LILCO
LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
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SNRC-503

August 29, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Light Water Reactors, Branch 4
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SHOREHAM NUCLEAR POWER STATION - UNIT 1
RESPONSE TO NUREG 0578 REQUIREMENTS

Reference: D. B. Vassallo letter to "All Pending License Applicants", dated November 9, 1979

D. B. Vassallo letter to "All Pending Operating License Applicants", dated September 27, 1979

Dear Mr. Denton:

Forwarded herein are fifteen (15) copies of Long Island Lighting Company's commitments to meet the requirements outlined in NUREG 0578 "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" as clarified in NRC letters, dated September 27 and November 9, 1979.

We are currently evaluating the requirements of NUREG 0694 "TMI-Related Requirements for New Operating Licenses" and plan to address all the issues contained therein in a separate submittal to the NRC in the near future. However, since NUREG 0694 envelops the requirements of NUREG 0578, we hereby request your concurrence and/or any comments on the enclosed material as soon as possible.

We consider that the actions taken at Shoreham, as described in the enclosed document, when finalized, will incorporate the Lessons Learned from the TMI-2 Accident.

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If you have any questions, please do not hesitate to contact this office.

Very truly yours,

J. P. Novarro
Project Manager
Shoreham Nuclear Power Station

LG/mt
Enclosures

cc: Mr. J. Higgins, NRC Site Trailer

SHOREHAM NUCLEAR POWER STATION

DOCKET NO. 50-322

RESPONSES TO

NUREG 0578 SHORT TERM LESSONS LEARNED REQUIREMENTS

AUGUST 1980

LILCO
LONG ISLAND LIGHTING

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INTRODUCTION

On March 28, 1979 an accident occurred at the Three Mile Island Nuclear Power Plant - Unit 2 (TMI-2). Shortly thereafter, the Nuclear Regulatory Commission (NRC) formed the Lessons Learned Task Force to identify and evaluate safety concerns originating with the TMI-2 accident that would require licensing actions for operating reactors, pending operating license and construction permit applications. On July 19, 1979, the NRC issued NUREG-0578 "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations". During the subsequent reviews of NUREG-0578 by the NRC and the ACRS, the NRC issued a letter on September 27, 1979, on the subject of Follow-up Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island - Unit 2 Accident. On November 9, 1979, the NRC issued clarifications to some of the recommendations of NUREG-0578 resulting in modifications to some of the implementation commitments.

Contained in this report are Long Island Lighting Company's Responses and Commitments regarding the implementation of the requirements of NUREG-0578, as clarified in the above-mentioned letters, in the Shoreham Nuclear Power Station. NUREG-0694 "TMI-Related Requirements For New Operating Licenses" is currently under evaluation. Responses to requirements in NUREG-0694 not addressed in this report will be forwarded via a separate submittal to the NRC in the near future.

INTRODUCTION (continued)

The BWR Owners' Group is actively engaged in the evaluation of many requirements of NUREGs 0578 and 0694. Accordingly, LILCO will continue to monitor closely and participate actively in the efforts of the Owners' Group as well as the NRC and other industry groups.

RESPONSES

The LILCO Responses to NUREG 0578 Requirements are contained in individual sections of this report. Each Section has been numbered with the same numbering sequence used in the original NUREG 0578 document. The additional four ACRS concerns appear at the end of the report. For completeness and information, each Section contains the NRC clarification and the BWR Owners' Group Discussion and Implementation Criteria. Information referenced in this report has been provided with the applicable section, wherever possible.

2.1.1 Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWRs

NUREG 0578 POSITION:

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

NRC CLARIFICATION:

Pressurizer Heater Power Supply

1. In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class IE division power supply.

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situation.

   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.

   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

7. Being non-class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See item 5.b. above)

Power Supply for Pressurizer Relief and Block valves and Pressurizer Level Indicators

1. While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

3. Any change over of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirement should be capable of being manually connected to the emergency power sources.

BWR OWNERS' GROUP DISCUSSION:

As discussed in NEDO-24708, natural circulation in the BWR is strong and inherent in all off-normal modes of operation, independent of any powered system, as long as sufficient inventory is maintained. This is because even in normal operation the BWR is essentially an augmented natural circulation machine. Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure (the BWR does not have a pressurizer). Thus there is no need for emergency power to maintain natural circulation or to keep the system pressurized.

The power-operated relief valves in BWR's are already powered by emergency power. They have no block valves.

The reactor vessel level indication instrument channels for safety system activation and control are already powered by emergency power.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

For the reasons stated above, there is no need for action in response to Recommendation 2.1.1 for any General Electric BWR.

LILCO'S RESPONSE:

LILCO endorses the BWR Owners' Group position. This requirement is not applicable to BWR plants such as Shoreham.

2.1.2 Performance Testing for BWR and PWR Relief and Safety Valves

NUREG 0578 POSITION:

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves.

NRC CLARIFICATION:

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70.

2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.

3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.

4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.

5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

6. A description of the test program and the schedule for testing should be submitted by January 1, 1980.

7. Testing shall be complete by July 1, 1981.

BWR OWNERS' GROUP DISCUSSION:

The BWR Owners' Group has been performing detailed analysis and evaluation of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2 assuming single active component failure or single operator error. The results and conclusions of this work is anticipated to be submitted to the Staff in September 1980. In addition, the Owners' Group has acknowledged the alternate shutdown cooling event and has committed to flow testing of S/RVs under the low pressure liquid conditions anticipated for this event.

LILCO'S RESPONSE:

LILCO is participating in the S/RV performance verification test program being conducted on a generic basis by the BWR Owners' Group. The S/RVs utilized by Shoreham are being included in the test program. The Owners' Group has authorized General Electric Co. to proceed with the program which provides for facility fabrication and testing to be performed by Wyle Laboratories. Further details regarding this test program will be submitted to the NRC by the Owners' Group.

General Electric, under contract with LILCO, has performed low pressure S/RV testing for Shoreham. This test was performed to demonstrate an alternate means of core cooling using the S/RVs as back up to shutdown cooling. The result of this test will be shared with the BWR Owners' Group and submitted to the NRC.

2.1.3.a Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs

NUREG 0578 POSITION:

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

NRC CLARIFICATION:

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis and action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualification.

5. The position indication should be qualified for its appropriate environment, (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environment qualification program should be provided.

BWR OWNERS' GROUP DISCUSSION:

BWR safety and relief valves (S/RV) are arranged in three ways in the various operating reactors:

1. Valve discharges piped to the containment suppression pool;

2. Valve discharges manifolded and piped to suppression pool;

3. Discharging directly to the drywell free volume, in pressure suppression containments, or to the containment free volume in dry containments.

The configuration of the valve discharge, and the operator's ability to diagnose and act on stuck-open valve events, will determine what information is to be provided in the control room. The environment experienced by the installed instrumentation during a stuck-open valve event will determine the proper qualification requirements.

Valve Discharges Individually Piped to the Suppression Pool

All dual-function safety/relief valves and most relief valves are configured this way. Given a stuck-open valve, the containment pressure will not increase because of the submerged discharge. There is benefit in direct indication, not only because the operator would be given an early warning of S/RV discharge, but because he can attempt to reset a stuck-open valve from the control room. Most such valves have no external stem, which precludes direct position indication.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

Valve Discharges Individually Pipes to the Suppression Pool

The Owners' Group considers two types of monitoring to be acceptable methods of positive valve indication: pressure switches in the valve discharge lines and acoustic monitors. A suitable pressure switch system is outlined in the Appendix (enclosed herein), in response to an NRC request in the September 24, 1979, Region I meeting.

Either type of system will be designed to the following broad requirements:

1. There will be at least one sensing device per discharge line;

2. Sensing devices may be either inside or outside the drywell;

3. Sensing devices and other components need not be qualified for a LOCA (pipe break) environment, but only for the environment expected during S/RV discharge to the suppression pool;

4. All components will be seismically qualified;

5. The system will be powered by one division of emergency power;

6. With sensing devices inside the drywell, non-class IE electrical penetrations may be used if insufficient IE penetrations are available.

LILCO'S RESPONSE:

There are a total of eleven (11) dual function safety relief valves (S/RV) in the Shoreham Reactor System. The S/RVs installed in this facility are of the Target Rock two-stage pilot operated design. Direct main stem position indication is not accessible in a valve of this type. Accordingly, positive position indication will be provided utilizing pressure transmitters on each S/RV discharge line.

The discharge of each safety/relief valve is independently piped to approximately five (5) feet from the bottom of the suppression pool. The calculated steady state pressure near the valve discharge is in the range of 300 psig when the valve relieves at set pressure. This pressure is sufficiently high that a positive and unambiguous signal is available with ample margin for tolerances in calibration and variance in line pressure. When a valve recloses, pressure will return to normal in a fraction of a second. Thus, pressure measurement does not have the slow response time which characterizes discharge pipe temperature monitoring instrumentation. Since each valve discharge is independently piped, the pressure signal provides unique indication for the associated valve.

Nonredundant safety-grade instrumentation will be provided to monitor pressure in the discharge pipe of each safety/relief valve. The transmitters will be located in the secondary containment and connected to the S/RV discharge piping by instrument lines penetrating the primary containment. Individual display and trip set point instrumentation will be provided for each safety/relief valve in the main control room. A common alarm will also be provided in the control room to promptly alert the operator when any S/RV is open. The display instrumentation will be located as close as possible to the safety/relief valve control station in the main control room.

In addition to being qualified for the environment expected during events resulting in safety/relief valve discharge to the suppression pool, the instrumentation will meet seismic Category I requirements in accordance with IEEE 344-1971 and be powered from a Class IE power supply.

The existing temperature monitoring instrumentation will be retained for its original function, detection of valve leakage conditions as backup/confirmatory indication for the pressure instrumentation being provided.

APPENDIX to OWNERS' GROUP POSITION 2.1.3A

USE OF DISCHARGE PIPING PRESSURE SWITCHES

FOR

S/R VALVE POSITION MONITORING

Main stem position is not accessible on the manufacturer's designs of safety/relief valves in BWR service. General Electric has evaluated several concepts including magnetic or proximity switches, acoustic devices, temperature, and pressure switches.

The use of pressure switches on the discharge lines has been selected as the most simple, direct and proven technique for monitoring valve position. The Safety/Relief valve discharge is piped to the torus, discharging below the water line. Pressure near the valve discharge can be straightforwardly calculated and tested; it is in the range of 250 psig when the reactor is at rated pressure. This pressure is sufficiently high that a positive and unambiguous signal is available with ample margin for tolerances in set point calibration. When the valve re-closes, pressure returns to normal in a fraction of a second. Thus a pressure signal does not have the slow response time which characterizes temperature monitoring.

Test data are available confirming the transient and steady-state response of S/RV discharge line pressure. These data were obtained during extensive in-plant measurements of suppression-pool loading resulting from safety relief valve actuations. These test data confirm the analytical basis for selection of set points.

Pressure switches are available in industry which are suitable for this service. Similar devices are used routinely for the protection of plant and equipment. Plant personnel will be familiar with the calibration, testing, and maintenance of these devices. No development testing is required to prove a satisfactory device, other than qualification tests which would be required for any device.

With the use of pressure switches, no device is mounted on or near the safety/relief valve. The technique will work for all types of piped BWR safety/relief valves in service. It will have no effect on valve performance. The pressure switches may be located at some distance from the safety/relief valve where they will not be subjected to severe temperature or vibration conditions. Where suitable piping penetrations are available it is possible to locate the switches outside the drywell.

The pressure switches will be qualified for a 212°F, 100% humidity environment. This is adequate for the intended service even if the pressure switches are inside the drywell because actuation of the S/R valves, inadvertent or planned, will not cause these environmental conditions to be exceeded. In the event of a small pipe break the safety/relief valves in the ADS system would be initiated early in the transient, before degradation of the switches could have occurred. In the event of a large pipe break the safety/relief valves are not required to operate. No failure mode has been identified that would result in an erroneous indication that the valve was open.

The signals from pressure switches may be interfaced with indicating lights, control room annunciators, an event counter, or the process computer. Any one or all of these functions may be implemented. Each safety/relief valve can be monitored independently of the other valves.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

NUREG 0578 POSITION:

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used in developing these procedures, and a schedule for installing the equipment shall be provided.

NRC CLARIFICATION:

1. The analysis and procedures addressed in paragraph one above will be reviewed and should be submitted to the NRC for review.

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety grade temperature input from each hot leg (or use of multiple core exit in T/C's) are required.

4. Redundant safety grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

6. Each PWR should have: (A.) Safety grade calculational devices and display (minimum of two meters) or (B.) a highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident) which is under development.

8. In all cases appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

9. The attachment provides a definition of information required on the subcooling meter.

BWR OWNERS' GROUP DISCUSSION:

Additional hardware to identify inadequate core cooling on BWRs has not been determined to be necessary at this time. Licensees' procedures will identify the diverse methods of determining inadequate core cooling, using existing instrumentation. The results of analysis being performed in response to 2.1.9 will be factored into procedures as required, after the analysis is complete.

Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure. Thus there is no need for a subcooling meter in the BWR.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. Analyses and operator guidelines for the detection and mitigation of inadequate core cooling are currently being developed per Requirement 2.1.9 and questions from the Bulletins and Orders Task Force. These studies include an evaluation of currently installed reactor vessel water level instrumentation, and the possible use of other instrumentation, to detect inadequate core cooling. The need for further measures, if any, will be addressed after these analyses and operator guidelines are complete. Implementation of emergency procedures and retraining will be done on a schedule consistent with those established with the Bulletins and Orders Task Force.

2. A subcooling meter, as required by Enclosure 6 of NUREG 0578 Implementation Letter of September 13, 1979 will not be provided.

LILCO'S RESPONSE:

LILCO concurs with the BWR Owners' Group position.

1. The BWR Owners' Group, of which LILCO is a member, and General Electric, have participated in the preparation of a report entitled, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", NEDO-24708, August 1979. The NEDO-24708 document, together with information subsequently supplied by the BWR Owners' Group, presents generic operator guidelines for dealing with scenarios which have the potential for leading to inadequate core cooling. The operator guidelines were developed using "state-of-the-art" analytic techniques. In addition to the generic guidelines, NEDO-24708 describes instrumentation and methods that can be used by reactor operators to detect inadequate core cooling. Safety and backup systems that can prevent or mitigate the consequences of inadequate core cooling are also addressed in NEDO-24708. Procedures specifically applicable to Shoreham, for detection and mitigation of inadequate core cooling, will be developed prior to Shoreham's startup. The operator training program for Shoreham will address the use of Shoreham procedures to detect, and deal with, inadequate core cooling.

2. The BWR, unlike the PWR, operates with both liquid and steam in the reactor pressure vessel. Since a saturation condition is always maintained, regardless of system pressure, the concept of a saturation margin meter is not applicable to BWRs.

2.1.4 Containment Isolation Provisions for PWRs and BWRs

NUREG 0578 POSITION:

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

NRC POSITION CLARIFICATION:

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Non-essential systems should be automatically isolated by containment isolation signals.

4. Resetting of containment isolation signals shall not result in the automatic loss of containment isolation.

BWR OWNERS' GROUP DISCUSSION:

There is diversity in the parameters sensed for the initiation of BWR containment isolation. Following an isolation, deliberate operator action is required to open valves in most cases.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. Diversity of parameters sensed for the initiation of containment isolation shall be provided in accordance with SRP 6.2.4.

2. A review shall be made of all systems penetrating primary containment to identify all essential systems. The basis of such classification shall be documented and supplied to the NRC.

3. All systems not identified as essential will be reviewed. If automatic isolation is not provided, justification for not isolating will be presented to the NRC.

4. Licensees will review and modify isolation control systems and administrative controls, as appropriate, such that no isolation valve will open when the isolation logic is reset. Those plants that have valves that will automatically open when the isolation logic is reset will change the isolation logic to prevent the valves from opening when reset. Administrative controls to prevent valves from reopening will be implemented by 1/1/80; logic modifications will be implemented by 1/1/81.

LILCO'S RESPONSE:

A review of all systems penetrating the containment has been performed to:

a. ensure that diverse containment isolation signals that satisfy safety-grade requirements are provided;

b. identify essential and non-essential systems;

c. ensure that non-essential systems are automatically isolated by containment isolation signals; and

d. ensure that resetting of containment isolation signals does not result in the automatic loss of containment isolation.

Penetrations not provided with automatic isolation or diversity of parameters sensed for the initiation of containment isolation signal are under evaluation. For these penetrations, either design modifications will be made to provide automatic isolation and/or diversity of initiating signal, or appropriate justification will be provided. The penetrations determined not to require automatic isolation or diversity of signal, along with the justification for this approach will be identified in a supplement to this report.

The design of the isolation system for the lines penetrating the Shoreham primary containment conforms to the intent of 10CFR50, Appendix A, General Design Criteria 54, 55, 56 and 57. A description of the isolation provision for each containment penetration is provided in FSAR Table 6.2.4-1, enclosed herein. A number of specific signals are used for isolation of various process and safety systems. A summary of the containment isolation signals is given on page eight of this table. The detail of the lines penetrating the containment are presented on FSAR Figures 6.2.4-1, 2 and 3, also enclosed in this report.

Essential and non-essential systems, as defined for containment isolation purposes, are identified in Table 2.1.4-1. Essential systems are those that may be needed within ten minutes of a LOCA, a normal reactor scram or a scram system failure. All other systems are designated as non-essential. All non-essential systems with possible release path are either isolated automatically by isolation signals, by check valves that would prevent flow out of containment, or by remote manual operated valves which are closed during normal operation.

The design review of control systems for automatic isolation valves demonstrated that the resetting of the isolation signal(s) will not result in the automatic reopening of containment isolation valves. This criterion is met in all cases except the outboard feedwater testable check valves. These valves, upon receipt of an isolation signal, receive a spring assist in the close direction. Resetting of the isolation signal will remove spring assist, but will not provide an opening force. Therefore, a design modification based on this criteria is not warranted.

