LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

July 23, 1981 SNRC-605

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

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U.S. NUCLEAR REGULATORY COMMISSION

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

RWG/mh

Enclosures

cc: J. Higgins

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S 1/1

SNRC-605, July 23, 1981

ATTACHMENT A

Additional information is provided for the following items:

1) SER Open Item No. 11 - ECCS Calculation

2) SER Open Item No. 46 - OIE Bulletin 79-27

3) NUREG-0737 Item I.D.1 - CR Design Review

4) NUREG-0737 Item II.B.3 - Post Accident Sampling

5) NUREG-0737 Item II.F.1 - Accident Monitoring

6) Additional background information with respect to Jack Notaro's professional qualifications

Item #11 - Supplemental ECCS Calculations with NUREG-0630 Model (Revised Response)

The application of NUREG-0630 to the BWR LOCA is a generic issue for all BWR's. GE and the NRC met on July 8, 1981 to establish a resolution approach and assure LOCA model conservatism and therefore conformance 10CFR50 Appendix K.

The NRC concluded that this issue would be finally resolved pending completion of a sensitivity study by GE to further demonstrate model conservatism.

Submittal of these results to the NRC is targeted for July 31, 1981. It is expected that these studies will confirm the conservatism of the existing model (used in the Shoreham analysis) and that no impact upon the Shoreham LOCA analysis results is appropriate.

Pending final generic resolution of this issue, and the determination of a PCT penalty, if any, the Shoreham specific FSAR Tables 6.3.3-1 (page 1 of 2) and 6.3.3-2 and proposed technical specification Figures 3.2.1-1 and 3.2.1-3 will be temporarily revised to incorporate the maximum potential PCT penalty suggested by the NRC.

This 40° F PCT penalty will be applied by adjusting PCT values to a maximum of 2160° F thus assuring compliance with the 2200° F limit required by 10CFR50.46. A proportional adjustment in MAPLHGR will then be applied to assure acceptable plant operating power levels.

Upon final generic resolution of this issue, the FSAR table and Technical Specification Figures may again be revised to reflect the ultimate solution to this concern.

TABLE 6.3.3-1 (Temporary Revision)

MAPLHGR, MAXIMUM LOCAL OXIDATION, AND PEAK CLAD TEMPERATURE VERSUS EXPOSURE

FUEL TYPE 8CR233

EXPOSURE MWD/T	MAPLHGR kW/FT	P.C.T. DEG-F	OXID FRAC
200.0	11.7	2158.	0.035
1000.0	11.7	2157.	0.035
5000.0	11.8	2159.	0.034
10000.0	11.7	2159.	0.034
15000.0	11.7	2158.	0.034
20000.0	11.6	2158.	0.035
25000.0	11.4	2153.	0.034
30000.0	11.2	2130.	0.028

FUEL TYPE 8CR183

EXPOSURE MWD/T	MAPLHGR kW/FT	P.C.T. DEG-F	OXID FRAC
200.0	11.6	2158.	0.035
1000.0	11.7	2159.	0.035
5000.0	11.8	2158.	0.034
10000.0	11.8	2160.	0.034
15000.0	11.8	2158.	0.034
20000.0	11.7	2155.	0.034
25000.0	11.6	2158.	0.032
30000.0	10.8	2052.	0.021

TABLE 6.3.3-2 (Temporary Revision)

BREAK SPECTRUM SUMMARY

Failure	Break Location	Break Size (Ft^2)	Peak Local Oxidation (%)	Core Wide Metal-Water Reaction (%)	Peak Cladding Temp. (F)
LPCI IV	Recirc Discharge	2.37 (DBA)	Note 3	Note 4	2148(1)
LPCI IV	Recirc Discharge	2.01 (.85DBA)	3.5	0.21	2160 (1)
LPCI IV	Recirc Discharge	1.0	Note 3	Note 4	1918(1)
LPCI IV	Recirc Suction	4.22 (DBA)	Note 3	Note 4	1800(1)
LPCI IV	Recirc Discharge	1.0	Note 3	Note 4	1884(2)
LPCI IV	Recirc Discharge	0.9	Note 3	Note 4	2019(2)
D/G	Recirc Discharge	0.4	Note 3	Note 4	1309(2)
HPCI	Recirc Discharge	0.07	Note 3	Note 4	1474(2)

(1) Large Break Methods
(2) Small Break Methods
(3) Less than most limiting break (3.5%)
(4) Less than most limiting break (0.21%)

July 23, 1981

Item #46 - OIE Bulletin 79-27

LILCO has undertaken a number of efforts to ensure that the consequences of a loss of power to each electrical bus at Shoreham is understood by the operators and addressed in the applicable alarm response procedures. These efforts include a comprehensive design review and training program. The design review and the conclusions reached as a result of the review were summarized in our response to NRC information request 223.90 (ref. SER OI 46, SNRC-566, dated 5/15/81). The basis underlying those conclusions is stated below:

1) All control room instrumentation was reviewed and instrumentation required for shutdown was identified. The electrical bus supplying power to the instrument was reviewed to ensure the information is available in the event of the loss of power (see 223.90,1). Failure of nonessential systems and components will not affect any essential equipment nor the ability to safely shut down the plant. This review was conducted in conjunction with the Cable Separation Analysis and our evaluation of compliance with RG 1.97 rev. 2. A failure effects analysis will be conducted to evaluate the consequences of the loss of power to each Class 1E and non-Class 1E bus. The results will be reflected in operator training and the alarm response procedure.

2) Supplementing this review is a 2-part training program. Generic simulator training provides the operator with event specific (i.e., loss of instrumentation) challenges, designed to ensure that the operator reacts correctly. This supplements Shoreham specific system response evaluations, operator/system walkdowns, and the Shoreham specific alarm response training.

3) Furthermore, the startup test program assures a reliable 4160v power supply to Class 1E electrical loads by a) verifying the proper transfer capability of electric loads between the station service transformers and b) verifying the capability of the emergency diesels to supply power during loss of offsite power conditions.

These efforts contribute to the design and operational controls which ensure that the consequences of a potential loss of power to each control and instrument bus is understood, and that the operator can effectively mitigate a loss of power.

