RCIC test will demonstrate full flow capability while the HPCI test will verify a flow less than full flow due to auxiliary boiler steam capacity limitations. Based on this partial flow data, calculations and extrapolations using the HPCI pump curves will be performed to demonstrate full HPCI pump capability. The test procedures, data, analysis, and results will be on file as part of the Pre-op Program.