TABLE [illegible].1.4-1

SYSTEMS CLASSIFICATION

PRIMARY CONTAINMENT PENETRATION NUMBER	SYSTEM	ESSENTIAL	NON-ESSENTIAL
X-1A,B,C,D	Main Steam	X	
	Main Steam Line Drain and MSIV-Leakage Control System		
X-2A	Feedwater	X	
X-2B	Feedwater	X	
X-3	Main Steam Line Drain		X
X-4	RWCU Line from RPV		X
X-5	RHR Shutdown Cooling from RPV		X
X-6A,B	RHR Injection Line to Recirculation System Return	X	
X-7A,B	RHR-Containment Spray Drywell		X
X-8A,B	RHR-Containment Spray Suppression Chamber		X
X-9A,B,C,D	RHR Pump Suction	X	
X-10A	RHR Test Line Return to Suppression Chamber,		X
	Suppression Pool Cleanup Return,		X
	RHR Steam Condensing Discharge,		X
	RHR Minimum Flow,	X	
	Core Spray Test Line, and		X
	Core Spray Minimum Flow	X	

PRIMARY CONTAINMENT PENETRATION NUMBER	SYSTEM	ESSENTIAL	NON-ESSENTIAL
X-10B	RHR Test Line Return to Suppression Chamber,		X
	RCIC Minimum Flow,	X	
	HPCI Minimum Flow,	X	
	RHR Steam Condensing Discharge,		X
	RHR Minimum Flow,	X	
	Core Spray Test Line,		X
	Core Spray Minimum Flow, and	X	
	Relief Valve Discharge from RHR Supply to RCIC Pump Suction		X
X-11	RHR - Head Spray Line to RPV		X
X-12	HPCI Turbine Steam Inlet Line	X	
X-13	HPCI Turbine Exhaust	X	
X-15	HPCI Pump Suction	X	
X-16	RCIC Turbine Steam Inlet Line	X	
X-17	RCIC Turbine Exhaust	X	
X-18	RCIC Vacuum Pump Discharge	X	
X-19	RCIC Pump Suction	X	
X-20A,B	Core Spray Pump Discharge to RPV	X	
X-21A,B	Core Spray Pump Suction	X	
X-22A,B	RBCLCW to Recirc. Pump and Motor Coolers	X	
X-23A,B	RBCLCW from Recirc. Pump and Motor Coolers	X	

PRIMARY CONTAINMENT PENETRATION NUMBER	SYSTEM	ESSENTIAL	NON-ESSENTIAL
X-24A to H	RBCLCW to Drywell Unit Coolers		X
X-25A,B	RBCLCW from Drywell Unit Coolers		X
X-26	Purge Air to Drywell		X
X-27	Purge Air from Drywell		X
X-28	Purge Air to Suppression Chamber		X
X-29	Purge Air from Suppression Chamber		X
X-30	Sample Coolant from RPV		X
X-31	Equipment Drains from Drywell		X
X-32	Floor Drains from Drywell		X
X-36	Standby Liquid Coolant to RPV	X	
X-37A	Nitrogen/Air Purge for TIP		X
X-37B,C,D	TIP Drive Guide Tubes		X
X-38	TIP Drive Guide Tubes		X
X-39A,B	Instrument Air to Suppression Chamber		X
X-41	HPCI Vacuum Breaker	X	
X-42	RCIC Vacuum Breaker	X	
X-43	RHR Relief Valve Discharge Vacuum Breaker,		X
	RHR Heat Exchanger Vent,		X
	RHR Heat Exchanger (RV), and		X
	HPCI Steam Supply to RHR Heat Exchanger (RV)		X

PRIMARY CONTAINMENT PENETRATION NUMBER	SYSTEM	ESSENTIAL	NON-ESSENTIAL
X-44	Containment Atmospheric Control from Suppression Chamber, and		X
	Drywell Floor Seal Pressurization	X	
X-45	Containment Atmospheric Control from Suppression Chamber, and		X
	Drywell Floor Seal Pressurization	X	
X-46	Containment Atmospheric Control from Drywell		X
X-47	Containment Atmospheric Control from Drywell		X
	CRD Insert and Withdraw Lines	X	
XS-5	HPCI Steam Supply to RHR Heat Exchanger,		X
	RHR Heat Exchanger Vent, and		X
	RHR Heat Exchanger (RV)	X	
XS-6	Suppression Pool Cleanup/Pump Down		X
XS-7	Containment Atmospheric Control to Suppression Chamber		X
XS-8	Containment Atmospheric Control to Suppression Chamber		X
XS-11	Drywell Service Air		X
XS-12	Containment Drywell Radiation Monitoring Subsystem		X
XS-13	Containment Drywell Radiation Monitoring Subsystem		X

PRIMARY CONTAINMENT PENETRATION NUMBER	SYSTEM	ESSENTIAL	NON-ESSENTIAL
XS-20	Containment Atmospheric Control to Drywell		X
XS-21	Containment Atmospheric Control to Drywell		X
XS-22	Containment Vent to RBNVS		X
B-7	Instrument Air to Drywell		X[1]
D-5	Instrument Air to Drywell		X[1]
F-10	Recirc. Pump Seal Injection		X
F-11	Recirc. Pump Seal Injection		

(1) Even though instrumentation air is non-essential, its supply to the drywell is desirable. Hence it is not isolated. Check valves provide isolation in the event of loss of instrument air.

TABLE

PROCESS PIPELINES PENETRA

(Numbers in parentheses are keyed to notes on p

Primary Containment Penetrations	Lines Isolated (22)	GDC	Number of Lines	Valves Per Line	Nominal Pipe Size (In.)	Valve Location Relative to Primary Containment
X-1A,B,C,D	Main Steam	55	4	1	24	Inside
				1	24	Outside
	Main Steam Line Drain and MSIV-	55	4	1	2	Outside
	Leakage Control System	55	4	1	2 1/2	Outside
X-2A	Feedwater	55	1	1	18	Inside
				1	18	Outside
X-2B	Feedwater	55	1	1	18	Inside
				1	18	Outside
X-3	Main Steam Line Drain	55	1	1	3	Inside
				1	3	Outside
X-4	RWCU Line from RPV	55	1	1	6	Inside
				1	6	Outside
X-5	RHR Shutdown Cooling from RPV	55	1	1	20	Inside
				1	20	Outside
X-6A,B	RHR Injection Line to Recirculation System Return	55	2	1	24	Inside
				1	2	Inside
				1	24	Outside
X-7A,B	RHR - Containment Spray Drywell	56	2	1	10	Outside
				1	10	Outside
X-8A,B	RHR - Containment Spray Suppression **Chamber**	56	2	1	6	Outside
				1	16	Outside
				1	16	Outside
X-9A,B,C,D	RHR Pump Suction	56	4	1	20	Outside
X-10A	RHR Test Line Return to Suppression Chamber,	56	1	1	16	Outside
	Suppression Pool Cleanup Return,		1	2	6	Outside
	RHR Steam Condensing Discharge,		1	1	4	Outside
	RHR Minimum Flow,		1	1	4	Outside
	Core Spray Test Line, and		1	1	10	Outside
	Core Spray Minimum Flow		1	1	3	Outside

6.2.4-1

TING PRIMARY CONTAINMENT

ages 7 and 8; signal codes are listed on page 8.)

Valve and/or Operator Type (6,22)	Power to Open (5,6)	Power to Close (5,6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8,9)	Remarks
AO Globe	Air/AC/DC	Air/Spring	B,C,D,E,P,R,T,RM	3-5	Open	(1)
AO Globe	Air/AC/DC	Air/Spring	B,C,D,E,P,R,T,RM	3-5	Open	(1)
MO Globe	AC	AC	B,C,D,E,P,R,T,RM	4	Open	
MO Globe	AC	AC	RM	6	Closed	(19)
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
VTC	Flow	Reverse Flow/AC/ Spring	Reverse Flow/F,G,RM	N/A	Open	(11)
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
VTC	Flow	Reverse Flow/AC/ Spring	Reverse Flow/F,G,RM	N/A	Open	(11)
MO Gate	AC	AC	B,C,D,E,P,R,T,RM	16	Open	
MO Gate	DC	DC	B,C,D,E,P,R,T,RM	16	Open	
MO Gate	AC	AC	A,J,RM	30	Open	
MO Gate	DC	DC	A,J,W,Y,RM	30	Open	
MO Gate	AC	AC	A,F,U,RM	23	Closed	
MO Gate	DC	DC	A,F,U,RM	23	Closed	
VTC	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(3)
MO Gate	AC	AC	A,RM	19	Closed	
MO Gate	AC	AC	RM	24	Closed	(12)
MO Gate	AC	AC	F,G,RM	51	Closed	(2)
MO Angle	AC	AC	F,G,RM	10	Closed	(2)
MO Globe	AC	AC	F,G,RM	71	Closed	(2)
MO Gate	AC	AC	F,G,RM	71	Closed	(2)
MO Globe	AC	AC	F,G,RM	79	Closed	(2)
MO Gate	AC	AC	RM	106	Open	(13)
MO Globe	AC	AC	F,G,RM	79	Closed	(2)
MO Gate	AC	AC	A,F,RM	31	Closed	
MO Gate	AC	AC	F,G,RM	20	Closed	
MO Gate	AC	AC	RM	20	Open	(16)
MO Globe	AC	AC	F,G,RM	67	Closed	
MO Gate	AC	AC	RM	16	Open	(16)

TABLE

Primary Containment Penetrations	Lines Isolated (22)	GDC	Number of Lines	Valves Per Line	Nominal Pipe Size (In.)	Valve L Relati Pri Conta
X-10B	RHR Test Line Return to Suppression Chamber,	56	1	1	16	Out
	RCIC Minimum Flow,		1	1	2	Out
	HPCI Minimum Flow,		1	1	4	Out
	RHR Steam Condensing Discharge,		1	1	4	Out
	RHR Minimum Flow,		1	1	4	Out
	Core Spray Test Line,		1	1	10	Out
	Core Spray Minimum Flow, and		1	1	3	Out
	Relief Valve Discharge from RHR Supply to RCIC Pump Suction		1	1	2	Out
X-11	RHR - Head Spray Line to RPV	55	1	1	4	Ins
				1	4	Out
X-12	HPCI Turbine Steam Inlet Line	55	1	1	10	Ins
				1	1	Ins
				1	10	Outs
				1	1	Outs
X-13	HPCI Turbine Exhaust	56	1	1	18	Outs
				2	18	Outs
X-14	Spare	-	-	-	-	-
X-15	HPCI Pump Suction	56	1	1	16	Outs
X-16	RCIC Turbine Steam Inlet Line	55	1	1	3	Insi
				1	1	Insi
				1	3	Outs
				1	1	Outs
X-17	RCIC Turbine Exhaust	56	1	1	8	Outs
				2	8	Outs
X-18	RCIC Vacuum Pump Discharge	56	1	1	2	Outs
				1	2	Outs
X-19	RCIC Pump Suction	56	1	1	6	Outs

ocation ive to mary nment	Valve and/or Operator Type (6,22)	Power to Open (5,6)	Power to Close (5,6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8,9)	Remarks
side	MO Globe	AC	AC	F,G,RM	79	Closed	(2)
side	MO Globe	DC	DC	RM	18	Closed	(16)
side	MO Globe	DC	DC	RM	29	Closed	(16)
side	MO Gate	AC	AC	F,C,RM	20	Closed	
side	MO Gate	AC	AC	RM	20	Open	(16)
side	MO Globe	AC	AC	F,G,RM	67	Closed	
side	MO Gate	AC	AC	RM	16	Open	(16)
side	Relief Valve	High Differential Pressure	Spring	N/A	N/A	Closed	
de	MO Gate	AC	AC	A,F,U,RM	20	Closed	
side	MO Globe	DC	DC	A,F,U,RM	13	Closed	
de	MO Gate	AC	AC	K,RM	11	Open	(7)
de	MO Globe	AC	AC	K,RM	12	Open	(7)
side	MO Gate	DC	DC	K,RM	43	Closed	(7)
side	MO Globe	DC	DC	K,RM	12	Open	(7)
side	MO Gate	DC	DC	RM	102	Open	
side	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
	-	-	-	-	-	-	(15)
side	MO Gate	DC	DC	K,RM	71	Closed	
de	MO Gate	AC	AC	K,RM	16	Open	(7)
de	MO Globe	AC	AC	K,RM	12	Open	(7)
side	MO Gate	DC	DC	K,RM	16	Closed	(7)
side	MO Globe	DC	DC	K,RM	12	Open	(7)
side	MO Gate	DC	DC	RM	38	Open	
side	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(13)
ide	MO Stop Check	Flow/DC	Rev. Flow/DC	Rev. Flow/RM	13	Closed	(13,21)
ide	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
ide	MO Gate	DC	DC	RM	31	Closed	

TABLE 6.

Primary Containment Penetrations	Lines Isolated (22)	GDC	Number of Lines	Valves Per Line	Nominal Pipe Size (In.)	Valve Location Relative to Primary Containment
X-20A,B	Core Spray Pump Discharge to RPV	55	2	1 1 1	10 2 10	Inside Inside Outside
X-21A,B	Core Spray Pump Suction	56	2	1	14	Outside
X-22A,B	RBCLCW to Recirc. Pump and Motor Coolers	57	2	1	4	Outside
X-23A,B	RBCLCW from Recirc. Pump and Motor Coolers	57	2	1	4	Outside
X-24A to H	RBCLCW to Drywell Unit Coolers	56	8	1 1	3 3	Inside Outside
X-25A,B	RBCLCW from Drywell Unit Coolers	56	2	1 1	4 4	Inside Outside
X-26	Purge Air to Drywell	56	1	1 1	18 18	Inside Outside
X-27	Purge Air from Drywell	[illegible]	1	1 1	18 18	Inside Outside
X-28	Purge Air to Suppression Chamber	56	1	2	18	Outside
X-29	Purge Air from Suppression Chamber	56	1	2	18	Outside
X-30	Sample Coolant from RPV	55	1	1 1	3/4 3/4	Inside Outside
X-31	Equipment Drains from Drywell	56	1	2	3	Outside
X-32	Floor Drains from Drywell	56	1	2	4	Outside
X-33	Spare	-	-	-	-	-
X-34	Spare	-	-	-	-	-
X-35	Spare	-	-	-	-	-
X-36	Standby Liquid Coolant to RPV	55	1	1 1 2	1 1/2 1 1/2 1 1/2	Inside Outside Outside

Valve and/or Operator Type (6,22)	Power to Open (5,6)	Power to Close (5,6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8,9)	Remarks
VTC	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(3)
MO Globe	AC	AC	RM	18	Closed	
MO Gate	AC	AC	RM	43	Closed	(18)
MO Gate	AC	AC	RM	76	Open	
MO Gate	AC	AC	RM	23	Open	
MO Gate	AC	AC	RM	23	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
MO Gate	AC	AC	F,G,Z,RM	16	Open	
MO Gate	AC	AC	F,G,Z,RM	20	Open	
MO Gate	AC	AC	F,G,Z,RM	20	Open	
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Globe	AC/Air	Spring	B,C,RM	15	Open	
AO Globe	AC/Air	Spring	B,C,RM	15	Open	
MO Gate	AC	AC	A,F,RM	16	Open	
MO Gate	AC	AC	A,F,RM	16	Open	
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
Explosive	AC	N/A	RM	Instantaneous	Closed	

Primary Containment Penetrations	Lines Isolated (22)	GDC	Number of Lines	Valves Per Line	Nominal Pipe Size (In.)	Valve Location Relative to Primary Containment
X-37A	Nitrogen/Air Purge for TIP	57	1	1	3/8	Outside
X-37B,C,D	TIP Drive Guide Tubes	57	3	1 1	3/8 3/8	Outside Outside
X-38	TIP Drive Guide Tubes	57	1	1 1	3/8 3/8	Outside Outside
X-39A,B	Instrument Air to Suppression Chamber	56	2	1 1	1 1	Outside Outside
X-40	Spare	-	-	-	-	-
X-41	HPCI Vacuum Breaker	56	1	1 2	2 18	Outside Outside
X-42	RCIC Vacuum Breaker	56	1	1 2	1 1/2 8	Outside Outside
X-43	RHR Relief Valve Discharge Vacuum Breaker,	56	1	N/A	N/A	N/A
	RHR Heat Exchanger Vent,		2	2	1	Outside
	RHR Heat Exchanger, and		2	1	1	Outside
	HPCI Steam Supply to RHR Heat Exchanger		2	1	6	Outside
X-44	Containment Atmospheric Control from Suppression Chamber, and	56	1	1 1	6 4	Outside Outside
	Drywell Floor Seal Pressurization	57	1	1	1/2	Outside
X-45	Containment Atmospheric Control from Suppression Chamber, and	56	1	1 1	6 4	Outside Outside
	Drywell Floor Seal Pressurization	57	1	1	1/2	Outside
X-46	Containment Atmospheric Control from Drywell	56	1	1 1	6 6	Inside Outside
X-47	Containment Atmospheric Control from Drywell	56	1	1 1	6 6	Inside Outside
	CRD Insert and Withdraw Lines	55	137 137	1 1	3/4 1	Outside Outside

Valve and/or Operator Type (6,22)	Power to Open (5,6)	Power to Close (5,6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8,9)	Remarks
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
Ball	AC	Spring	-	0.5	Closed	(14)
Explosive Shear	N/A	DC	RM	Instantaneous	Open	(14)
Ball	AC	Spring	-	0.5	Closed	(14)
Explosive Shear	N/A	DC	RM	Instantaneous	Open	(14)
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
MO Globe	AC	AC	F,G,RM	5	Closed	
-	-	-	-	-	-	(15)
MO Globe	DC	DC	F and X, RM	13	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
MO Globe	DC	DC	F and X, RM	16	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
N/A	N/A	N/A	N/A	N/A	N/A	
MO Globe	AC	AC	RM	10	Closed	
Relief Valve	High Pressure	Spring	N/A	N/A	Closed	
Relief Valve	High Pressure	Spring	N/A	N/A	Closed	
MO Gate	AC	AC	RM	31	Closed	
MO Gate	AC	AC	RM	20	Closed	
MO Globe	AC	AC	RM	6	Open	
MO Gate	AC	AC	RM	31	Closed	
MO Gate	AC	AC	RM	20	Closed	
MO Globe	AC	AC	RM	6	Open	
MO Gate	AC	AC	RM	31	Closed	
MO Gate	AC	AC	RM	16	Closed	
MO Gate	AC	AC	RM	31	Closed	
MO Gate	AC	AC	RM	16	Closed	
Globe	Manual	Manual	N/A	N/A	Open	(20)
Globe	Manual	Manual	N/A	N/A	Open	(20)