HUMAN FACTORS ENGINEERING
CONTROL ROOM DESIGN REVIEW

LILCO RESPONSE TO NRC AUDIT FINDINGS
SHOREHAM NUCLEAR POWER STATION - UNIT 1

3. ANNUNCIATORS AND AUDITORY SIGNALS

NRC Finding 3.1

The annunciator panel identification scheme is inconsistent and does not follow a conventional sequential order around the main control boards. (1) (5.3)

LILCO Response

As presently designed, each annunciator window within the Control Room has a unique number engraved on the window. Corresponding to each number is an "Alarm Response Procedure." These procedures will be kept in appropriate locations to allow for ready access by the operator. An annunciator front panel identification scheme, consistent with the elementary electrical wiring diagrams, will be completed by fuel load. This will include annunciator panel designations, as well as, a matrix identification scheme on each individual annunciator panel. The alarm response procedures will also have the matrix identification scheme number on the procedure.

NRC Finding 3.2

The controls provided for operator response to the annunciator system provide only SILENCE, ACKNOWLEDGE, and TEST controls. (1)
Recommended: provide SILENCE, ACKNOWLEDGE, RESET and TEST controls for annunciators.

LILCO Response

The current main Control Room annunciator system sequence of operation is standard throughout the LILCO system. A ring-back sequence was deliberately not selected since it was felt that with the large number of alarms in the Shoreham Control Room, the addition of another color and/or flash rate with additional audible signals would cause operator confusion and possible cause incoming alarms to go unrecognized. In addition, part of the alarm response procedure that the operator follows when responding to an alarm is the verification that the actions taken have cleared the alarm. For these reasons, we do not consider it to be necessary or desirable to reconfigure or replace the present Control Room annunciator system to provide a ring-back sequence. No further LILCO action will be taken.

NRC Findings 3.3

No first-out alarm system is provided in the annunciator system to identify trip initiating events. (3)

LILCO Response

The first-out alarm function is provided by the sequence of events log in the plant process computer. We do not consider it to be necessary to reconfigure or replace portions of the present Control Room annunciator system to provide a first out sequence. Enclosure 1 contains a description of the Shoreham sequence of events log.

NRC Findings 3.4

There is no audio or visual annunciator indication to signify that an alarmed condition has cleared. Once an alarm occurs, the annunciator tile remains illuminated until the operator presses the Acknowledge control. Then the tile illumination turns off if the alarm has cleared. (1)

LILCO Response

Duplicate item. See LILCO Response to NRC Finding 3.2.

NRC Findings 3.5

Annunciator tile locations are not identified by a matrix identification scheme with the matrix location code inscribed on each tile. Each tile is number coded but there is no systematic correlation between the number code and the tile location. Tiles are not physically keyed to prevent placement in an incorrect location. (1) (5.3)

LILCO Response

See LILCO response to NRC finding 3.1.

NRC Findings 3.6

The flash rate of annunciator displays (approximately 1.5 - 2 flashes per second) is slower than recommended. (3)
Recommended: 3 - 5 flashes per second flash rate.

LILCO Response

LILCO will modify or replace the existing flasher cards in the annunciator logic cabinets to provide the recommeneded 3 - 5 flashes per second. This will be only the audible alarm associated with that section of the annunciator will be silenced as the flashing windows in that section go to a steady on condition. The test pushbutton has no operational function. It serves only to test the annunciator lamps and logic. Operating the annunciator test pushbu.ton will not cause any incoming alarms to be lost. They will go to a steady on condition after the test has been cleared by depressing the acknowledge pushbutton.

NRC Findings 3.13

Alarm procedures are not keyed to annunciator panel identifiers and annunciator tile matrix coordinates. They are indexed only to the four digit code on each tile which is not systemically related to the tile location. (1) (5.3)

LILCO Response

See LILCO response to NRC Finding 3.1.

NRC Findings 3.14

There is no input to the Annunciator Inoperative alarm when the HPCI and RCIC controls are switched from AUTO to MANUAL. (1)

LILCO Response

The Auto/Manual mode switch for the HPCI or RCIC system flow controllers does not make either system inoperative when the switch is in the manual position. Should an accident signal be received while either of these mode switches is in the manual position, the system logic will automatically revert to the automatic mode of operation by bypassing the flow controller.

NRC Findings 3.15

Localizing quality of audible alarms is not adequate. While the frequencies of the annunciator alarms at the RCC benchboard and at the RWC and RCIC benchboard are different, their separation is not sufficient to provide a clear localizing quality. It is difficult to determine which annunciator panel is alarming when only one alarm is sounding. (1)

LILCO Response

The subject annunciator horns on panels 6Ø2 and 6Ø3 have been verified with the LILCO preliminary Control Room Audit to be at least 2Ø dB(A) above the general ambient level (See General Physics Shoreham Preliminary Human Factors Engineering Recommendations, Table 5.5.1). As stated, the frequencies of the alarms are different. Pending the precise definitions of the degree of differentiation required, LILCO feels that the applicable criteria have been met and that the

6. PANEL LAYOUT (Cont'd)

NRC Finding 6.14

Reactor Building Standby Ventilation System (RBSVS) Chiller Inoperative Alarm Controls are located on the Turbine Building Air Exhaust Panel which is 10-12 feet from the RBSVS panel and RBSVS annunciator panel A2. (3)

LILCO Response

The Manual INOP switch is provided for the operator to annunciate any maintenance operations that make the system INOP, but are not automatically alarmed. Since this switch is non-IE, it is located in the Turbine Building Air Exhaust Section, which is the closest non-IE section to the RBSVS chiller section. This is done for electrical separation.

NRC Finding 6.15

The spring loaded RESET control switch for RBSVS is located high on the VC2 board and must be held down while the operator moves other controls located much lower on the panel and offset to the left and right. This is unreasonably awkward. (3)

LILCO Response

If all automatic initiation signals have cleared, the reset switch must be held in Reset position only long enough to reset a lock-out relay. If automatic initiation signals have not cleared, then turning reset switch to Reset does nothing. No simultaneous operator actions are required in conjunction with the reset switch operation.

NRC Finding 6.16

On the Hydrogen Recombioner Panel, the Leeds and Northrup recorders are too low to read easily. (2) (7.15)

LILCO Response

The subject recorders will be relocated prior to fuel load.

NRC Finding 6.17

On Panel ACH, meters are more than 70 inches above floor level. (2)

LILCO Response

The ACH panel is a back panel which contains the controls for the post-LOCA hydrogen recombiners and for the new primary containment inerting system. The subject meters present information on cooling water flow to the hydrogen recombiners and containment purge flow. With the addition of containment inerting, the hydrogen recombiners become a back-up, long-term mitigation mechanism, thus the subject meters are not of primary importance to immediate actions. The purge system is a back-up feed and bleed dilution mode for the containment atmosphere control and further back-up the recombiners. Finally, panel seismic straightening ribs prevent moving the subject meter to a lower elevation.