TABLE 6

Primary Containment Penetrations	Lines Isolated (22)	GDC	Number of Lines	Valves Per Line	Nominal Pipe Size (In.)	Valve Locatio Relative to Primary Containment
XS-1	Spare	-	-	-	-	-
XS-2	Spare	-	-	-	-	-
XS-3	Spare	-	-	-	-	-
XS-4	Spare	-	-	-	-	-
XS-5	HPCI Steam Supply to RHR Heat Exchanger,	56	2	1	6	Outside
	RHR Heat Exchanger Vent, and		2	2	1	Outside
	RHR Heat Exchanger		2	1	1	Outside
XS-6	Suppression Pool Cleanup/Pump Down	56	1	2	10	Outside
XS-7	Containment Atmospheric Control to Suppression Chamber	56	1	2	6	Outside
XS-8	Containment Atmospheric Control to Suppression Chamber	56	1	2	6	Outside
XS-9	Spare	-	-	-	-	-
XS-10	Spare	-	-	-	-	-
XS-11	Drywell Service Air	56	1	1	1 1/2	Inside
				1	1 1/2	Outside
XS-12	Containment Drywell Radiation Monitoring Subsystem	56	1	1	1 1/2	Inside
				1	1 1/2	Outside
XS-13	Containment Drywell Radiation Monitoring Subsystem	56	1	1	1	Inside
				1	1	Outside
XS-14	Spare	-	-	-	-	-
XS-15	Spare	-	-	-	-	-
XS-16	Spare	-	-	-	-	-
XS-17	Spare	-	-	-	-	-

n	Valve and/or Operator Type (6,22)	Power to Open (5,6)	Power to Close (5,6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8,9)	Remarks
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)
	Relief Valve	High Pressure	Spring	N/A	N/A	Closed	
	MO Globe	AC	AC	RM	10	Closed	
	Relief Valve	High Pressure	Spring	N/A	N/A	Closed	
	MO Gate	AC	AC	A,F,RM	51	Closed	
	MO Gate	AC	AC	RM	32	Closed	
	MO Gate	AC	AC	RM	32	Closed	
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)
	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(4)
	Gate	Manual	Manual	N/A	N/A	Locked Closed	(4)
	MO Globe	AC	AC	F,G,RM	14	Open	
	MO Globe	AC	AC	F,G,RM	14	Open	
	MO Globe	AC	AC	F,G,RM	14	Open	
	MO Globe	AC	AC	F,G,RM	14	Open	
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)
	-	-	-	-	-	-	(15)

Primary Containment Penetrations	Lines Isolated (22)	GDC	Number of Lines	Valves Per Line	Nominal Pipe Size (In.)	Valve Locatio Relative to Primary Containment
XS-18	Spare	-	-	-	-	-
XS-19	Spare	-	-	-	-	-
XS-20	Containment Atmospheric Control to Drywell	56	1	1 1	6 6	Inside Outside
XS-21	Containment Atmospheric Control to Drywell	56	1	1 1	6 6	Inside Outside
XS-22	Containment Vent to RBNVS	56	1	1 1	6 6	Inside Outside
XS-23	Spare (Reserved for RPV Internal Inspection)	-	-	-	-	-
XS-24	Spare	-	-	-	-	-
XS-25	Spare	-	-	-	-	-
XS-26	Spare	-	-	-	-	-
XS-27	Spare	-	-	-	-	-
XS-28	Spare	-	-	-	-	-
XS-29	Spare	-	-	-	-	-
XS-30	Spare	-	-	-	-	-
B-7	Instrument Air to Drywell	56	1	1 1	1 1/2 1 1/2	Inside Outside
D-5	Instrument Air to Drywell	56	1	1 1	1 1/2 1 1/2	Inside Outside
F-10	Recirc. Pump Seal Injection	55	1	1 1	3/4 3/4	Inside Outside
F-11	Recirc. Pump Seal Injection	55	1	1 1	3/4 3/4	Inside Outside

Valve and/or Operator Type (6,22)	Power to Open (5,6)	Power to Close (5,6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8,9)	Remarks
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
MO Gate	AC	AC	RM	32	Closed	
MO Gate	AC	AC	RM	32	Closed	
MO Gate	AC	AC	RM	32	Closed	
MO Gate	AC	AC	RM	32	Closed	
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
-	-	-	-	-	-	(15)
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
MO Globe	AC	AC	RM	7.5	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
MO Globe	AC	AC	RM	7.5	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	

These notes are keyed by number to correspond to numbers in parentheses.

1. Main steam isolation valves require that both solenoid pilots be deenergized to close valves. Accumulator air pressure plus spring set together close valves when both pilots are de-energized. Voltage failure at only one pilot will not cause valve closure. The valves are set to fully close in less than 5 sec.

2. Containment spray to drywell and suppression chamber and RHR test line return to suppression chamber isolation valves will have the capability to be manually reopened after automatic closure. This setup will permit containment spray, for high drywell pressure conditions, and/or suppression water cooling. When automatic signals are not present, these valves may be opened for test or operating convenience.

3. Testable check valves are designed for remote opening with zero differential pressure across the valve seat. The valves will close on reverse flow even though the test switches may be positioned for open. The valves will open when pump discharge pressure exceeds reactor pressure even though the test switch may be positioned for close.

4. This line is only needed during maintenance. Service air supply is disconnected during plant operation by administrative control.

5. AC motor operated valves required for isolation functions are powered from the emergency AC power buses. DC operated isolation valves are powered from the station batteries.

6. All motor operate isolation valves will remain in the last position upon fai ure of valve power. All air-operated isolation valves will close upon air failure.

7. Signal B opens, signal K overrides to close.

8. Power operated valve can be opened or closed by remote manual switch for operating convenience during any mode of reactor operation except when automatic signal is present (see Note 2).

9. Normal status position of valve (open or closed) is the position during normal power operation of the reactor.

10. The specified closure rates are as required for containment isolation only.

11. Special air testable check valves designed for remote testing durin mechanical operability of the val will cause only a partial movemen with only a minor effect on flow. the actuator spring force will ei when the feedwater system is avai providing a positive closure diff when the feedwater flow is not av

12. This valve will open when both a and an accident signal are presen

13. The motor operator of this valve operating conditions.

14. Traversing In-Core Probe (TIP) Sy

When the TIP system cable is inse tube opens automatically so that A maximum of four valves may be o the calibration, and any one guid per year.

If closure of the line is require by a containment isolation signal retracted and the ball valve clos of cable withdrawal. To ensure i cable fails to withdraw or a ball shear valve is installed in each manual signal, this explosive val seal the guide tube.

15. All unused penetrations (designat welded.

16. Valve will close on system high f

17. Isolation signals A or F will ini ventilation system which in turn valves.

18. This valve will open when both a the valve and an accident signal

NOTES

with a positive closing feature are
g normal operation to assure
ve disc. The remote testing feature
t of the disc, into the flow stream,
Upon receipt of an isolation signal,
ther cause a slight reduction in flow
lable or cause the valve to close,
erential pressure on the seated disc,
ailable.

low reactor pressure vessel pressure
t.

is keylocked open during normal

stem

rted, the ball valve of the selected
the probe and cable may advance.
pened at any one time to conduct
e tube is used, at most, a few hours

d during calibration, as indicated
, the cable is automatically
es automatically after completion
solation capability, if a TIP
valve fails to close, an explosive,
line. Upon receipt of a remote
ve will shear the TIP cable and

ed "Spare") are capped and seal

low.

iate the reactor building standby
isolates the purge air isolation

ow differential pressure across
re present.

19. Pressure sensors, sensing steam line pressure are used for interlock control to prevent inadvertent valve opening at high steam line pressures (above 35 psig).

20. Control Rod Drive (CRD) Insert and Withdraw Lines

Criteria 55 concerns those lines of the reactor coolant pressure boundary penetrating the primary reactor containment. The CRD insert and withdraw lines are not part of the reactor coolant pressure boundary. The classification of the insert and withdraw lines is Quality Group B, and therefore, designed in accordance with ASME Section III, Class 2. The basis to which the CRD lines are designed is commensurate with the safety importance of isolating these lines. Since these lines are vital to the scram function, their operability is of utmost concern.

In the design of this system, it has been accepted practice to omit automatic valves for isolation purposes as this introduces a possible failure mechanism. As a means of providing positive actuation, manual shutoff valves are used. In the event of a break on these lines, the manual valves may be closed to ensure isolation. In addition, a ball check valve located in the insert line inside the CRD is designed to automatically seal this line in the event of a break.

21. This MO stop check valve is normally in a closed position due to its check valve feature, but its MO is in the open position. The MO provides a backup to close the valve to provide additional high leak tight integrity.

NOTES (CONT'D)

22. Abbreviations used in table:

AO	- Air operated
MO	- Motor operated
VTC	- Pneumatic testable check valve
RHR	- Residual Heat Removal System
RPV	- Reactor Pressure Vessel
RCIC	- Reactor Core Isolation Cooling System
RWCU	- Reactor Water Cleanup
HPCI	- High Pressure Coolant Injection
GDC	- General Design Criterion
RBCLCW	- Reactor Building Closed Loop Cooling Water
TIP	- Transversing Incore Probe
CRD	- Control Rod Drive
MSIV	- Main Steam Isolation Valve

TABLE 6.

ISOLATI

Signal	
A*	Reactor vessel low water
B*	Reactor vessel low water at this level, and recir
C*	High radiation - main st
D*	Line break - main steam l
E*	Line break - main steam l
F*	High drywell pressure
G	Reactor vessel low water level)
J*	Line break in reactor wat
K*	Line break in steam line diaphragm pressure)
L	Reactor building standby
M	High radiation signal dow
P*	Low main steam line press
R	Low condenser vacuum
T	High temperature in Turbi
U	High reactor vessel press
W*	High temperature at outle
X	Low steam pressure
Y	Standby liquid control sy
Z	Low level in RBCLCW head
RM*	Remote manual switch from

* These are the isolation f other functions are give

Description

level 3 - (A scram will occur at this level)

level 2 - (The reactor core isolation cooling system and the high pressure coolant injection system will be initiated
ulation pumps are tripped)

am line

ine (high steam flow)

ine (steam line tunnel high temperature)

level 1 - (The core spray systems and the low pressure core injection mode of RHR systems will be initiated at this

er cleanup system - high space temperature, high differential flow, high differential temperature

to/from turbine (high steam line space temperature, high steam flow, low steam line pressure or high turbine exhaust

ventilation system initiation

nstream of primary containment purge filter train

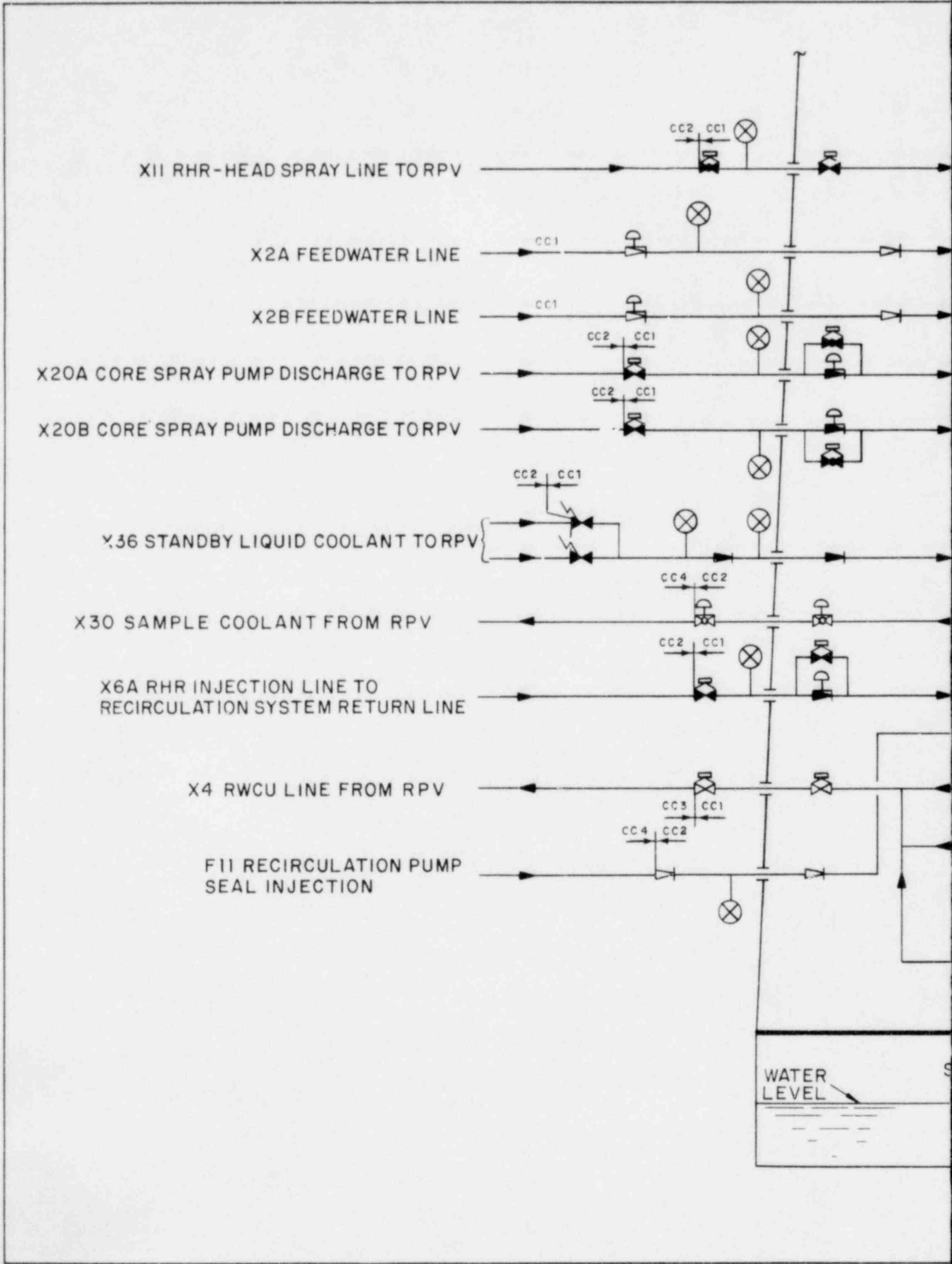
ure at inlet to turbine (RUN mode only)

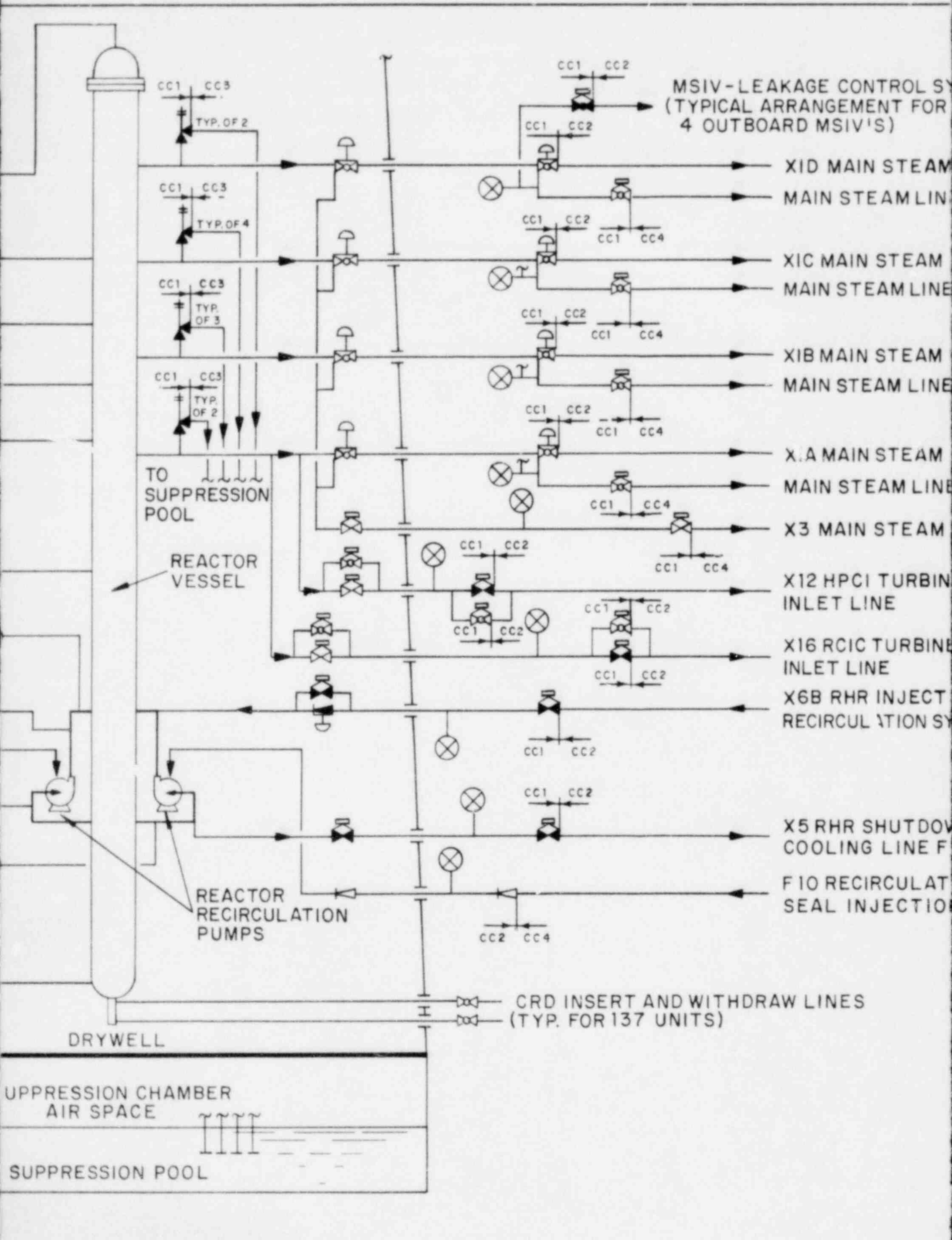
ne Building
ure

t of cleanup system nonregenerative heat exchanger

stem actuated
tank
main control room

unctions of the primary containment and reactor vessel isolation control system;
n for information only.


X11 RHR-HEAD SPRAY LINE TO RPV
CC2
CC1
X2A FEEDWATER LINE
CC1
X2B FEEDWATER LINE
CC1
X20A CORE SPRAY PUMP DISCHARGE TO RPV
CC2
CC1
X20B CORE SPRAY PUMP DISCHARGE TO RPV
CC2
CC1
X36 STANDBY LIQUID COOLANT TO RPV
CC2
CC1
X30 SAMPLE COOLANT FROM RPV
CC4
CC2
X6A RHR INJECTION LINE TO
RECIRCULATION SYSTEM RETURN LINE
CC2
CC1
X4 RWCU LINE FROM RPV
CC3
CC1
F11 RECIRCULATION PUMP
SEAL INJECTION
CC4
CC2
WATER
LEVEL



MSIV-LEAKAGE CONTROL SY
(TYPICAL ARRANGEMENT FOR
4 OUTBOARD MSIV'S)
CC1 CC2
CC1 CC3
TYP. OF 2
TYP. OF 4
TYP. OF 3
X1D MAIN STEAM
MAIN STEAM LIN
X1C MAIN STEAM
MAIN STEAM LINE
X1B MAIN STEAM
MAIN STEAM LINE
X1A MAIN STEAM
MAIN STEAM LINE
CC1 CC4
TO SUPPRESSION POOL
X3 MAIN STEAM
REACTOR VESSEL
X12 HPCI TURBIN
INLET LINE
X16 RCIC TURBINE
INLET LINE
X6B RHR INJECT
RECIRCULATION SY
X5 RHR SHUTDOW
COOLING LINE F
F10 RECIRCULAT
SEAL INJECTION
CC2 CC4
REACTOR RECIRCULATION PUMPS
CRD INSERT AND WITHDRAW LINES
(TYP. FOR 137 UNITS)
DRYWELL
UPPRESSION CHAMBER
AIR SPACE
SUPPRESSION POOL

YSTEM

E DRAIN

DRAIN

DRAIN

E DRAIN

LINE DRAIN

E STEAM

E STEAM

ION LINE TO
YSTEM RETURN LINE

WN
ROM RPV

ION PUMP
N

LEGEND

- GLOBE VALVE (CLOSED)
- GLOBE VALVE (OPEN)
- GATE VALVE (OPEN)
- GATE VALVE (CLOSED)
- CHECK VALVE (OPEN)
- CHECK VALVE (CLOSED)
- MOTOR OPERATOR
- AIR OPERATOR
- SAFETY / RELIEF VALVE
- PUMP
- EXPLOSIVE VALVE
- LEAK TEST CONNECTION (L.T.C.)

HPCI - HIGH PRESSURE COOLANT INJECTION

RCIC - REACTOR CORE ISOLATION COOLING

RHR - RESIDUAL HEAT REMOVAL

RPV - REACTOR PRESSURE VESSEL

CRD - CONTROL ROD DRIVE

RWCU - REACTOR WATER CLEANUP

MSIV - MAIN STEAM ISOLATION VALVE

NOTES

1. MAIN STEAM PIPING UP TO ISOLATION VALVES WAS PURCHASED TO B31.1 AND ANALYZED TO ASME III (CODE CLASS 1) (CC1).
2. ALL L.T.C. VALVES (EITHER GATE OR GLOBE) AND LINE SIZES ARE 3/4 INCH, ASME III CC2 AND HAVE AT LEAST ONE OF THE TWO VALVES IN SERIES LOCKED CLOSED.

FIG. 6.2.4-1

CRITERION 55 CONTAINMENT ISOLATION VALVES

SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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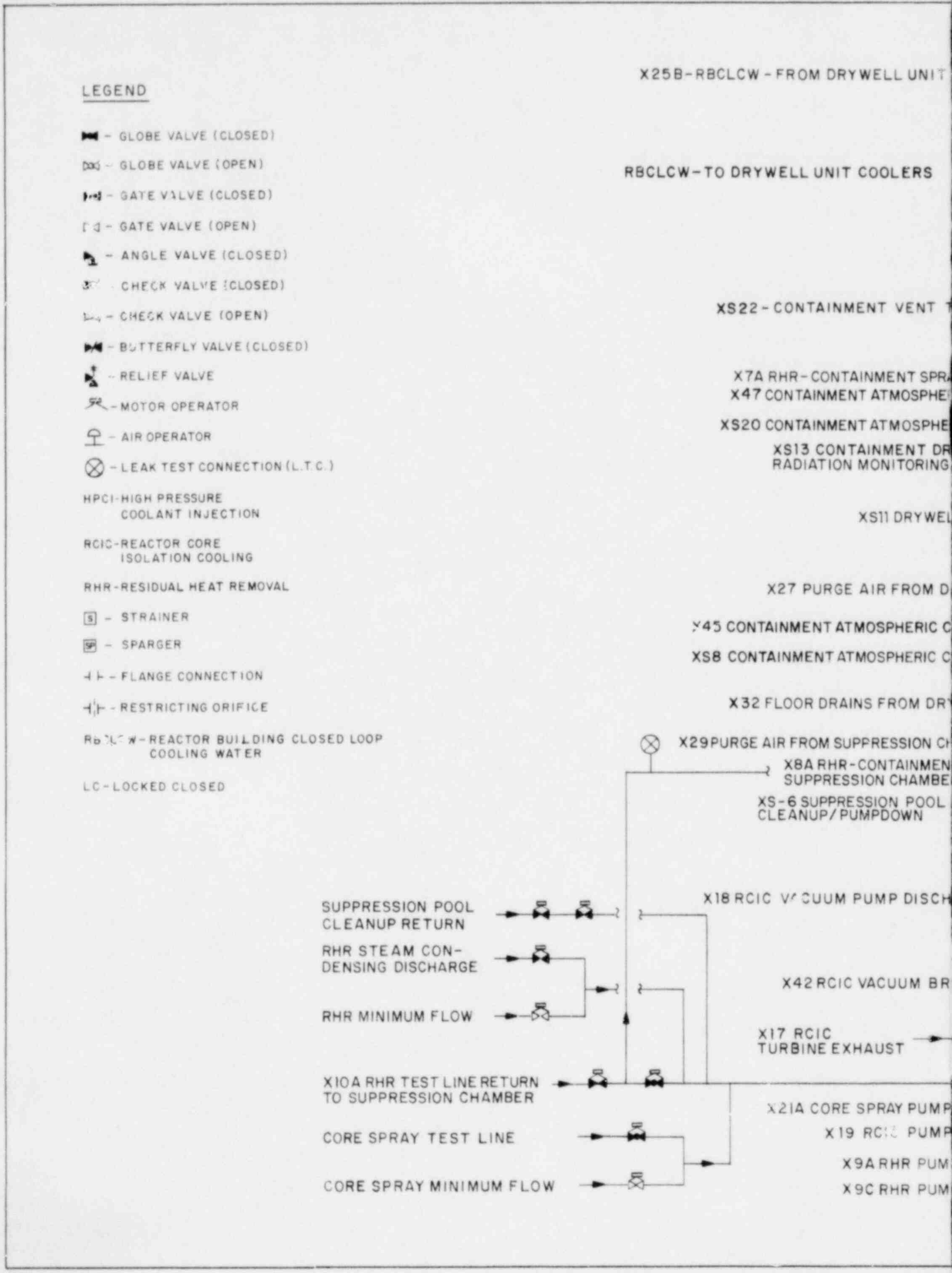
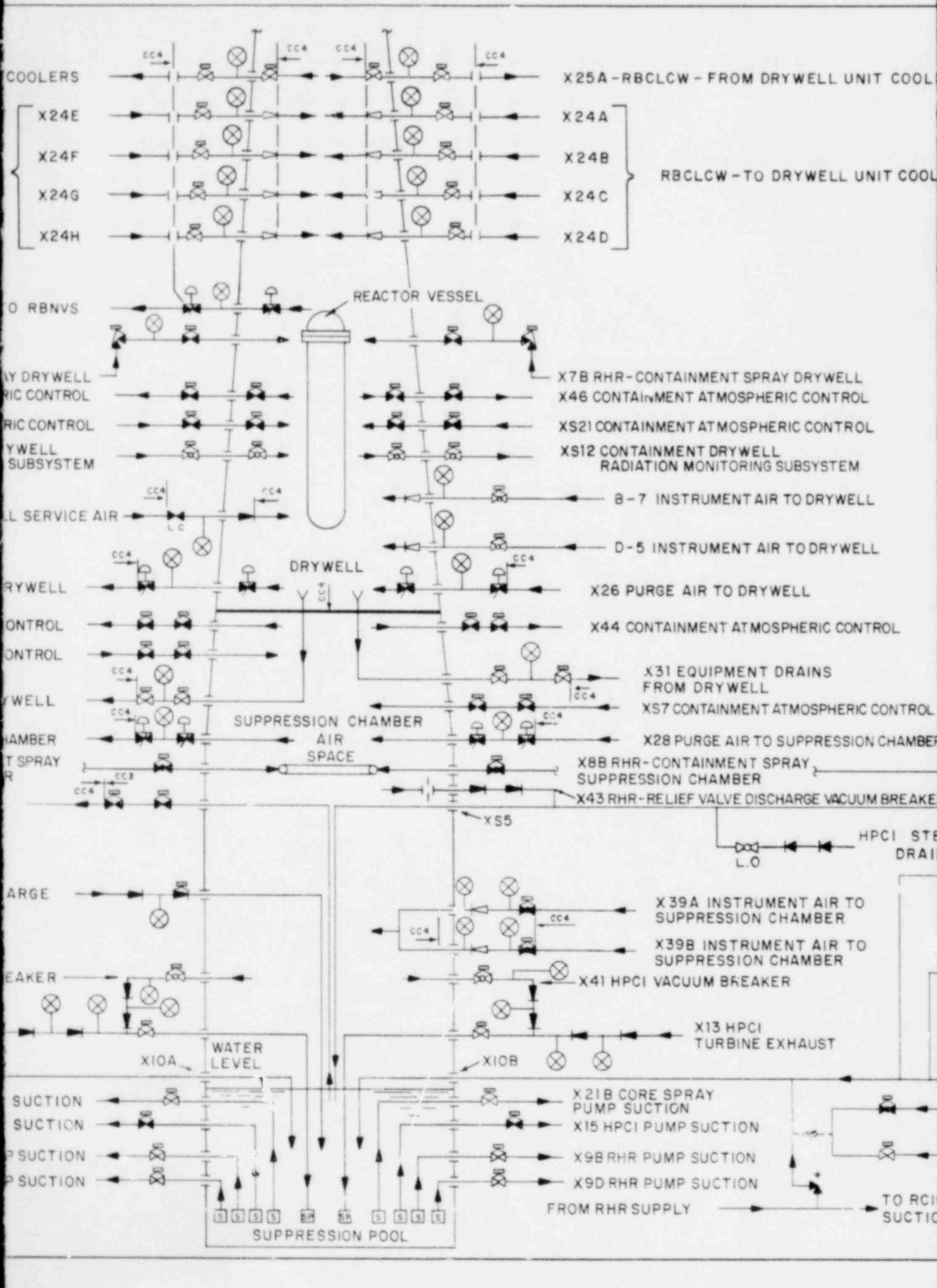

LEGEND
- GLOBE VALVE (CLOSED)
- GLOBE VALVE (OPEN)
- GATE VALVE (CLOSED)
- GATE VALVE (OPEN)
- ANGLE VALVE (CLOSED)
CHECK VALVE (CLOSED)
- CHECK VALVE (OPEN)
- BUTTERFLY VALVE (CLOSED)
- RELIEF VALVE
- MOTOR OPERATOR
- AIR OPERATOR
- LEAK TEST CONNECTION (L.T.C.)
HPCI-HIGH PRESSURE COOLANT INJECTION
RCIC-REACTOR CORE ISOLATION COOLING
RHR-RESIDUAL HEAT REMOVAL
- STRAINER
- SPARGER
- FLANGE CONNECTION
- RESTRICTING ORIFICE
W-REACTOR BUILDING CLOSED LOOP COOLING WATER
LC-LOCKED CLOSED
X25B-RBCLCW-FROM DRYWELL UNIT
RBCLCW-TO DRYWELL UNIT COOLERS
XS22-CONTAINMENT VENT
X7A RHR-CONTAINMENT SPR
X47 CONTAINMENT ATMOSPHE
XS20 CONTAINMENT ATMOSPHE
XS13 CONTAINMENT DR
RADIATION MONITORING
XS11 DRYWEL
X27 PURGE AIR FROM D
X45 CONTAINMENT ATMOSPHERIC C
XS8 CONTAINMENT ATMOSPHERIC C
X32 FLOOR DRAINS FROM DR
X29 PURGE AIR FROM SUPPRESSION C
X8A RHR-CONTAINMEN
SUPPRESSION CHAMBE
XS-6 SUPPRESSION POOL
CLEANUP/PUMPDOWN
SUPPRESSION POOL CLEANUP RETURN
RHR STEAM CONDENSING DISCHARGE
RHR MINIMUM FLOW
X18 RCIC VACUUM PUMP DISCH
X42 RCIC VACUUM BR
X17 RCIC TURBINE EXHAUST
X10A RHR TEST LINE RETURN TO SUPPRESSION CHAMBER
CORE SPRAY TEST LINE
CORE SPRAY MINIMUM FLOW
X21A CORE SPRAY PUMP
X19 RCIC PUMP
X9A RHR PUM
X9C RHR PUM



COOLERS
X25A - RBCLCW - FROM DRYWELL UNIT COOL
X24E
X24F
X24G
X24H
X24A
X24B
X24C
X24D
RBCLCW - TO DRYWELL UNIT COOL
CC4
O RBNVS
REACTOR VESSEL
AY DRYWELL
RIC CONTROL
RIC CONTROL
RYWELL
SUBSYSTEM
X7B RHR - CONTAINMENT SPRAY DRYWELL
X46 CONTAINMENT ATMOSPHERIC CONTROL
XS21 CONTAINMENT ATMOSPHERIC CONTROL
XS12 CONTAINMENT DRYWELL
RADIATION MONITORING SUBSYSTEM
B-7 INSTRUMENT AIR TO DRYWELL
L SERVICE AIR
L C
D-5 INSTRUMENT AIR TO DRYWELL
DRYWELL
RYWELL
X26 PURGE AIR TO DRYWELL
ONTROL
X44 CONTAINMENT ATMOSPHERIC CONTROL
ONTROL
X31 EQUIPMENT DRAINS
FROM DRYWELL
YWELL
XS7 CONTAINMENT ATMOSPHERIC CONTROL
SUPPRESSION CHAMBER
AIR
SPACE
HAMBER
X28 PURGE AIR TO SUPPRESSION CHAMBER
T SPRAY
X8B RHR - CONTAINMENT SPRAY
SUPPRESSION CHAMBER
CC2
X43 RHR - RELIEF VALVE DISCHARGE VACUUM BREAKE
XS5
HPCI STE
DRAI
L.O
ARGE
X39A INSTRUMENT AIR TO
SUPPRESSION CHAMBER
X39B INSTRUMENT AIR TO
SUPPRESSION CHAMBER
EAKER
X41 HPCI VACUUM BREAKER
X13 HPCI
TURBINE EXHAUST
X10A
WATER
LEVEL
X10B
SUCTION
X21B CORE SPRAY
PUMP SUCTION
SUCTION
X15 HPCI PUMP SUCTION
P SUCTION
X9B RHR PUMP SUCTION
P SUCTION
X9D RHR PUMP SUCTION
FROM RHR SUPPLY
TO RCIC
SUCTIO
SUPPRESSION POOL

NOTES

1. THESE ARE ALL ASME III CODE CLASS 2 (CC2) SYSTEMS UNLESS OTHERWISE NOTED.
2. ALL L.T.C. VALVES (EITHER GATE OR GLOBE) AND LINE SIZES ARE 3/4 INCH, ASME III CC2 AND HAVE AT LEAST ONE LOCKED CLOSED VALVE.