NRC Finding 6.18

Top row of chart recorders on HVAC Board (over 1T47 and 1M50) and two meters are too high to be read easily. Poor readability is compounded by glare. (3) (7.1, 7.6)

LILCO Response

These recorders and meters are associated with system controls located on panel VC2 and, due to operational and separation requirements, must remain in their present location.

LILCO will verify that there is minimum glare on meter faces and chart recorder glass after the light diffusing ceiling material is installed. Based on the results of the survey, corrections will be made prior to fuel load.

NRC Finding 6.19

The annunciators for NSSS A/B Isolation and NSSS C/D Isolation read logically from left to right although they are on different annunciator panels. The associated Manual Isolation control buttons are arranged in left to right sequence CDAB. (2)

LILCO Response

The annunciators and/or switches will be relocated to have the same sequence on both. This will be accomplished by fuel load.

NRC Finding 6.20

Containment Purge Control Valves 38 C and D are reversed with respect to their depiction on the mimic above. (2)

NRC Finding 6.25

On the Reactor Control Benchboard (Panel 603) the IRM range switches are mirror imaged on either side of the rod control. The left to right order is ACEG - rod controls - HFDB. (3)

LILCO Response

Duplicate item. See LILCO Response to NRC Finding 4.3.

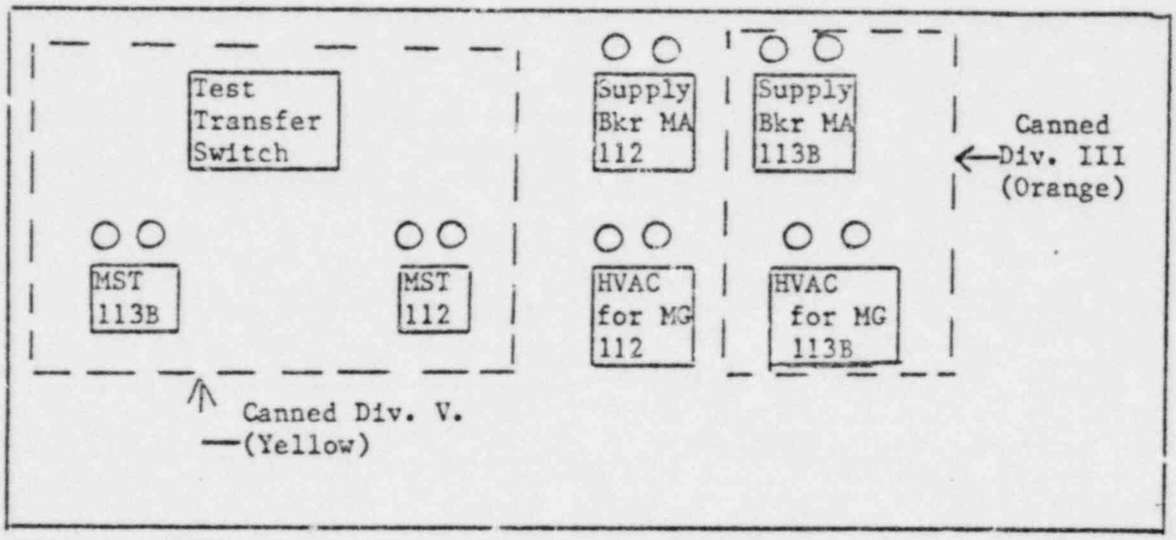
NRC Finding 6.26

Transfer switch MST 113B and related switch MG 113B are separated by switch 112 and a breaker. Similar arrangements are found for switch 111 and switches 113A. (2)

LILCO Response

The layout was designed for electrical separation. See diagram below:

Div. II Section of H11-P601



LILCO will reverse switches MST 112 and 113B and provide color pad enhancement for the entire switch cluster diagramed above. A similar change will be accomplished for Division I switches on panel 601. This will be accomplished prior to fuel load.

7. CONTROL/DISPLAY INTEGRATION

NRC Finding 7.1

On Panel 6Ø3 the SRM A, B, C, and D displays are located too far from the Rod Pull controls for easy readability. (2)

LILCO Response

SRM A, B, C & D displays (counts rate and reactor period meters) on panel 603 will be relocated on the vertical portion of the board. The count rate meters will be located between alarm CRT and core display. The period meters will be located between graphic CRT and core display. The meters will be made vertical type and the exponents will be enlarged. The changes will be accomplished prior to fuel load.

NRC Finding 7.2

The computer printer outputs are too far from the computer console to be read easily. (3)

LILCO Response

Considering the size of the Shoreham Control Room, it is impossible for an operator to stand or sit in any one particular location and have the ability to read every meter and indicator from that position. Annunciator systems are arranged such that the operator must go to the area in which the annunciator and feedback controls are located and read the annunciator, assess the feedback instrumentation, and take appropriate action. The same holds true for the computer printouts. Once the operator selects a particular function out of the computer he must walk over to the computer typers, trend recorders or digital display units to assess the information being transferred from the computer to the readout devices. These typers, trend recorders and digital display units have been located to allow for maximum interface with associated control panel indications and displays. LILCO feels that this general philosophy is appropriate to the Shoreham operation.

NRC Finding 7.3

On Panels 6Ø1 and 6Ø2, the Rosemont SRV pressure indicators are too far from the SRV auto-depressurization controls to be read. (3)

LILCO Response

The operator does not need to read the pressure indicators only to see the trip lights on the front of each module. The operator can tell from the lights how many valves are open or if a valve is open when not required.

NRC Finding 7.4

On Panel 6Ø3 the "Withdraw" or "Continuous Withdraw" push-buttons are located too far from the SRM counts/second and period meters and SRM recorders. (2)

LILCO Response

See LILCO Response to NRC Finding 7.1.

NRC Finding 7.5

Controls related to annunciators on annunciator panel "E" on Panel MCB are located remotely on Panel MXP. (2) (5.4)

LILCO Response

Controls will be relocated prior to Fuel Load using the guidance of the final issued version of NUREG-0700.