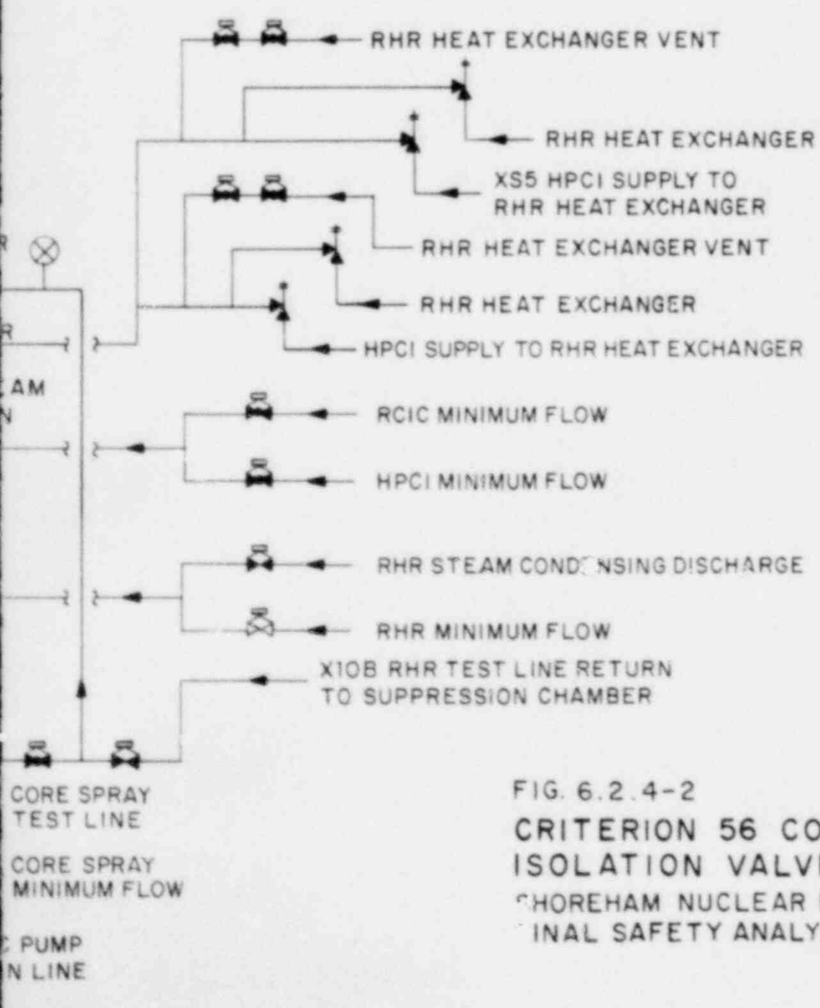
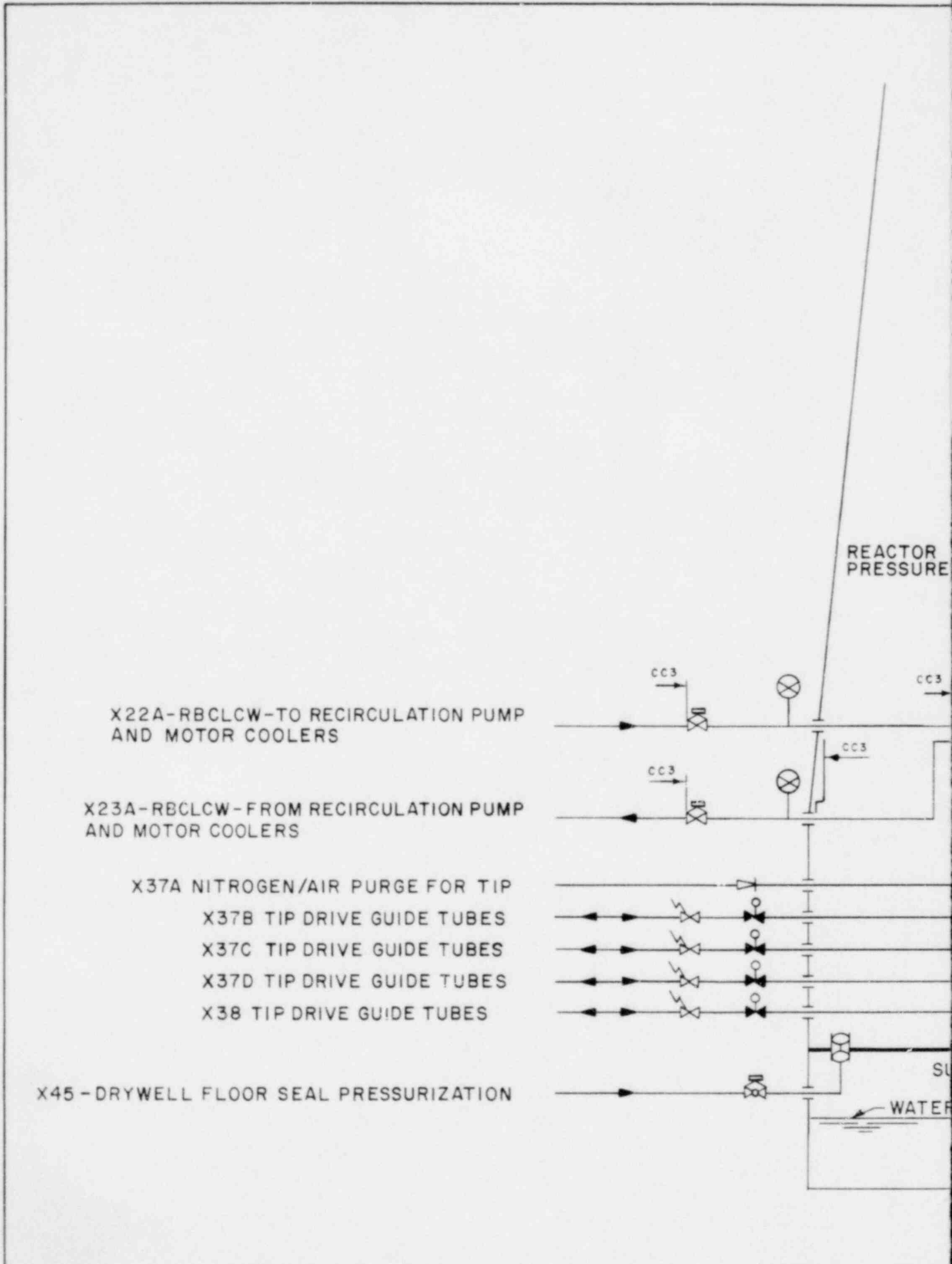
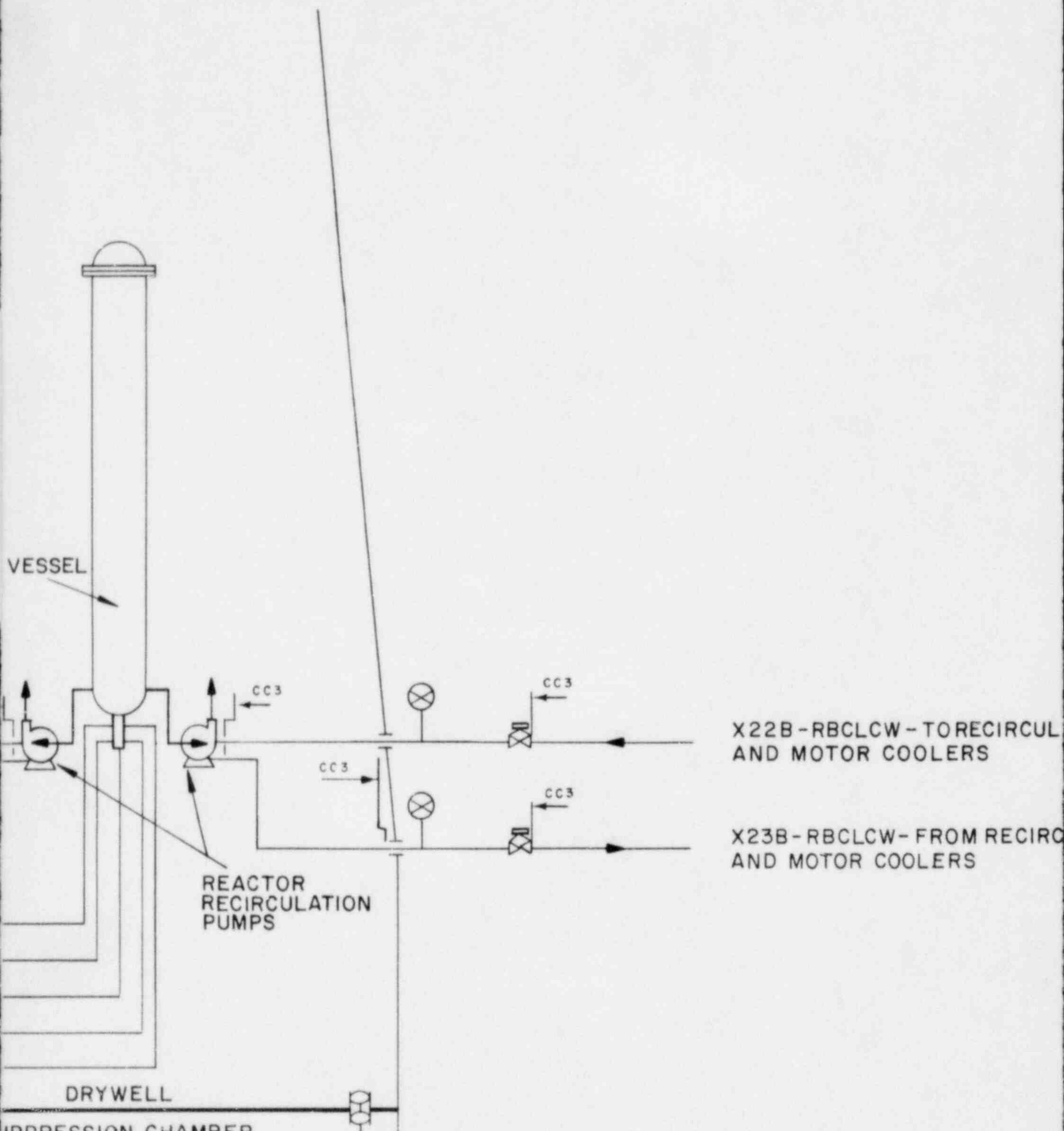


FIG. 6.2.4-2

CRITERION 56 CONTAINMENT ISOLATION VALVES

SHOREHAM NUCLEAR POWER STATION – UNIT 1
FINAL SAFETY ANALYSIS REPORT


REACTOR
PRESSURE
CC3
X22A-RBCLCW-TO RECIRCULATION PUMP AND MOTOR COOLERS
X23A-RBCLCW-FROM RECIRCULATION PUMP AND MOTOR COOLERS
X37A NITROGEN/AIR PURGE FOR TIP
X37B TIP DRIVE GUIDE TUBES
X37C TIP DRIVE GUIDE TUBES
X37D TIP DRIVE GUIDE TUBES
X38 TIP DRIVE GUIDE TUBES
X45-DRYWELL FLOOR SEAL PRESSURIZATION
WATER



VESSEL
CC3
CC3
CC3
CC3
X22B-RBCLCW-TORECIRCUL
AND MOTOR COOLERS
X23B-RBCLCW-FROM RECIRC
AND MOTOR COOLERS
REACTOR
RECIRCULATION
PUMPS
DRYWELL
UPPRESSION CHAMBER
AIR SPACE
R LEVEL
X44-DRYWELL FLOOR SEAL
SUPPRESSION POOL

LEGEND

- GLOBE VALVE (OPEN)
- GLOBE VALVE (CLOSED)
- GATE VALVE (OPEN)
- GATE VALVE (CLOSED)
- MOTOR OPERATOR
- PUMP

TIP-TRAVERSING INCORE PROBE

- BALL VALVE (CLOSED)
- EXPLOSIVE VALVE
- LEAK TEST CONNECTION (L.T.C.)

RBCLCW - REACTOR BUILDING CLOSED LOOP COOLING WATER

- DRYWELL FLOOR SEAL

NOTES:

1. ALL PENETRATIONS, PIPING AND ISOLATION VALVES ARE ASME III CODE CLASS 2 (CC2)
2. ALL L.T.C. VALVES (EITHER GATE OR GLOBE) AND LINE SIZES ARE 3/4 INCH, ASME III CC2 AND HAVE AT LEAST ONE LOCKED CLOSED VALVE.

ATION PUMP

ULATION PUMP

PRESSURIZATION

FIG. 6.2.4-3

CRITERION 57 CONTAINMENT ISOLATION VALVES

SHOREHAM NUCLEAR POWER STATION-UNIT 1

FINAL SAFETY ANALYSIS REPORT

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2.1.5.a Dedicated Penetrations for External Recombiners or Post Accident Purge Systems

NUREG 0578 POSITION:

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

NRC CLARIFICATION:

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

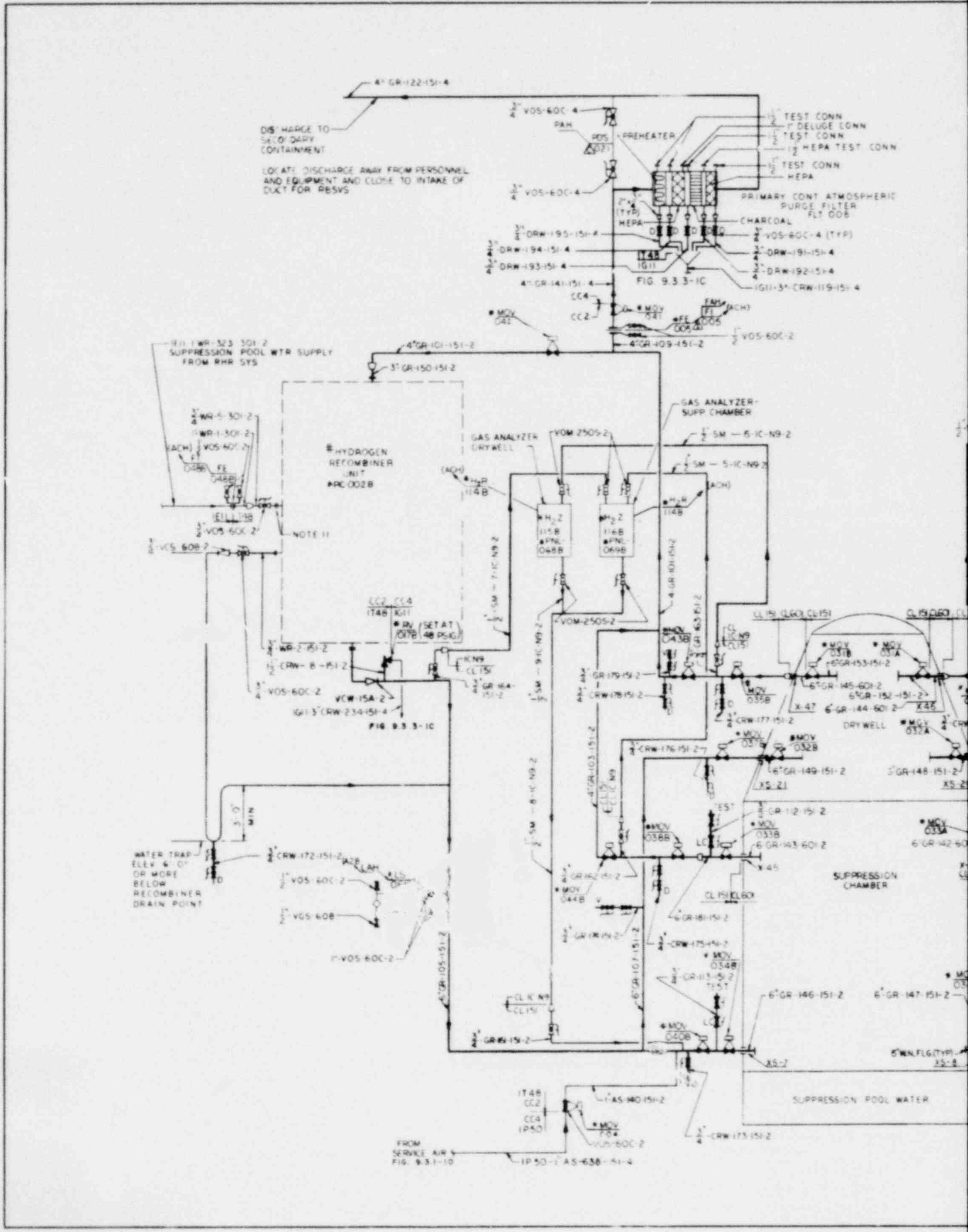
None

LILCO'S RESPONSE:

The Shoreham design presently incorporates redundant external recombiners for the control of combustible gases inside the primary containment.

Two 100 percent capacity hydrogen recombiners are currently installed. The system is safety-related, Seismic Category I and designed in accordance with ASME III, Code Class 2. The recombiners are located in the reactor building outside the primary containment. Four dedicated penetrations are provided for each recombiner as shown in the enclosed FSAR Figure 6.2.5-1.[1] Two isolation valves are provided for each primary containment penetration in accordance with the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50. The combustible gas control system is considered non-essential. However, automatic isolation is not provided, since all isolation valves in this system are closed during normal operation.

[1] This Figure is provided for information only. The hydrogen analyzers have been provided with dedicated penetrations and are no longer connected to the hydrogen recombiner penetrations as shown in this Figure. Thus, the hydrogen recombiners are provided with dedicated penetrations. This Figure will be updated in a future amendment to the FSAR.



4" GR-122-151-4
DISCHARGE TO SECONDARY CONTAINMENT
LOCATE DISCHARGE AWAY FROM PERSONNEL AND EQUIPMENT AND CLOSE TO INTAKE OF DUCT FOR RBSVS
PREHEATER
TEST CONN
1" DELUGE CONN
TEST CONN
HEPA TEST CONN
TEST CONN
HEPA
PRIMARY CONT ATMOSPHERIC PURGE FILTER FLT 008
CHARCOAL
HEPA
FIG. 9.3.3-1C
SUPPRESSION POOL WTR SUPPLY FROM RHR SYS
HYDROGEN RECOMBINER UNIT
GAS ANALYZER DRYWELL
GAS ANALYZER SUPP CHAMBER
NOTE 11
SET AT 48 PSIG
FIG 9.3.3-1C
WATER TRAP ELEV 6'-0" OR MORE BELOW RECOMBINER DRAIN POINT
DRYWELL
SUPPRESSION CHAMBER
SUPPRESSION POOL WATER
FROM SERVICE AIR FIG. 9.3.1-10

NOTES:

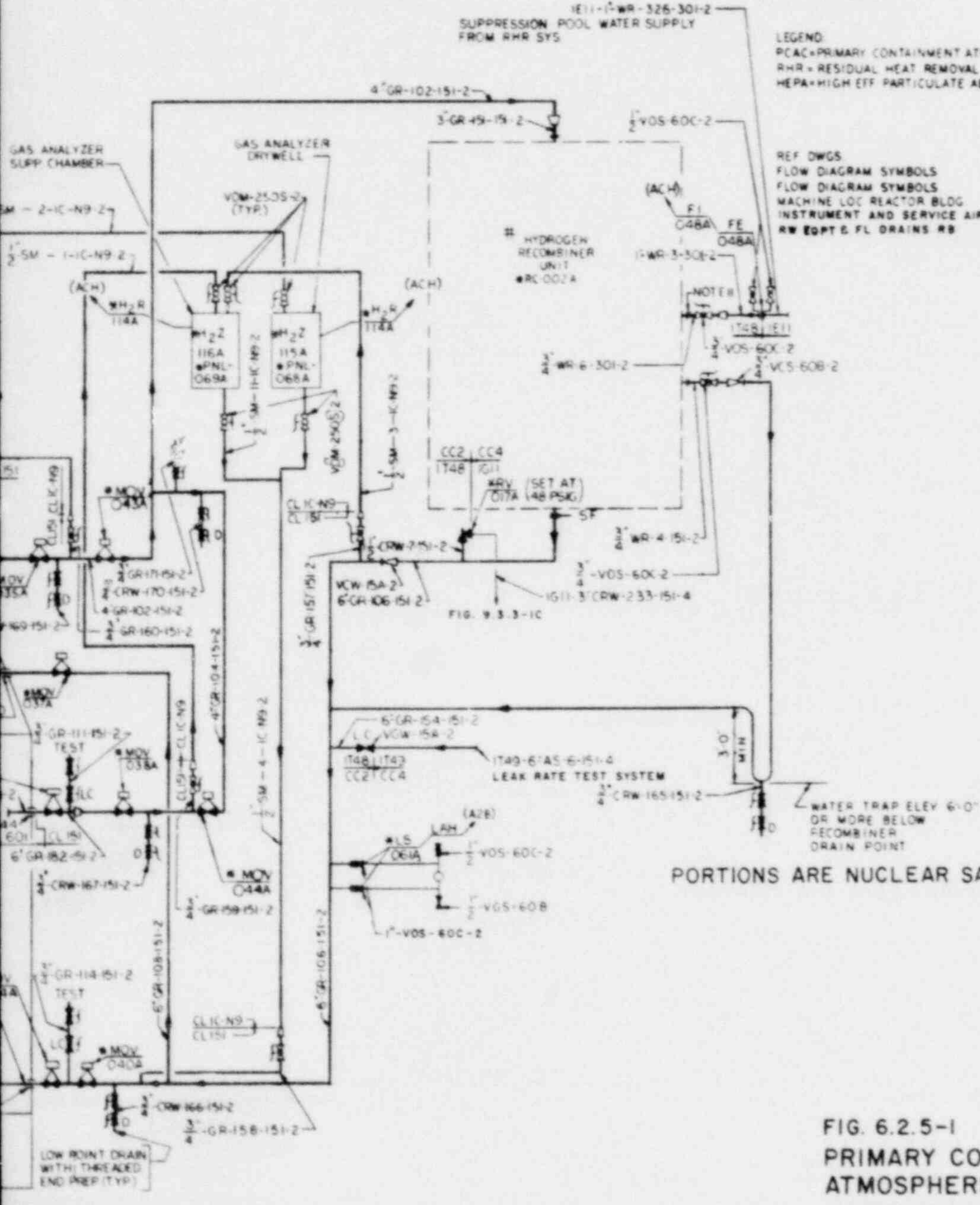
1. ALL LINE DESIGNATION NOS. & EQUIPMENT MARK NOS. TO BE PREFIXED WITH UNIT & SYSTEM NO. 1T48 UNLESS OTHERWISE NOTED; AS FOLLOWS, 1T48-6-N-143-151-2.
2. ALL INSTRUMENT NOS. ARE PREFIXED BY THE UNIT & SYSTEM NO 1T48, UNLESS OTHERWISE NOTED, AS FOLLOWS: 1T48*MOV 031A
3. ALL CONNS FOR PRESS INSTRUMENTS, SAMPLING AND TESTING TO BE 3/4" VOS-60C-2 EXCEPT AS NOTED.
4. ALL MOTOR OPERATED VALVES (MOV) TO BE VGW-15A-2, EXCEPT AS NOTED.
5. * INDICATES "SAFETY RELATED EQUIPMENT & INSTRUMENTATION"
6.
7. * INDICATES "FURNISHED BY EQUIPMENT MANUFACTURER"
8. ALL PRIMARY CONTAINMENT PENETRATIONS ARE 6 INCHES
9. ALL VENT AND DRAIN VALVES TO BE 3/4"-VOS-60C-2 UNLESS OTHERWISE NOTED.
10. ALL PIPING TO BE CODE CLASS 2 UNLESS OTHERWISE NOTED
11. INTERFACE ON HYDROGEN RECOMBINER UNIT UPGRADED TO CLASS 301.