NRC Finding 7.6

The Safety Valve Temperature Indicator/Recorder, which provides positive indications of open safety relief valves, is on a back-panel behind Panel 6Ø1. (3)

7. CONTROLS/DISPLAY INTEGRATION (Cont'd)

LILCO Response

The TMI incident has brought serious questions in both the NRC and Nuclear Community as to the ability of a tail pipe temperature indicator to provide a positive indication of an open safety relief valve. Consequently, NRC position papers, ACRS findings, and general Industry evaluations have concluded that a more positive Safety Relief Valve position indication system was required. LILCO responded to NRC requirements to install a positive position indication system in our response to NUREG Ø578. In that response, LILCO indicated that a tail pipe differential pressure system would be installed on each Safety Relief Valve to provide a positive indication of relief valve position. This LILCO position is similar to the system installed at other BWR's. Within our response, LILCO indicated that the tail pipe temperature indication system would be retained as a back up system to be used by the operator to confirm the tail pipe differential pressure indications. Thus, the temperature recorder for the SRV tail pipes will remain on the back panel, and only the common annunciator will be retained on the front panel.

NRC Finding 7.7

The safety relief valves located on Panel 6Ø2 have corresponding annunciator tiles located remotely on annunciator panel G on Panel MCB. (2) (5.4)

LILCO Response

See LILCO Response to NRC Finding 7.5.

NRC Finding 7.8

The controls located under Panel MCB annunciator panel 2Ø9-H relate to annunciators located remotely on annunciator panels 2Ø9-C and 2Ø9-D. (2) (5.4)

LILCO Response

This finding is more accurately written as: "The controls located under Panel MCB annunciator panels 2Ø9C and D relate to annunciators located remotely on annunciator panel H."

Also see LILCO response to 7.5.

NRC Finding 7.9

The Seal Water Pump controls on Panel MCB relate to annunciators located remotely on annunciator panels 2Ø9-A and 2Ø9-B. (2) (5.4)

LILCO Response

The annunciator in question is displaced on annunciator panel 2Ø9-E, while the seal water pump controls and system annunciators are located under annunciator panels A and B.

Also see LILCO response to NRC Finding 7.5.

NRC Finding 7.1Ø

The controls for by-pass valves on Panel MCB under annunciator panel 2Ø9-B relate to remotely located valve position indicators under annunciator panel 2Ø9-F. Separate indicator lights for the by-pass valve controls are needed either above the EHC panel or below the feedwater and condensate mimic. (3)

LILCO Response

Duplicate item. See LILCO Response to NRC Finding 6.11.

NRC Finding 7.11

See LILCO response to NRC Finding 7.5.

LILCO Response

LILCO has generally complied with the criteria to functionally group annunciators with their corresponding systems. LILCO agrees with the finding. However, the final disposition is deferred to the long term review, pending final definition of annunciator system review criteria.

NRC Finding 7.12

Some annunciators are separated too far from their annunciator response controls and are difficult to read from the annunciator control location. (2) (5.4)

LILCO Response

See LILCO response to NRC Finding 7.5.

NRC Finding 7.13

On Panel 6Ø3 the arrangement of selector controls and recorder displays for the IRM/RBM/APRM 2 pen recorders is confusing and does not show which rod block monitor is selected. (3)

LILCO Response

Instrumentation and controls associated with IRM's, APRM's, and RBM's on panel 6Ø3 are state of the art equipment. In addition to color coding of switches and recorders, operator training, both at the simulator and on site, have reenforced this arrangement. LILCO believs that this present arrangement is efficient.

NRC Finding 7.14

The Feedwater Turbine Test group on Panel MCB contains control shapes and colors which are inconsistent with corresponding indicators. (3)

LILCO Response

See LILCO Response to NRC Finding 6.27.

NRC Finding 7.15

On Panel 6Ø2 there is an inconsistent relationship among the valve switches and the corresponding indicator lights in both position and orientation. (1) (9.1)

LILCO Response

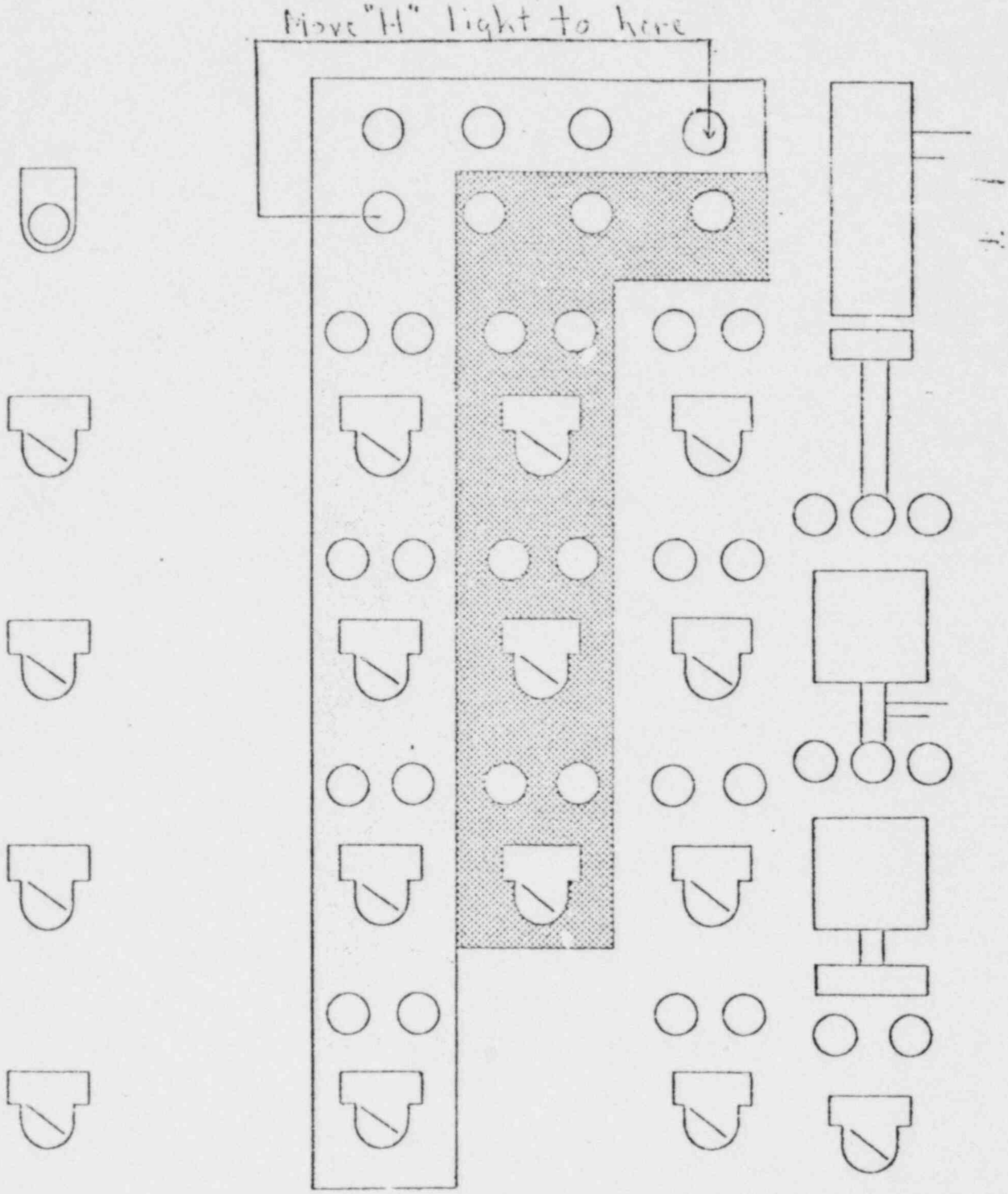
The NRC finding is concerned with the relationship of the switches associated with the ADS and Safety Relief Valves and the corresponding indicator lights. LILCO has identified this problem in their preliminary assessment report as item 9.1. The ADS system will be enhanced through color padding to improve the switch/indication relationship (see enclosure 2). The B solenoid red indicating light for the SRV "H" will be relocated as shown on enclosure 2. This will be accomplished prior to fuel load.

NRC Finding 7.16

On Panel 654, the relationships between switches and indicators is confusing. The left switch corresponds to the bottom light and the right switch corresponds to the top light. (3)

ENCLOSURE 2

SHOREHAM ADS SYSTEM COLOR PADDING



8. LABELS & LOCATION AIDS

NRC Finding 8.1

Some labels are located below the controls they identify. (1)
Example: Panel ACH - Primary Containment Gas Analyzer Control.

LILCO Response

Labels are located above control devices and below displays, except in cases where clear readability requires that they be located in a different location which improves readability.

NRC Finding 8.2

Throughout the Control Room, display labels are not consistently located either above or below the displays they identify. (1)

LILCO Response

See LILCO Response to NRC Finding 8.1.

NRC Finding 8.3

Labels on panel below recorders that protrude from the panel are hidden from view of standing operators.
Example: MXP Panel and Panel 603 (1)

LILCO Response

See LILCO Response to NRC Finding 8.1.

NRC Finding 8.4

Label on reactor water cleanup return J - switch that states TO REACTOR VESSEL is misleading since the return flow path is through HPCI, RCIC, and feedwater lines to reach the reactor vessel. (1)

NRC Finding 8.28

The use of red/black color coding to denote the alternate ranges for each pen of the 2 pen IRM/RBM/APRM recorders and to correlate the color of each recorder pen trace with its associated range selector control is confusing. (3)

LILCO Response

Duplicate item. See LILCO Response to NRC Finding 7.13.

NRC Finding 8.29

On Panel 603, the color coding of Rod Drift, Rod Motion Override, RODS Status, RIPS Status, Rod Motion Blocks, and Rod Insert and Withdraw Push-buttons are not correlated with other controls and displays. (2)

LILCO Response

Relative to the NRC finding regarding color coding within the Rod Select Module on Panel 603 the following is offered:

a) The two white lights dealing with the selection of the Stabilizer Valves Set A or Set B are controlled by the black push-button labeled Valve Selector. The use of a black push-button to select the Stabilizer Valves for operation is consistant with selector push-buttons used at other points on the control board. The use of white indicators to show which set of stabilizer valves has been put into operation is also consistant with other status indications, since this is not an off normal or alarm function.

b) The following yellow push-buttons will be changed to black push-buttons prior to fuel load.

1. Accumulator Trouble Acknowledge
2. Continuous Insert
3. Continuous Withdraw
4. Insert
5. Withdraw
6. Rod Drift Test
7. Rod Drift Reset

The use of black push-buttons for acknowledge and reset as well as test operations is consistent with the colors utilized for similar functions at control stations at other points in the Control Room. The rod motion controls dealing with continuous insert and withdrawal and normal insert

NRC Finding 9.7

Panel 6Ø3 CRTs are mounted with the bottom of the screen 69 inches above floor level making it difficult to read the upper portion of the displays. The location and orientation of the CRT displays on Panel 6Ø3 combined with the curvature of the CRT screens make it impossible to read the upper edge. Recommended maximum mounting height is 61 inches. (3)

LILCO Response

Readability of the CRT displays on panel 6Ø3 is within the 11Ø° vertical visual field coupled with the 5th to 95th percentile eye height data.

NRC Finding 9.8

Combined with NRC Finding 9.7.

NRC Finding 9.9

The wording used to identify alarms displayed by the computer and the wording used on annunciator tiles differ for the same alarm conditions. (1)

Example: Annunciator - CRD HYDRAULICS TEMP HI
Computer - CONTROL ROD DRIVE TEMP ALARM

LILCO Response

The wording differences between alarms displayed by the computer and the annunciator windows will be corrected prior to fuel load such that the alarm windows and computer printout convey the same meaning.

NRC Finding 9.1Ø

There is inconsistent use of color conventions between the CRT displays and conventional control board and annunciator displays. (3)

9. COMPUTERS AND CRT DISPLAY (Cont'd)

LILCO Response

Computer graphic display color schemes are limited by hardware restraints within the 4010 Video Display Subsystem. Where possible, every effort was made to match the color conventions to the control board schematics. In those cases where duplicate colors were not available, consistent substitutions were employed.

NRC Finding 9.11

Unconventional symbols are used on the CRT displays. (3)

LILCO Response

Video hardware limitations (available symbols, CRT resolution) does not always allow the generation of conventional symbols. A review will be made of all displays as part of a long-term Control Room review and modifications (symbol revision, additional labels) will be made where appropriate.

NRC Finding 9.12

There is inconsistent use of color coding between different computer generated CRT displays. (3)

LILCO Response

Color coding of variable data (analog, digital, and calculated values) is consistent for all displays. Inconsistent color coding in the background portion of the more detailed displays has been made for clarity as requested by operations personnel.

NRC Finding 9.13

It is difficult to read printer printouts of decimal data because decimal points ar not aligned on the printouts. (1)

LILCO Response

The software will be modified prior to fuel load to allow for the alignment of the analog alarm message decimal points.