LEGEND:

PCAC=PRIMARY CONTAINMENT ATMOSPHERIC CONTROL SYSTEM
RHR = RESIDUAL HEAT REMOVAL
HEPA=HIGH EFF PARTICULATE ABSOLUTE

REF DWGS

FLOW DIAGRAM SYMBOLS	FIG. 1.7-1A
FLOW DIAGRAM SYMBOLS	FIG. 1.7-1B
MACHINE LOC REACTOR BLDG	FIG. 3.8.1-3
INSTRUMENT AND SERVICE AIR SYSTEM	FIG. 9.3.1-1D
RW EQPT & FL DRAINS RB	FIG. 9.3.3-1C



PORTIONS ARE NUCLEAR SAFETY RELATED

FIG. 6.2.5-1
PRIMARY CONTAINMENT
ATMOSPHERIC CONTROL SYSTEM
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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2.1.5.b Inerting BWR Containments

NUREG 0578 POSITION:

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and II BWRs.

NRC CLARIFICATION:

None

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

None

LILCO POSITION:

In accordance with NRC letter from D. B. Vassallo to all pending operating license applicants, dated September 27, 1979, the proposed rule making on inerting has been delayed and no action is required at this time.

2.1.5.c Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant

NUREG 0578 POSITION:

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

NRC CLARIFICATION:

1. This requirement applies only to those plants that included Hydrogen Recombiners as a design basis for licensing.

2. The shielding and associated personnel exposure limitations associated with recombiner use should be evaluated as part of licensee response to requirement 2.1.6.b, "Design review for Plant Shielding."

3. Each licensee should review and upgrade, as necessary, those criteria and procedures dealing with recombiners use. Action taken on this requirement should be submitted by January 1, 1980.

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

None

LILCO'S RESPONSE:

The Shoreham Design employs permanently installed hydrogen recombiners. These recombiners are 100 percent redundant, safety grade and have been designed to deal with quantities of hydrogen that may be generated during and after a LOCA as predicted by ECCS analyses discussed in FSAR Section 6.3.3. Following an accident, each hydrogen recombiner is controlled from the main control room, and no access to the equipment nor local manipulation by plant personnel is required for its operation. In addition, the hydrogen recombiners are located in an area where personnel access for other purposes is not required during accident conditions. Thus, operation of this equipment will not contribute to the personnel exposures. Refer to response to Requirement 2.1.5.a. No further action is required.

2.1.6.a Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWRs and BWRs

NUREG 0578 POSITION:

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

NRC CLARIFICATION:

Licensees shall, by January 1, 1980, provide a summary description of their program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. Examples of such systems are given on page A-26 of NUREG-0578. Other examples include the Reactor Core Isolation Cooling and Reactor Water Cleanup (Letdown function) Systems for BWRs. Include a list of systems which are excluded from this program. Testing of gaseous systems should include helium leak detection or equivalent testing methods. Consider in your program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to you regarding North Anna and Related Incidents dated October 17, 1979.

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION:

Practical leakage reduction measures will be investigated for systems which may contain radioactive fluids post-LOCA. Such systems as the reactor core isolation cooling system, high-pressure coolant injection system, core spray system, residual heat removal system, and waste disposal system will be examined.

This examination will include a study of valve stem packing leakoffs, rotating seals on equipment, gasketed connections or joints, drains piped to open connections, and reactor drainage system.

Those components in the above systems from which leakage may be measured will be identified and measured leakage from these components will be reported to NRC. A periodic leak inspection program will be implemented on these components.

LILCO's RESPONSE:

A surveillance testing program, in accordance with 10CFR50 Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors", and the plant Technical Specifications, will be implemented at Shoreham. This testing program includes performance of Type A tests to measure the overall integrated primary containment leakage rates; Type B tests to detect and measure local leakage from certain containment penetrations and components; and Type C tests, to measure containment isolation valve leakage rates. These tests will be performed during preoperational testing and periodically at test intervals required by 10CFR50 Appendix J.

Periodic surveillance testing will be performed on items such as Main Steam Isolation Valves (MSIV) and Air Locks to maintain leakage within the allowable limits as specified in the plant's Technical Specifications. In addition, system hydrostatic tests, and inspections will be performed as required by ASME Section XI. During these tests appropriate corrective actions will be implemented as required.

Additional systems such as the MSIV leakage control system and the primary to secondary containment leakage detection and leakage return system have been incorporated in the plant design in order to minimize and control leakage to the maximum extent possible.

The MSIV leakage control system (MSIV-LCS) collects post LOCA leakage from the MSIV's to a maximum of 100 standard cubic feet per hour for all main steam lines. This system may be manually actuated by the operator 20 minutes after an accident. The MSIV-LCS consists of physically separated redundant blowers which route any leakage from the closed MSIV's to areas served by the Reactor Building Standby Ventilation System (RBSVS). These blowers maintain the steam lines at a pressure slightly below atmospheric thus assuring that any leakage will be directed to the RBSVS filters prior to release to the atmosphere.

The primary to secondary containment leakage detection and return system will assist in identifying and controlling post LOCA Emergency Core Cooling Systems (ECCS) leakage. Any abnormal leakage is detected by a level switch in the EL 8'-0" floor drain sump which will actuate an alarm in the main control room at high sump level. In addition, redundant safety related level detectors are provided on el. 8'-0", which will alarm in the control room when the floor water level (in the detector area) exceeds approximately 1/2 inch corresponding to approximately 2,000 gallons. The leakage return portion of the system consists of a self-priming leakage return pump with a capacity of 180 gpm which include recirculation of 50 gpm. This pump will be manually started as required and will operate to return postulated ECCS leakage to the suppression pool. The pump will be powered from the emergency power supply and will be seismically qualified. The use of the leakage return system during post LOCA conditions will allow sufficient time for operator action to identify and isolate suspected leakage paths while continuing to maintain suppression pool water inventory and preventing excessive buildup of water on el. 8'-0" of the reactor building.

An additional leakage detection program is presently under evaluation. The program will include measures to reduce and maintain leakage to as low as practical for systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident. Major features of the program currently under consideration are as follows:

1. Preparation of system list, identifying methods that may be used to test systems, the system involved, frequency of testing. A preliminary list of the systems affected is prescribed in Table 2.1.6.a-1.

2. The preparation of guidelines for evaluating a) leakage from systems, identified in 1 above, into the secondary containment through valve stems, pump seals, fittings, relief valve discharge lines, drains, vents and instrument loops and b) leakage through valve seats into interfacing systems outside of the secondary containment.

3. The implementation of a periodic visual inspection program. These inspections shall be performed on accessible portions of applicable systems during system operational testing, or by evaluation of leakage at lower pressures during operation.

4. A leak testing program shall be implemented on specific valves or connections on the systems which provide an interface to equipment or systems located external to the secondary containment and which can bypass secondary containment. This testing could be accomplished by hydrostatic leak testing of the individual valves or evaluating the total accumulated leakage during the system ASME XI hydrostatic testing.

5. Various methods to detect and control leakage from gaseous systems outside containment shall be evaluated.

6. Records shall be maintained on the tests and inspections performed on the system listed on table 2.1.6. a-1. These records shall be used to identify chronic and generic leakage problems in order to implement modifications and/or corrective mainenance measures.

Inspection and Enforcement Bulletin 79-21 is currently under evaluation. Appropriate protective measures, as identified therein or modification as applicable to Shoreham, will be implemented as necessary.

TABLE 2.1.6.a-1

PRELIMINARY LISTS OF SYSTEMS TO BE CONSIDERED FOR PERIODIC LEAKAGE INSPECTION AND CONTROL

1. Core Spray (CS)
2. High Pressure Coolant Injection (HPCI)
3. Residual Heat Removal (RHR)
4. Reactor Core Isolation Cooling (RCIC)
5. Hydrogen Recombiners (combustible Gas Control)
6. Reactor Water Cleanup
7. Coolant Sampling
8. Reactor Building Equipment Drain System
9. Reactor Building Floor Drain System
10. Reactor Building Standby Ventilation System

2.1.6.b Design Review of Plants Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations

NUREG 0578 POSITION:

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas inventory are contained in the primary plant), 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 post-accident 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 post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

NRC CLARIFICATION:

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. In order to assure that personnel can perform necessary post-accident operations in the vital areas, we are providing the following guidance 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, 1.7 and Standard Review Plan 15.6.5. with appropriate decay times based on plant design.

   a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and liquids injected by HPCI and LPCI.

b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For gas 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 gas space in the primary coolant system.

2. Dose Rate Criteria

The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates 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 basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. 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: $\leq$ 15mr/hr. 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 a regular basis, but 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 where occupancy may be needed often but not continuously.

BWR OWNERS' GROUP DISCUSSION:

BWR plants are specifically designed to mitigate major design basis events with no access outside the main control room being required. With this goal in mind, the plants were not specifically designed for any access outside the main control room. To specifically design for guaranteed access at any time in most parts of the reactor building is not feasible. However, the current designs may allow for access for short times if the entry time into the area can be selectively chosen. Design changes in shielding will be made if evaluations identify feasible modifications which should significantly enhance desirable access. The guidelines for the evaluations are given below.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

A TID 14844 radioactivity release will be assumed into the primary containment. A summation of the radioactivity levels from sump water leakage from process systems in the reactor building will be made. The next step will be to calculate the source terms for the suppression pool recirculating piping, pumps, and valves installed in the reactor building assuming that a TID 14844 release had occurred. The vital areas will be identified in the reactor building which may need to be entered during an accident recovery period. The shielding in these vital areas will be reevaluated to assess its effectiveness in such a circumstance. The occupancy time limits, taking into consideration transit time, airborne radioactivity levels, and gamma shine intensities, will then be calculated for the vital reactor building areas.

LILCO'S RESPONSE:

LILCO concurs with the BWR Owners' Group position. A radiation and shielding review is currently being performed for Shoreham to ensure adequate access is provided to vital areas following an accident. However, based upon the fact that no operator actions other than those which take place in the main control room are critical for the safe shutdown of the plant, only this area, the post-accident sampling station(s), onsite operational support center, and the technical support center, are considered to be vital for continuous post-accident personnel access. The NRC-prescribed post-accident distribution of radioactivity[1/] and General Design Criteria 19, along with the occupancy time requirements, will be applied to each of the vital areas identified above to assess the dose rate acceptability for plant personnel.

[1/] Studies are presently being conducted by the Owners' Group to verify the NRC-post accident distribution or to develop more realistic/practical assumptions.

Based on this evaluation, appropriate design changes, such as additional permanent or temporary shielding, and/or post accident procedural controls will be made to optimize access to the vital areas identified.

The major part of the Shoreham design assessment is the evaluation of the environmental qualifications of essential equipment. This evaluation will be performed using TID 14844 source terms.

2.1.7.a Automatic Initiation of the Auxiliary Feedwater System for PWRs

NUREG 0578 POSITION:

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

NRC CLARIFICATION:

Control Grade (Short-Term)

1. Provide automatic/manual initiation of ATWS

2. Testability of the initiating signals and circuits is required.

3. Initiating signals and circuits shall be powered from the emergency buses.

4. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not compromise the emergency diesel generating capacity.

5. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the ATWS from the control room.

6. Other Considerations

   a. For those designs where instrument air is needed for operation the electric power supply requirement should be capable of being manually connected to emergency power sources.

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

None

LILCO'S RESPONSE:

Shoreham is a General Electric BWR. Since this requirement is PWR specific, it does not apply to Shoreham.

2.1.7.b Auxiliary Feedwater Indication to Steam Generators for PWRs

NUREG 0578 POSITION:

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the ATWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

NRC CLARIFICATION:

A. Control Grade (Short-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy the single failure criterion.

2. Testability of the auxiliary feedwater flow indication channels shall be a feature of the design.

3. Auxiliary feedwater flow instrument channels shall be powered from the vital instrument buses.

B. Safety-Grade (Long-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy safety-grade requirements.

C. Other

1. For the Short-Term the flow indication channels should by themselves satisfy the single failure criterion for each steam generator. As a fall-back position, one auxiliary feedwater flow channel may be backed up by a steam generator level channel.

2. Each auxiliary feedwater channel should provide an indication of feed flow with an accuracy on the order of ± 10%.

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

None

LILCO'S RESPONSE:

Shoreham is a General Electric BWR. Since this requirement is PWR specific, it does not apply to Shoreham.

2.1.8.a Improved Post-Accident Sampling Capability

NUREG 0578 POSITION:

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes 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 temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

NRC CLARIFICATION:

The licensee shall have the capability to promptly obtain (in less than 1 hour) pressurized and unpressurized reactor coolant samples and a containment atmosphere (air) sample.

The licensee shall establish a plan for an onsite radiological and chemical analysis facility with the capability to provide, within 1 hour of obtaining the sample, quantification of the following:

1. certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums and non-volatile isotopes),

2. hydrogen levels in the containment atmosphere in the range 0 to 10 volume percent,

3. dissolved gases (i.e., H_2, O_2) and boron concentration of liquids.

or have in-line monitoring capabilities to perform the above analysis. Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples and a design review and procedural modifications (if necessary) shall be completed by January 1, 1980. Major plant modifications shall be completed by January 1, 1981.

During the review of the post accident sampling capability consideration should be given to the following items:

1. Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

2. The licensee shall consider provisions for purging samples lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

3. If changes or modifications to the existing sampling system are required, the seismic design and quality group classification or sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and piping downstream of the second isolation valve can be designed to quality Group D and non-seismic Category I requirements.

The licensee's radiological sample analysis capability should include provisions to:

a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Lessons Learned Item 2.1.6.b. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel exposure, should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/gm to the upper levels indicated here.

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

c. Maintain plant procedures which identify the analysis required, measurement techniques and provisions for reducing background levels.

The licensees chemical analysis capability shall consider the presence of the radiological source term indicated for the radiological analysis.

In performing the review of sampling and analysis capability, consideration shall be given to personnel occupational exposure. Procedural changes and/or plant modifications must assure that it shall be possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as reasonably achievable and not in excess of GDC 19. In assuring that these limits are met, the following criteria will be used by the staff.

1. For shielding calculations, source terms shall be as given in Lessons Learned Item 2.1.6.b.

2. Access to the sample station and the radiological and chemical analysis facilities shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples

in the analysis facilities shall be such that the radiation dose criteria are met. This may involve sufficient shielding of personnel from the samples and/or the dilution of samples for analysis. If the existing facilities do not satisfy these criteria, then additional design features, e.g., additional shielding, remote handling, etc., shall be provided. The radioactive sample lines in the sample station, the samples themselves in the analysis facilities, and other radioactive lines of the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

4. High range portable survey instruments and personnel dosimeters should be provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

The licensees shall demonstrate their capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria given in this section.

BWR OWNERS' GROUP DISCUSSION:

The BWR Owners' Group agrees with the intent of the staff's position.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

A design and operational review of existing reactor coolant and containment atmosphere sampling facilities was completed by January 1, 1980.

Modifications will be made to provide the capability to promptly obtain pressurized and unpressurized reactor coolant samples and containment atmosphere samples. Analysis capability shall be provided to identify and quantify (1) certain isotopes that are indicators of core damage (i.e., noble gases, iodines and cesiums, and non-volatile isotopes), and (2) dissolved gases (i.e., H_2 and O_2) and boron concentration of liquids. These modifications will be complete by January 1, 1981.

Until the design modifications are complete, procedures will be devised to evaluate the primary coolant system and containment environment activity depending on the accessibility of the sampling stations for particular degraded conditions.

LILCO'S RESPONSE:

A post Accident Sampling System will be provided for the Shoreham facility. This system will be housed in a new Post Accident Sample Building adjacent to the Reactor Building. A conceptual layout of this building is shown on Figure 2.1.8.a-1. This building will be accessible following an accident (Reactor Building entry not necessary) to obtain and analyze samples of the reactor coolant, containment atmosphere and suppression pool water. The outer walls and roof of the Sample Building will be shielded to reduce interior radiation levels below acceptable background levels for personnel protection. A separate intake filter Heating, Ventilating and Air Conditioning (HVAC) System will be provided for this building. During accident conditions the sampling enclosure within this building will be isolated and tied into the Reactor Building Standby Ventilation Building.

The Post Accident Sampling System, with the exception of containment isolation valves, will be operated from a control panel located in the Sample Building. This System will provide the following capability:

1. Sampling of reactor coolant and suppression pool liquids and containment atmosphere.

2. On-line gross gamma activity levels monitoring.

3. Dilution of liquid, reactor coolant gases, or containment atmosphere samples by either volumetric or feed and bleed methods, for laboratory analysis including gamma spectrum analysis.

4. Either diluted, unpressurized degassed grab samples or pressurized, undegassed, and undiluted grab samples may be obtained.

5. Dissolved gas detection range from 1% to maximum gas concentration with better than ± 10% accuracy.

Continuous monitoring of containment hydrogen and oxygen levels is provided as part of the primary containment atmospheric control system.

The system will be designed and shielded so that required samples can be taken inside the facility under worst case conditions such that the combined dose to the operator from sample fluids and from the accident environment does not exceed 3 Rem whole body or 18 3/4 Rem to the extremities. In addition, the system will be designed to keep routine operating, testing and maintenance doses As Low As Reasonably Achievable.