NRC Finding 9.14

Some annunciator alarams for auxiliary systems are not shown by the computer alàrm display. (3)

II.B.3 POST-ACCIDENT SAMPLING

LILCO Position

The following describes the post accident sampling system and facility.

DESIGN BASES

The design basis of the post accident sampling system (PASS) and facility (PASF) is to provide site personnel with the capability of promptly drawing and analyzing samples (less than three hours) under accident conditions while insuring that the radiation exposure is 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.

Sample supply lines are routed such that an isolated auxiliary system will not have to be started up to provide a sample.

All sample point isolation valves are Class 1E, seismic valves qualified to IEEE-323-1974 with accident isolation signals, and override capability.

Sample system equipment has been purchased to Regulatory Guide 1.97, Rev. 2, Qualification Criteria--Category 3.

SYSTEM DESCRIPTION

The post accident sampling system, 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). This system will provide the following samples:

1. Atmospheric Samples

   Locations: 2 redundant wetwell air samples
   2 redundant drywell air samples

   Types: Hydrogen/oxygen samples

   Atmosphere for isotopic analysis

   Design Basis: A representative sample is assured by by drywell fans, atmospheric diffusion, redundant containment sprays and sample point placement. Furthermore, line size and line routing was selected to minimize sample transport time and volume of sample outside the primary containment, which in turn aid in keeping the samples representative.

Line plugging is not expected but in the event of a plugged line, backpurge capability exists and redundant sample points exist.

2. Liquid Samples

Locations: 2 redundant suppression pool samples through the residual heat removal system. 1 reactor vessel sample from the jet pump number 10 high pressure connection.

Types: Pressurized reactor coolant sample
Suppression pool sample via RHR

Design Basis: Initially a jet pump sample will provide all the isotopic information necessary to determine type and degree of core damage. Under all accident conditions, liquid sample will carry nonvolatile isotopes and in combination with isotopic analysis of wetwell atmosphere will provide prompt quantification of nuclides that are indicators of the type and degree of core damage.

The PASS will provide the following capability:

1. Prompt quantification (less than 3 hours) of certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperature), and nonvolatile isotopes (which indicate fuel melting). See Table II.B.3-1.

2. Chemical analyses within 3 hours of commencement of the post accident sampling operation. (Note - gas chromatography is not used.)

3. On line gross gamma activity level for both monitoring and dilution control. The dilute sample to be isotopically monitored for core damage indicative radionuclides. This method of dilution eliminates standard analytical and procedural errors.

4. Dilution of liquid, reactor coolant gases or containment atmosphere samples by either volumetric or feed and bleed methods for off-line gamma spectrum analysis. Dilution monitoring for feed and bleed dilution is by gross gamma activity level as discussed above.

5. Isotopic analysis of any sample taken by the post accident sample system, either liquid or gas.

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

7. Pressurized full strength grab samples of reactor coolant for analysis by an outside laboratory if required.

8. Measurement of the total gas concentration in the reactor coolant by the observed pressure, temperature, and volume relationships of the sample.

9. The ability to perform all sampling and analysis assuming a highly radioactive initial sample (Regulatory Guide 1.3 source term).

10. The ability to perform a full system flush at the conclusion of each sample evolution to minimize the possibility of plateout or cross contamination of any future sample.

11. Periodic testing for technician training and familiarity and to ensure reliable system operation.

Both diluted and undiluted grab samples are provided. The diluted grab samples are used for radioactive spectrum analysis of liquids and gases. An undiluted grab sample capability exists for laboratory analysis as a back-up for the on-line equipment. Certain chemical analyses can only be accurately performed on undiluted coolant samples off-site. Provision has been made in the design of the sampling facility to obtain such undiluted pressurized coolant samples at the required frequency. An agreement and plan to perform such analyses at an off-site facility is being actively pursued, including design of a mutually acceptable sample container. This back-up capability will be implemented at the earliest practical date, but not necessarily prior to fuel load. Equipment will be provided to transplant the sample in a manner that will minimize exposure consistent with NRC guidelines.

SAMPLE BUILDING (PASF)

The PASF is located on the south side of the reactor secondary containment at zero azimuth directly adjacent to the truck access 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. 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:

1. Sample enclosure,

2. Sample collection room,

3. Support (HVAC) room, and

4. 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.

HABITABILITY OF THE PASF

The PASF is designed to limit radiation exposure of the operator to less than 5 Rem whole body and 75 Rem to the extremities.

1. The PASS Operator is able to draw and analyze a prompt sample (less than 3 hours) under accident conditions (Regulatory Guide 1.3 source term combined with a conservatively assumed ground level release meteorology).

2. Background radiation levels are maintained low enough to assure proper function of all radiological equipment. For the isotopic analysis equipment, a redundant counting area is available on the opposite side of the Reactor Building should background radiation in the PASF exceed the design specification for the equipment.

RANGE AND ACCURACY OF CHEMICAL ANALYTICAL EQUIPMENT

1. Conductivity

   Range - 0-50, 0-500, 0-5000 micromhos/cm

   Accuracy - meter - ±2.5% of full scale range
   cell - ± 1.0% of full scale range

   Stability - ±0.5% 0^{o} - 50^{o} C ambient
   ±0.1% 105 - 135 VAC

2. Boron/pH

   Range - 10-6500 ppm of Boron
   0-14 pH

   Accuracy & Repeatability - Boron ±5%
   pH ±0.01

   Interferences - Lithium Hydroxide not to exceed 50 ppm

3. Chloride

   Range - 0.1 - 100 ppm of chloride

Accuracy & Repeatability	- ±5% or 0.1 ppm - larger of two
Interferences	- total elemental iodine not to exceed 10 ppm

RANGE AND ACCURACY OF RADIOLOGICAL ANALYTICAL EQUIPMENT

1. Gross Gamma Detectors

   Range - 8 decades 0.1 mR/hr to 10^7mR/hr

   Accuracy - ±10% for 80 kev to 3 mev

   Sensitivity - 0.1 mR/hr

2. Detectors for Isotopic Analysis (later)

TABLE II.B.3-1

RELEASED NUCLIDES FOR SPECIFIC CASES

CLADDING FAILURE

Kr-85
Xe-133
I-131
(Rb-Ru)-106
Cs-137

FUEL OVERHEATING

Cs, I, + more Xe and Kr
I-131
Cs-137

FUEL MELTING

I-131
Cs-137
+ other nonvolatiles such as Ba-140

NOTE: Radionuclides in this table were selected based on predicted and experimental release quantities and the ability to detect the radionuclides by gamma spectography.