The sample system will be designed to non-safety grade requirements, but will be supplied with a reliable source of electric power to assure proper operation following an accident. In addition, the sample isolation valves in the reactor building will be safety grade and redundant to comply with containment isolation requirements. Containment isolation valves will be provided with automatic isolation signals and override capability from the main control room.

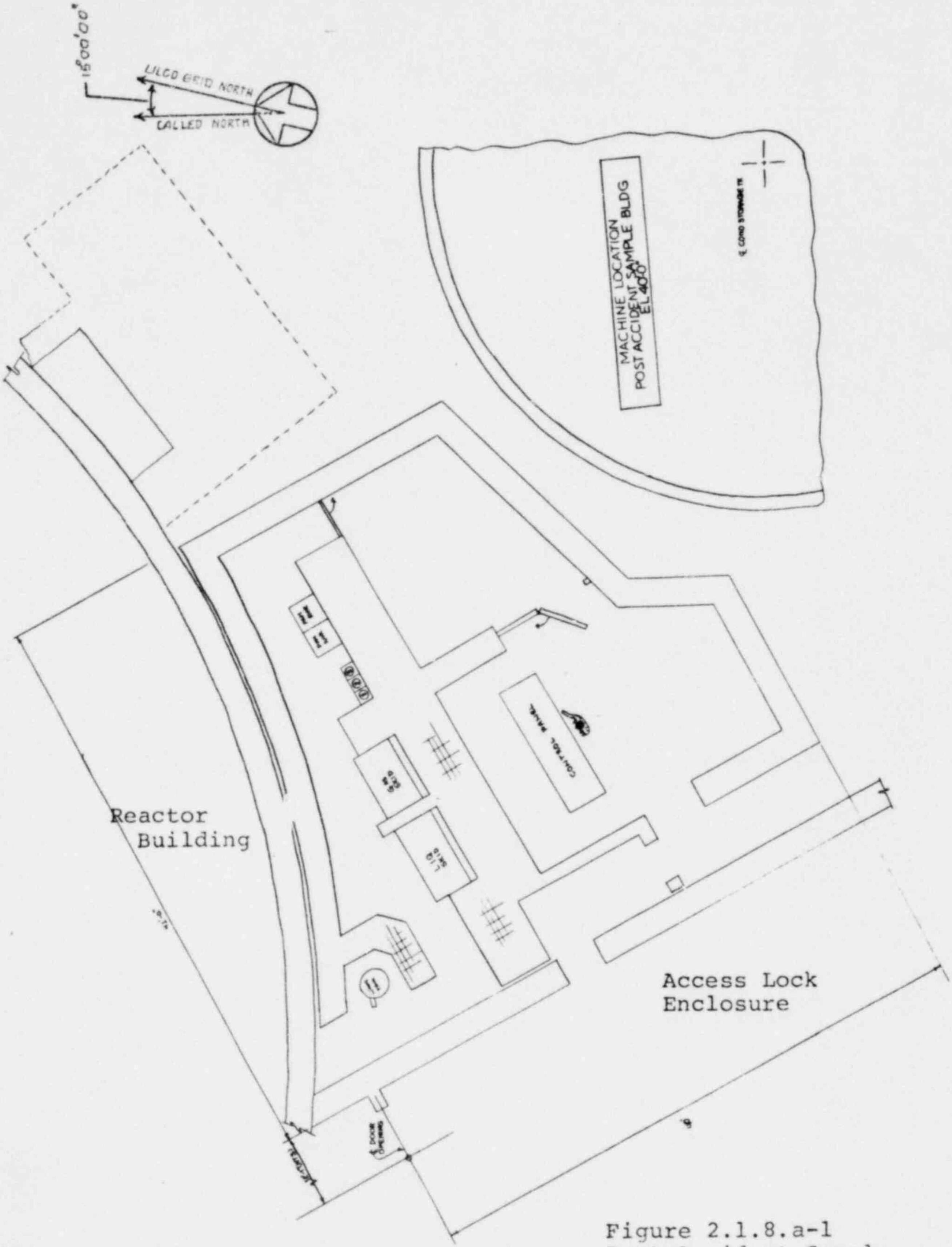


Figure 2.1.8.a-1
Post-Accident Sample
Building Conceptual
Layout

2.1.8.b Increase Range of Radiation Monitors

NUREG 0578 POSITION:

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident", which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. 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 to be necessary to cover the ranges of interest.

    a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

    b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 μCi/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.

2. Since 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 absorption on charcoal or other media, followed by onsite laboratory analysis.

3. 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 designed and qualified to function in an accident environment.

NRC CLARIFICATION (LETTER OF NOVEMBER 9, 1979)

1. Radiological Noble Gas Effluent Monitors

   a. January 1, 1980 Requirements

      Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to serve only as a provisional fix with the more detailed, exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble gases from all potential release points (e.g., auxiliary building, radwaste building, fuel handling building, reactor building, waste gas decay tank releases, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves and BWR turbine buildings) and any other areas that communicate directly with systems which may contain primary coolant or containment gases, (e.g., letdown and emergency core cooling systems and external recombiners). Measurements/analysis capabilities of the effluents at the final release point (e.g., stack) should be such that measurements of individual sources which contribute to a common release point may not be necessary. For assessing radioiodine and particulate releases, special procedures must be developed for the removal and analysis of the radioiodine/particulate sampling media (i.e., charcoal canister/filter paper). Existing, sampling locations are expected to be adequate; however, special procedures for retrieval and analysis of the sampling media under accident conditions (e.g., high air and surface contamination and direct radiation levels) are needed.

      It is intended that the monitoring capabilities called for in the interim can be accomplished with existing instrumentation or readily available instrumentation. For noble gases, modifications to existing monitoring systems, such as the use of portable high range survey instruments, set in shielded collimators so that they "see" small sections of sampling lines is an acceptable method for meeting the intent of this requirement. Conversion of the measured dose rate (mR/hr) into concentration (μCi/cc) can be performed using standard volume source calculations. A method must be developed with sufficient accuracy to quantify the iodine releases in the presence of high background radiation from noble gases collected on charcoal filters. Seimically qualified equipment and equipment meeting IEEE-279 is not required.

The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

(1) Noble Gas Effluents

a) System/Method description including:

i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, Control room read-out is preferred: however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b) Procedures for conducting all aspects of the measurement/analysis including:

i) Procedures for minimizing occupational exposures.

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration.

b. January 1, 1981 Requirements

By January 1, 1981, the licensee shall provide high range noble gas effluent monitors for each release path. The noble gas effluent monitor should meet the requirements of Table 2.1.8.b.2. The licensee shall also provide the information given in Sections 1.A.1.a.i., 1.A.1.a.ii, 1.A.1.b.ii, 1.A.1.b.iii, and 1.A.1.b.iv above for the noble gas effluent monitors.

2. Radioiodine and Particulate Effluents

   a. For January 1, 1980, the licensee should provide the following:

      (1) System/Method description including:

         a) Instrumentation to be used for analysis of the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

         b) Monitoring/sampling location.

         c) Method to be used for retrieval and handling of sampling media to minimize occupational exposure.

         d) Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

         e) If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous read-out for 7 consecutive days.

      (2) Procedures for conducting all aspects of the measurement analysis including:

         a) Minimizing occupational exposure.

         b) Calculational methods for determining release rates.

         c) Procedures for dissemination of information.

         d) Calibration frequency and technique.

   b. For January 1, 1981, the licensee should have the capability to continuously sample and provide onsite analysis of the sampling media. The licensee should also provide the information required in 2.a. above.

3. Containment Radiation Monitors

Provide by January 1, 1981, two radiation monitor systems in containment which are documented to meet the requirements of Table 2.1.8.b.3. It is possible that future regulatory requirements for emergency planning interfaces may necessitate identification of different types of radionuclides in the containment air, e.g., noble gases (indication of core damage) and non-volatiles (indication of core melt). Consequently, consideration should be given to the possible installation or future conversion of these monitors to perform this function.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group recognizes and concurs with the positions as modified in the NRC regional meetings the week of September 24, 1979.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA

1. The Owners will implement the requirements of position 2.1.8.b, items 1, 2 and 3 (provided in NRC clarification letter of November 9, 1979), consistent with commercial availability of equipment.

2. Procedures will be developed to estimate noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

LILCO'S RESPONSE:

The effluent monitors in NUREG 0578, as clarified in NRC letter from D. B. Vassallo dated November 9, 1972, Table 2.1.8.b-2, which apply to the Shoreham Nuclear Power Station are: (a) "diluted containment exhaust", (b) "other buildings", and (c) "buildings with systems containing primary coolant or gases". See Figure 2.1.8.b.1 for a simplified diagram of Shoreham's gaseous effluent layout.

The maximum anticipated primary containment leakage rate is 0.005 volumes per day into the secondary containment which has a volume of 2 X 10^6 cubic feet. The primary containment exhaust in highly diluted in the secondary containment atmosphere. This mixture will be discharged after passing through high efficiency particulate absolute filters and charcoal adsorber banks via the Reactor Building Standby Ventilation System (RBSVS) discharge pipe, at the top of the Station Vent Exhaust. The two Class 1E radiation monitors (RE-021 and RE-022) serving this system downstream of the filters and adsorbers will have a range from 1 X 10^{-6} to 1 X 10^{+4} microcuries/cc.

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 capability to provide readout of these monitors in the Technical Support Center and in the Emergency Off-site Facility is under evaluation.

The criteria in Table 2.1.8.b-2 for other buildings and buildings with systems containing primary coolant or gases are applicable to the Station Vent Exhaust monitor (RE-042). 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. During an accident 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 cubic feet/minute of outside air for dilution. This single discharge point for the combined ventilation flow from all potentially contaminated buildings will be monitored by a noble gas radiation monitor (RE-042) having a range of 1×10^{-6} to $1 \times 10^{+2}$ microcuries/cc. The monitor is backed up by RE-069 with an upper range of $1 \times 10^{+3}$. 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, RE-067, RE-068) with an upper range of $1 \times 10^{+3}$ microcuries/cc. All these monitors, except RE-042, are powered from a vital instrument bus.

The normal Station Vent Exhaust monitor (RE-042) is not powered from a vital instrument bus and, 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 RE-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 (RE-021, RE-022). 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. More directly, the Station Vent Exhaust monitor's (RE-042) radioiodine and particulate sample media can be obtained for analysis if the secondary containment is accessible.

Initial calibration will include detector response for a minimum of three decades using standard sources of two different energies and intensities. These calibration curves will be initially generated using both gaseous and solid sources, where practical. Routine calibration of these monitors will be in accordance with technical specifications provisions using solid sources related to the initial calibration.

The conversion of the instrument readings to release rates will be determined using the energy response of the detectors obtained during calibration. Accident release rates will then be 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 rates calculated.

Background radiation will not substantially affect readings on the RBSVS noble gas monitors (RE-021 and 022) during an accident, due to their location in the control building and the detector's location in a 4π lead shield. For the Station Vent Exhaust Monitor (RE-042), background radiation in the vicinity of the monitor within the secondary containment will not substantially affect the noble gas detector, due to its location in a 4π 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 they are efficient for detecting the beta radiation associated with the sample stream's noble gases brought in close contact with the detector.

The capability to readout the Station Vent Exhaust noble gas monitor (RE-042) and the individual building ventilation and Station Vent Exhaust in-line high range monitors (RE-066, RE-067, RE-068, RE-069) at the TSC and the Emergency Off-Site Facility is also under evaluation.

The radioiodine and particulate sampling media will be analyzed in the counting room at Shoreham. Charcoal cartridges will be purged with air to remove interfering noble gases. In order to facilitate analysis of the radioiodine and particulate sample media, various features have been included in the Radiochemistry laboratory and counting equipment designs to permit analysis under adverse conditions. In addition, consideration is being given to establishing a separate accident laboratory area to include counting equipment at a location on-site and establishing backup counting capability at a nearby facility with the required equipment and expertise. Further, procedures will be prepared for conducting all aspects of the measurement and analyses correctly and in a manner to minimize personnel exposure. Procedures for dissemination of information will also be prepared.

Two physically separate monitors will be installed inside the drywell having a range of 1 X 10^1 to 1 X 10^7 Roentgens/hour for photon radiation. These monitors will be each powered by a vital instrument bus, will be seismic qualified, and will be designed to withstand the temperatures, pressures, humidity and total radiation in the drywell containment through an accident. Monitor readouts will be displayed continuously and recorded on a Category I panel in the Main Control Room. Additionally, two monitors, range 1 to 1 X 10^6 R/hr, will be mounted one on the outside of the personnel hatch and the other on the outside of the equipment hatch. These monitors will provide containment radiation readings during an accident. These monitors meet the requirements of Table 2.1.8.b.3 with the exception of qualification to ANSI-N320-1979. For a listing of the radiation monitors with the ranges provided, refer to Table 2.1.8.b.4.

ATTACHMENT TO NRC CLARIFICATION

TABLE 2.1.8.b.1

INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL ACCIDENTAL RADIOACTIVITY RELEASES

Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

Examples of major elements of highly radioactive effluent release special procedures (noble gas).

- Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.

- Provide shielding to minimize background interference.

- Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.

- Predetermined calculational method to convert the radiation level to radioactive effluent release rate.

ATTACHMENT TO NRC CLARIFICATION

TABLE 2.1.8.b.2

HIGH RANGE EFFLUENT MONITOR

. NOBLE GASES ONLY

. RANGE: (Overlap with Normal Effluent Instrument Range)

- UNDILUTED CONTAINMENT EXHAUST	10^{+5}	μCi/CC
- DILUTED (©10: 1) CONTAINMENT EXHAUST	10^{+4}	μCi/CC
- MARK I BWR REACTOR BUILDING EXHAUST	10^{+4}	μCi/CC
- PWR SECONDARY CONTAINMENT EXHAUST	10^{+4}	μCi/CC
- BUILDINGS WITH SYSTEMS CONTAINING PRIMARY COOLANT OR GASES	10^{+3}	μCo/CC
- OTHER BUILDINGS (E.G., RADWASTE)	10^{+2}	μCi/CC

. NOT REDUNDANT - 1 PER NORMAL RELEASE POINT

. SEISMIC - NO

. POWER - VITAL INSTRUMENT BUS

. SPECIFICATIONS - PER R.G. 1.97 AND ANSI N320-1979

. DISPLAY*: CONTINUOUS AND RECORDING WITH READOUTS IN THE TECHNICAL SUPPORT CENTER (TSC) AND EMERGENCY OPERATIONS CENTER (EOC)

. QUALIFICATIONS - NO

*Although not a present requirement, it is likely that this information may have to be transmitted to the NRC. Consequently, consideration should be given to this possible future requirement when designing the display interfaces.

ATTACHMENT TO NRC CLARIFICATION

TABLE 2.1.8.b.3

HIGH RANGE CONTAINMENT RADIATION MONITOR

- RADIATION: TOTAL RADIATION (ALTERNATE: PHOTON ONLY)
- RANGE:
  - UP TO 10^8 RAD/HR (TOTAL RADIATION)
  - ALTERNATE: 10^7 R/HR (PHOTON RADIATION ONLY)
  - SENSITIVE DOWN TO 60 KEV PHOTONS*
- REDUNDANT: TWO PHYSICALLY SEPARATED UNITS
- SEISMIC: PER R. G. 1.97
- POWER: VITAL INSTRUMENT BUS
- SPECIFICATIONS: PER R. G. 1.97 REV. 2 and ANSI N320-1978
- DISPLAY: CONTINUOUS AND RECORDING
- CALIBRATION: LABORATORY CALIBRATION ACCEPTABLE

*Monitors must not provide misleading information to the operators assuming delayed core damage when the 80 keV photon Xe-133 is the major noble gas present.

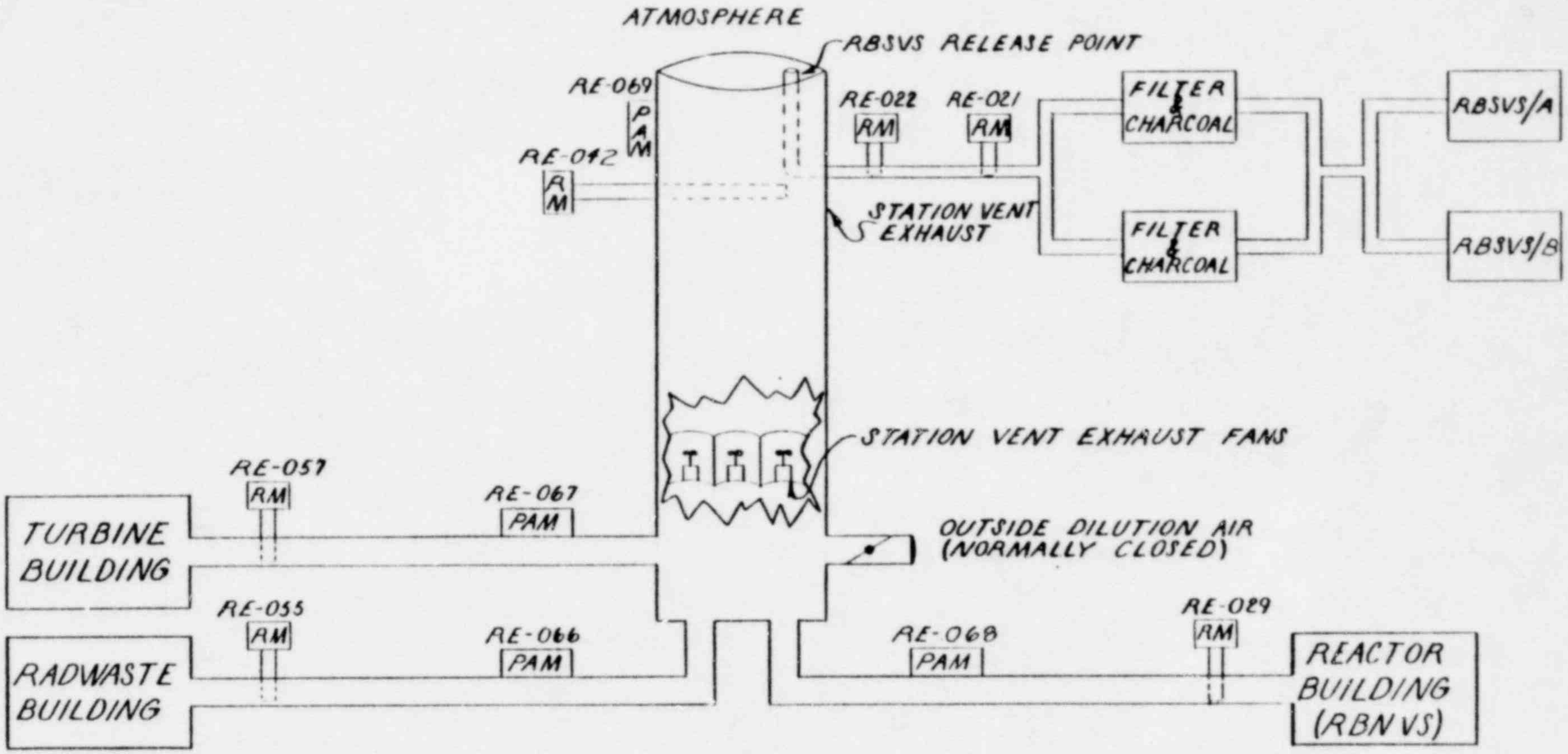
TABLE 2.1.8.b.4

RADIOACTIVITY CONCENTRATION RANGES FOR SHOREHAM GASEOUS EFFLUENT RADIATION MONITORS

GASEOUS EFFLUENT MONITOR	RANGE (microcuries/cc)
Reactor Building Standby Ventilation RE-021, RE-022*	$1x10^{-6}$ to $1x10^{+4}$
Reactor Building Normal Ventilation RE-029*	$1x10^{-6}$ to $1x10^{-1}$
Turbine Building Ventilation RE-057*	$1x10^{-6}$ to $1x10^{-1}$
Radwaste Building Ventilation RE-055*	$1x10^{-6}$ to $1x10^{-1}$
Station Vent Exhaust RE-042*	$1x10^{-6}$ to $1x10^{+2}$
Reactor Building Normal Ventilation RE-068	$1x10^{-2}$ to $1x10^{+3}$
Turbine Building Ventilation RE-067	$1x10^{-2}$ to $1x10^{+3}$
Radwaste Building Ventilation RE-066	$1x10^{-2}$ to $1x10^{+3}$
Station Vent Exhaust RE-069	$1x10^{-2}$ to $1x10^{+3}$
Drywell Monitors	$1x10^{1}$ to $1x10^{7}$ R/hr
Personnel Hatch	$1x10^{0}$ to $1x10^{6}$ R/hr
Equipment Hatch	$1x10^{0}$ to $1x10^{6}$ R/hr

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

GASEOUS EFFLUENT RADIATION MONITORS



RM= RADIATION MONITOR. THESE MONITORS DETECT NOBLE GASES, AND CONTINUOUSLY COLLECT SAMPLES FOR RADIOIODINE AND PARTICULATE RELEASE ANALSIS.