REFERENCES:

1. TMI: A Report To The Commissioners And To The Public (Rogovin), dated April 1979.
2. NSAC-24 "TMI-2 Accident Core Heat-Up Analysis" dated January 1981.

LILCO will provide post-accident sampling of the Shoreham Reactor via a sample line connected to the high pressure flow sensing line of jet pump number 10. There is no need to extract a reactor coolant sample from any other region of the reactor for the following reasons:

(1) General Electric has evaluated reactor coolant sampling techniques in Section 7.4.1 of NEDO-24782, "BWR Generic Sampling System Conceptual Design." Their conclusion is that the jet pump instrument line will provide a representative sample and is well protected from damage or blockage during an accident. All other potential sampling points would either provide a non-representative sample or would be prone to damage or blockage.

(2) The PASS is not required to provide useful sample analysis results until as much as three hours after the initiating event. In that time frame, there will not be a major concentration of released waterborne fission products in the core region versus the annulus outside the shroud or the suppression pool where our samples can be taken. During a LOCA, waterborne fission products will be flushed into the suppression pool through the break and recirculated to the vessel by ECCS. During non-LOCA events where fuel failure has occurred, once the reactor coolant level has been restored, minimal coolant make-up will be required due only to decay heat boil-off three hours after shutdown. Natural circulation through the core will assure good mixing of water in the core and annulus.

(3) Once vessel depressurization has occurred, either due to LOCA blowdown or operator action, reactor coolant samples cannot be drawn directly from the vessel, but must be taken from one of the following locations:

a. an RHR loop operating in the shutdown cooling mode, if post accident activity is low,

b. an RHR loop operating in the suppression pool cooling mode while using special shutdown procedure III (see FSAR Section 3C.3.4.3.2), or

c. directly from the suppression pool while using special shutdown procedure III.

It should be noted that special shutdown procedure III, which utilizes a circulating coolant path from the suppression pool, through the RPV, and back to the pool via held-open safety relief valves, is assumed after an accident if the TID source terms are present. This method of shutdown cooling minimizes the activity of the coolant outside containment by an approximate factor of ten due to pool dilution of coolant. As a secondary consideration, this method also provides for representative sampling of reactor coolant via locations b and c described above.

Therefore, in the long run, the location of the RPV reactor coolant sample tap is not critical due to limited useful service time.

1) Hardware for direct sampling from the station vent exhaust is provided for both normal and accident conditions. For accident conditions in other than the reactor building, the radioiodine and particulate samples are available from the station vent exhaust flow. The reactor building is vented via RBSVS which has noble gas monitors and radioiodine/particulate filters. In the event of an accident within the reactor building, the station vent flow (if any) is a composite of vent flows from the Radwaste Building and Turbine Building exhaust, and outside air to maintain constant fan performance. Each of the contributions from the Turbine Building and Radwaste Building is individually monitored and sampled for radioiodine and particulates. The effects of an event within the reactor building will not affect the atmosphere within this building and the sampling system is adequate for the service, and meets the criteria of II.F.1.

2) Sampling media for radioiodine has been specified to achieve greater than 90% efficiency for iodine collection and greater than 90% for particulate greater than 3 micron in size, consistent with Table II.F.1-2, Attachment 2. Capability for continuous sample collection of radioiodines to the design basis range (ie. 10^{+2} uCiCC) will be provided by either a charcoal or silver zeolite sample media as appropriate.

3) Flow control of sample is provided to maintain a constant flowrate for collection of particulates and radioiodines. Ventilation system flowrate is constant for all systems monitored with the exception of the station vent. In this case, the reactor building exhaust is isolated following an accident within the reactor building, however, outside air is introduced to maintain a constant working point on the fan and thereby a constant velocity through the station vent. Based on this the current design will comply with $\pm$ 20% requirement of II.F.1, Attachment 2, effectively providing a flow past the sample probe which approximates isokinetic conditions.

4) Shielding design basis for sampling media associated with accident releases via the station vent and RBSVS discharge are based on RG 1.97 Draft 3 (Oct. 1980, page 34 note 13). The design basis is consistent with the criteria of Table II.F.1-2.

5) ANSI N13.1 is a design criteria imposed on the supplier of monitoring/collection equipment and an installation criteria imposed on the sample lines interconnecting the exhaust flow ductwork and sampling/monitoring equipment. Implementation of this criteria as practical will ensure a representative sample in accordance with Table II.F.1-2.

6) Procedures will be prepared for conducting all aspects of handling, transport and analysis of samples to assure that occupational exposures comply with the limits of II.F.1 Attachment 2. These procedures will be submitted as part of the plant procedure submitted prior to fuel load. As previously noted, the shielding design basis for sampling media associated with accident releases via the station vent exhaust and RBSVS discharge are consistent with the criteria of Table II.F.1-2.

7) A human factor analysis will be performed by LILCO as part of the Long Term Control Room Review program in accordance with NUREG-0700.

8) As noted in Item 2, sampling media have the capability to adsorb or retain iodines and particulates as stated in Table II.F.1-2.

JACK A. NOTARO
Operating Engineer
Long Island Lighting Company

Assigned as Operating Engineer of the Shoreham Nuclear Power Station in July, 1978. Responsible for the development and implementation of the Station's operational activities including the direction of day-to-day operation of the unit; startup, operation and shutdown of all station equipment; implementation of initial, requalification, and replacement training programs for licensed and unlicensed operators; and development, review, and implementation of the operations section of the Station Operating Manual.

Graduated from Brooklyn Technical High School in 1965. Graduated from City College of New York in 1970 with a Bachelor of Mechanical Engineering Degree. Received a Masters of Business Administration Degree in 1974 from Adelphi University.

Completed the General Electric Company Boiling Water Reactor Simulator Program in July, 1976, and obtained certification as a Senior Reactor Operator.

Completed the following industry seminars and training programs:

a) BWR Design Orientation - General Electric Company
b) BWR Technology - General Electric Company
c) BWR Observation Training - General Electric Company
d) Nuclear Power Plant Technology - General Physics Corp.
e) Radiation Protection - LILCO Evening Institute
f) Basic Health Physics - Brookhaven National Laboratory
g) Vibration Analysis - IRD Mechanalysis, Inc.
h) Statics, Strength of Materials, & Dynamics - LILCO Evening Institute
i) Management of Maintenance Storekeeping & Inventories - Management Dynamics Institute
j) QA for the Nuclear Industry - Stat-A-Matrix and General Physics Corp.
k) Inservice Inspection & QA During Operations - Southwest Research Institute
l) Basic Radiography - Convair Division of General Dynamics
m) Magnetic Particle & Liquid Penetrant Testing - Magnaflux Corp.
n) Basic Ultrasonics - Automation Industries
o) Nuclear Power QA - Long Island Section of ASQC
p) Inservice Inspection Symposium - Mirror Insulation
q) Operations Quality Assurance - Stat-A-Matrix
r) Fire Fighting Training - Suffolk County Fire Department
s) Limerick Simulator Capability - General Physics Corp.
t) Simulator Refresher Training - General Electric Company

<u>June 1981 - August 1981</u>

Assigned to the Operations Section of the Millstone Nuclear Power Station. The scope of this assignment included power operation training at greater than 20% power. The assignment encompassed three months of actual hands-on experience in a two-month calendar period.

Participated in weekly and monthly routine BOP and NSSS system surveillance testing. Participated in high risk I&C and Operations equipment and system surveillance testing. Witnessed TIP traces and conducted heat balances, core flow calculations and subsequent nuclear instrumentation calibrations. These calculations were conducted with and without the main computer available. Participated in power downs from 100% power to complete control rod repositioning and repairs to main condenser cross-over valving. Assisted in maintaining power at less than 25%, as required by Tech Specs, as a result of main computer problems. Witnessed implementation of emergency notification procedures.

Manipulated controls for power downs, return to power, Tech Spec LCO's, control rod repositioning, and stuck control rod surveillance testing. Witnessed and participated in half scram and full scram recoveries, subsequent investigations, evaluations and notifications.

In addition to the above, attended daily Plant Manager's and Unit Superintendent's meetings, Operations Department meetings, Plant Operations Review Committee meetings, shift staffing, planning and scheduling evaluations.

March 1981 - May 1981

Assigned to the Operations Section of the Millstone Nuclear Power Station for the completion of the Unit 1 refueling outage. The scope of this assignment included refueling, cold shutdown to greater than 20% power, and greater than 20% power to cold shutdown. The assignment encompassed three months of actual hands-on experience in a two-month calendar period.

Participated in all significant pre and post refueling outage surveillance testing and inspections. Actively took part in refuel bridge operations including control rod removal and replacement, channeled and dechanneled fuel movements, core inspections and verifications, dropped fuel bundle evaluations and recovery. Assisted in the evaluations and calibrations resulting from abnormal nuclear instrumentation indications. Participated in integrated leak rate testing, primary system hydrostatic pressure testing and drywell inspections, assessed system status and return to normal. Conducted portions of pre-criticality testing including control rod functional, sub-critical checks and friction testing. Actively took part in returning the unit to service from cold shutdown to greater than 20% power including manipulation of controls during plant heat-up.

In addition to the above, participated in daily outage coordination meetings, Operations Department staff meetings, Plant Operations Review Committee meetings, shift staffing and scheduling evaluations.

November 1980

Completed one-week simulator refresher training program conducted by the General Electric Company at the BWR Training Center in Morris, Illinois. The scope of this program included equipment and system surveillance testing, integrated system operation, approach to and achievement of criticality, plant heat-ups, transfer to run, BOP and NSSS system operations, casualty/transient evaluations and recoveries.

May 1979

Completed one-week simulator refresher training program conducted by the General Electric Company at the BWR Training Center in Morris, Illinois. The scope of this program was the same as that described above.

April 1979 - May 1979

Completed the 160-hour General Electric Company Observation Training Program at Commonwealth Edison Company's Dresden Nuclear Power Station. Modification of the standard observation training program was effected in this instance including direct assignment to Dresden Operations and clearance for unescorted access.

Dresden Unit 2 was returning from a refueling outage and Unit 3 was returning from a forced outage to replace the main transformer during this training assignment.

On Unit 2, observed significant pre and post refueling outage surveillance testing. Witnessed integrated leak rate testing. Participated in the primary system hydrostatic pressure test and drywell inspections. Observed preparations for and accomplishment of approach to criticality, criticality, plant heat-up, transfer to run, placing the main turbine generator in service and power operation. Participated in daily refueling outage coordination meetings, Operations Department and shift crew meetings.

On Unit 3, observed significant post modification surveillance testing. Participated in the primary system hydrostatic pressure test and drywell inspections. Observed preparations for and accomplishment of approach to criticality, criticality, plant heat-up, transfer to run, placing the main turbine generator in service and power operation. Witnessed half and full scram recoveries. Manipulated controls to reduce power from 700MW to 200MW in preparation for stator cooling system filter replacement.

August 1978

Assigned to the Vermont Yankee Nuclear Power Station to observe startup of the unit following a refueling outage. Witnessed the completion of the integrated leak rate test. Witnessed the primary system hydrostatic pressure test and took part in the drywell inspection. Observed preparations for and accomplishment of approach to criticality, criticality, plant heat-up and transfer to run. Witnessed half scram recovery during plant heat-up.

March 1973 - July 1978

Assigned to the Shoreham Nuclear Power Station in the Quality Assurance Section and subsequently promoted to Station Operating Quality Assurance Engineer responsible for the Section in July, 1974.

Responsibility included initial development of the operational quality assurance program. Responsible for all aspects associated with its implementation at the station including reviews, audits, surveillance, inspections, selection and training of personnel, development of procedures and instructions, and the utilization of consultants and contractors. Additional responsibilities included licensing and inspection activities associated with the U.S. Nuclear Regulatory Commission and interfacing with external and internal organizations required to implement the operational quality assurance program.

May 1976 - July 1976

Completed the three-month General Electric Company BWR Simulator Training Program. This program was conducted at the BWR Training Center in Morris, Illinois. Obtained Senior Reactor Operator Certification. (This Senior Certification would have been granted even if the current 80/70 pass/fail criteria were in effect at the time.)

March 1976 - April 1976

Completed the five-week General Electric Company BWR Technology Course.

January 1972 - March 1973

Assigned to the Electric Production Department Staff. Assigned duties included maintenance scheduling, manpower allocation, equipment testing, station performance analysis and special projects.

June 1970 - January 1972

Assigned to the Maintenance Section in the Northport Power Station (two 400MW oil-fired units). Assigned duties included assisting in outages of both a scheduled and forced nature as well as maintaining plant equipment and systems, and completing special projects.

A member of the American Society for Quality Control, and past member of the Edison Electric Institute - Quality Assurance Task Force (EEI-QATF) and the EEI-QATF Operations Subcommittee.