PAM= POST ACCIDENT MONITOR. THESE ARE HIGH RANGE IN-LINE RADIATION MONITORS.

THERE ARE ADDITIONAL GASEOUS STREAM RADIATION MONITORS IN THE SHOREHAM PLANT. THIS SIMPLIFIED DIAGRAM SHOWS ONLY THOSE DISCUSSED IN THE TEXT.

FIG 2.1.8.6-1

2.1.8.c Improved In-Plant Iodine Instrumentation

NUREG 0578 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.

NRC CLARIFICATION:

Use of Portable versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/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.

Iodine Filters and Measurement Techniques

A. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 365 keV of ^{131}I. A representative air sample shall be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

B. By January 1, 1981:

The licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, 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 and effluent charcoal samples under accident conditions.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group recognizes and concurs with the position.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. The Owners will implement the requirements of position 2.1.8.c.

2. Procedures will be developed to accurately determine in-plant iodine concentrations.

LILCO'S RESPONSE:

The iodine concentrations will be determined by utilizing appropriate in-plant instrumentation. Portable, semi-portable or fixed air samplers will be used to pump a known quantity of air through a charcoal filter. A gross count will then be performed on the charcoal filter cartridge to ascertain if a significant amount of radioactivity has been absorbed. If the resulting gross activity is above 9.0×10^{-9} μci/cc unidentified, a gamma spectrum analysis will be performed.

The gamma spectrum analysis will identify the 364 keV peak for I-131 as well as its confirming secondary peak. This analysis along with previous energy and efficiency calibrations of the equipment will permit quantifying the radioactivity in the sample and identifying which nuclide(s) are present. The use of respiratory protection will then be based on the concentration of each identified nuclide present and its maximum permissable concentration as indicated in 10CFR20, Appendix B.

2.1.9 Analysis of Design and Off-Normal Transient and Accidents

NUREG 0578 POSITION:

Analyses, procedures and training addressing the following are required:

1. Small break loss-of-coolant accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CRF Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).
2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying apporpriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons, together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

NRC CLARIFICATION:

Containment Pressure Indication and Containment Hydrogen Indication

1. The containment pressure indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment pressure monitor shall be installed by January 1, 1981.

Reactor Coolant System Venting

A. General

1. The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent non-condensible gases will provide additional assurance of meeting the requirements of 10CFR50.46 (LOCA criteria) and 10CFR50.44 (containment criteria for hydrogen generation). For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of non-condensible gas without the loss of core cooling or containment integrity.

2. Procedures addressing the use of the RCS vents are required by January 1, 1981. The procedures should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be based on the following criteria: (1) assurance that the plant can meet the requirements of 10CFR50.46 and 10CFR50.44 for Design Basis Accidents; and (2) a substantial increase in the plants ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since the BWR Owners Group has suggested that the present BWR designs inherent capability of venting, this question relates to the capability of existing systems. The ability of these systems to vent the RCS of non-condensible gas must be documented. Since there are important differences among BWR's, each licensee should address the specific design features of his plant.

2. In addition to reactor coolant system venting, each BWR licensee should address the ability to vent other systems such as the isolation condenser, which may be required to

maintain adequate core cooling. If the production of a large amount of non-condensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. The locations for PWR Vents are as follows:

   a. Each PWR licensee should provide the capability to vent the reactor vessel head.

   b. The reactor vessel head vent should be capable of venting non-condensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths). Additional venting capability is required for those portions of each hot leg which cannot be vented through the reactor vessel head vent. The NRC recognizes that it is impractical to vent each of the many thousands of tubes in a U-tube steam generator. However, we believe that a procedure can be developed which assures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Such a procedure is required by January 1981.

   c. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations especially during natural circulation.

2. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly large range of sizes. The criteria for sizing a vent can be developed in several ways. One approach which we consider reasonable, is to specify a volume of non-condensible gas to be vented and a venting time, i.e., a vent capable of venting a gas volume of ½ the RCS in one hour. Other criteria and engineering approaches should be considered if desired.

3. Where practical the RCS vents should be kept smaller than the size corresponding to the definition of a LOCA (10CFR50 Appendix A). This will minimize the challenges to the ECCS since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation although it may result in leakage beyond Technical Specification Limits. On PWRs the use of new or existing valves which

are larger than the LOCA definition will require the addition of a block valve which can be closed remotely to terminate the LOCA resulting from the inadvertent opening of the vent.

4. An indication of valve position should be provided in the control room.

5. Each vent should be remotely operable from the control room.

6. Each vent should be seismically qualified.

7. The requirements for a safety grade system is the same as the safety grade requirement on other Short Term Lessons Learned items, that is, it should have the same qualifications as were accepted for the reactor protection system when the plant was licensed. The exception to this requirement is that we do not require redundant valves at each venting location. Each vent must have its power supplied from an emergency bus. A degree of redundancy should be provided by powering different vents from different emergency buses.

8. For systems where a block valve is required, the block valve should have the same qualifications as the vent.

9. Since the RCS vent system will be part of the reactor coolant systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of non-condensible gas could be associated with substantial core damage, venting to atmosphere is unacceptable because of the associated released radioactivity. Venting into containment is the only presently available alternative. Within containment those areas which provide good mixing with containment air are preferred. In addition, areas which provide for maximum cooling of the vented gas are preferred. Therefore the selection of a location for venting should take advantage of existing ventilation and heat removal systems.

11. The inadvertent opening of an RCS vent must be addressed. For vents smaller than the LOCA definition, leakage detection must be sufficient to identify the leakage. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46.

BWR OWNERS' GROUP DISCUSSION:

The specific requirements and schedules are being developed in a continuing series of meetings between the utility owners' groups and the NRC Bulletins and Orders Task Force.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

The implementation of emergency procedures and retraining will be done on a schedule consistent with those established with the Bulletins and Orders Task Force.

LILCO'S RESPONSE:

The NEDO-24708 report, prepared by the BWR Owners' Group, of which LILCO is a participant, contains state-of-the-art analyses for various postulated accidents. The postulated accidents considered in NEDO-24708 and other material subsequently submitted to the Bulletins and Orders Task Force by the BWR Owners' Group included:

a. Small Break Loss-of-Coolant Accident,

b. Steam Line Break,

c. Detection and Mitigation of Inadequate Core Cooling,

d. Feedwater Line Break, and

e. Other Transients and Accidents analyzed in Chapter 15 of the FSAR.

The above analyses considered various combinations of the safety-related equipment available at the time of the transient or accident with the effect of operator actions also considered. The purpose of performing these analyses was to better understand the course of these events so as to provide reactor operators with realistic guidelines. General emergency guidelines, symptom oriented, have been developed through the efforts of General Electric and the BWR Owners' Group. These guidlines have been submitted to the NRC by the Owners' Group and are being utilized in the development of Shoreham emergency procedures. These procedures will be completed prior to Shoreham startup. The Shoreham operator training program will assure that shift personnel are thoroughly familiar with the emergency procedures and respond adequately to transient and accident conditions.

2.2.1.a Shift Supervisor's Responsibility

NUREG 0578 POSITION:

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

NRC CLARIFICATION

Shift Supervisor Responsibility (2.2.1.A)

NUREG-0578 Position (Position No.)	Clarification
Highest Level of Corporate Mgmt. (1.)	V.P. for Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures

Shift Supervisor Responsibility (2.2.1.A) (Continued)

NUREG-0578 Position (Position No.)	Clarification
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V.P. for Operations.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group agrees with the intent of the staff's position. However, in order to remove any ambiguity from the meaning of the term "accident situation" in item 2.b of the staff's position in Appendix A of NUREG 0578*, the entire sentence will be interpreted as follows: The shift supervisor (or equivalent, such as the supervising control operator in some plants), until properly relieved, shall remain in the control room at all times whenever a site or general emergency has been declared to direct the activities of control room operators.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

The staff's position will be implemented as stated and subject to the interpretation of item 2.b, as discussed above.

LILCO'S RESPONSE:

LILCO endorses the BWR Owners' Group position.

1. A management directive from the Vice President of Operations will be issued prior to fuel loading and annually reissued to clearly reinforce the Watch Engineer's** command duties and to emphasize that the prime responsibility of the Watch Engineer is the safe operation of the plant.

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* The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

** At the Shoreham facility, the term Watch Engineer is synonymous with Shift Supervisor.

2. Station procedures will be updated so that the duties, responsibilities and authority of the Watch Engineer and Control Room Operators are explicitly defined and include the following items:

   a. The responsibility and authority of the Watch Engineer will be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The objective that the Watch Engineer should not become totally involved in any single operation in times of emergency, when multiple operations are required in the control room, will be reinforced.

   b. The Watch Engineer, until properly relieved, will remain in the control room at all times whenever a site or general emergency has been declared to direct the activities of control room operators. Persons authorized to relieve the Watch Engineer will be specified in appropriate procedures.

   c. Any time the Watch Engineer is temporarily absent from the control room during routine operations, the lead control room operator will be designated to assume the control room command function. These temporary duties, responsibilities, and authority will be clearly specified in appropriate procedures.

3. Training programs for Watch Engineers will emphasize and reinforce the management functions of the Watch Engineer, which are to provide important safe plant operations.

4. The administrative duties of the Watch Engineer shall be reviewed annually by the Vice President of Operations. Administrative functions that detract from or are subordinate to the Watch Engineer's management responsibility for assuring the safe operation of the plant will be delegated, whenever possible, to other operations personnel not on duty in the control room.

2.2.1.b Shift Technical Advisor

NUREG 0578 POSITION:

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

NRC CLARIFICATION:

1. Due to the similarity in the requirements for dedication to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions. The performance of these two functions may be split if it can be demonstrated the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the STA will be dedicated to concern for the safety of the plant, our position has been that STA's must have a clear measure of independence from duties associated with the commercial operation of the plant. This would minimize possible distractions from safety judgments by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the STA position both in the STA job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor to STA duties as defined herein.

3. It is our position that the STA should be available within 10 minutes of being summoned and therefore should be onsite. The onsite STA may be in a duty status for periods of time longer than one shift, and therefore asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the STA (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent onsite presence. We do not intend, at this time, to specify or advocate a minimum time onsite.

4. The implementation schedule for the STA requirements is to have the STA on duty by January 1, 1980, and to have STAs, who have all completed training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the STAs on duty by that time should enhance the accident and operating experience assessment function at the plant.

BWR OWNERS' GROUP DISCUSSION:

Implementation of the Shift Technical Advisor (STA) as proposed by the Task Force would place a graduate engineer independent and detached from plant operations, in the control room at or shortly following the occurrence of an accident or abnormal transient. Because the STA would not be in the direct operational chain of command and, in fact, would not need to be licensed, he could neither manipulate nor direct licensed operators to manipulate the controls of the reactor plant. He would be empowered to advise operations but not responsible to operations for his advice.

The shift supervisor is correctly charged with the responsibility for safe operation of the plant at all times. During the early phase of an accident, he discharges this responsibility by coordinating and directing the response of the control-room staff. The actions of the operators are procedural, being governed by their training and emergency procedures, and during this phase the entire control room staff, including the shift supervisor, is completely occupied with responding to the accident. Plant operating experience indicates that there is a period of time following initiation of any accident or transient wherein the shift supervisor has sufficient time to analyze, diagnose, and respond to the condition of the plant but does not have sufficient time to carefully consider an independent assessment of the accident, resolve any conflicts between his and the independent assessment and, on the basis of such assessment, decide to alter the procedural actions of the operators.

Dialogue regarding such an assessment or time spent resolving such conflicts can only distract and delay the shift supervisor and consequently degrade the response of the control-room staff to the accident.

Even though the roles of shift supervisor and STA can be carefully delineated by procedure and training, industrial and military experience indicates that a direct-line organization wherein authority and responsibility are interdependent is required to effectively operate in a crisis environment. The proposed STA is empowered to advise operations but not responsible to operations for his advice. His authority and responsibility are not interdependent.

A potential for conflict and confusion exists which cannot be completely eliminated by procedures or training because procedure and training can address only those event sequences which have been postulated in advance. One important lesson learned from the experience at Three Mile Island and at other facilities is that not all event sequences can be postulated in advance. Therefore, an alternative which avoids this potential for conflict and confusion but improves the functions intended by the proposed STA is recommended.

Two functions are intended to be improved by the proposed STA: (1) accident assessment and (2) operating-experience assessment. In order to improve the accident-assessment function while avoiding the degradation in accident response which accompanies the proposed STA, the course of an accident is considered in three sequential phases: immediate, intermediate and recovery.

The immediate phase extends from the point at which an abnormal condition affecting plant safety can be detected in the control room until the point at which the shift supervisor has sufficient time to carefully consider an independent assessment and, on the basis of such assessment, decide to alter the procedural actions of the operators. The intermediate phase extends from the end of the immediate phase until the point at which the Technical Support Center (TSC) is manned and ready. The recovery phase extends from the end of the intermediate phase until the point at which recovery is complete.

For the immediate phase, the accident-assessment function can be improved only by upgraded training to enhance the operators' abilities to recognize, diagnose, and respond to accident conditions. During this phase, the operators' actions are governed by training and emergency procedures, and by definition there is insufficient time for the careful consideration of an independent assessment which would be required before such an assessment could become the basis for altering the procedural actions of the operators.

For the intermediate phase, the accident assessment function can be improved by either of two alternative means. An operator can be educated in science and engineering in order that he might provide an assessment which could be considered and acted upon by the shift supervisor. Alternatively, a graduate engineer or equivalent can be trained in plant operations and made available to the shift supervisor on call in order that he might provide such an assessment. In either case, the shift supervisor must have sufficient time to carefully consider the assessment and, based on such assessment, decide to alter the procedural actions of the operators.

For the recovery phase, the accident assessment function can be improved by manning the TSC. The collective engineering resource within the TSC will be able to develop a detailed independent assessment of plant conditions and provide appropriate procedures with which to recover from the accident.

The operating experience assessment function can best be provided by a team which reviews operating experience at the plant and at plants of like design. Varying team membership, as appropriate to the operating experience being assessed, assures accomplishment of this function by the best qualified individuals.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

The two functions intended to be improved by the proposed STA will be improved as follows:

1. Accident Assessment

   a. Immediate Phases

      An operator or supervisor in the direct operational chain of command on each shift (normally in charge in the control room) will receive additional specific training in the response and analysis of the plant for transients and accidents. This training will be coordinated with the schedule for preparation and review of analysis and guidelines under the NRC Bulletins and Orders Task Force.

      All other operators and supervisors will receive additional training appropriate to their responsibilities in the response of the plant to transients and accidents. These longer term training and qualification criteria will be provided by the Institute of Nuclear Power Operations.

b. Intermediate Phase (Alternatives)

An operator or supervisor in the direct operational chain of command on each shift will receive substantial additional education in basic engineering and science sufficient to aid him in assessing unusual situations not explicitly covered in the current operator training.

- OR -

A graduate engineer or equivalent trained in the response and analysis of the plant for transients and accidents and in plant design and layout, including the capabilities of instrumentation and controls in the control room, will be available to the individual in charge in the control room on call. He may be stationed on or off site as appropriate to plant location, communication capabilities, operator training and education, extent and detail of emergency procedures, etc.

c. Recovery Phase

Individuals knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident will be available on call to staff the On-Site Technical Support Center.

2. Operating Experience Assessment

Where it does not already exist, a team will be designated by the licensee to assess the operating experience at his plant or plants and at plants of like design. Team membership may vary as appropriate to the operating experience being assessed but will include experience in systems engineering and familiarity with or routine access to persons experienced in the principles of human engineering or human factors.

LILCO RESPONSE:

LILCO endorses the BWR Owners' Group position. However, an on-shift technical advisor (STA) or alternate, in accordance with D. B. Vassallo's letter dated October 10, 1979 (Enclosure 2, "Alternatives to Shift Technical Advisors," provided herein) will be provided for the Shoreham facility. In addition, other guidelines and criteria developed by industry groups such as the Institute of Nuclear Power Operations will be evaluated.

ENCLOSURE 2

ALTERNATIVES TO SHIFT TECHNICAL ADVISORS

The recommendation by the Lessons Learned Task Force that an on-shift Technical Advisor be required at operating nuclear power plants has received much comment and attention by the ACRS and industry representatives since NUREG-0578 was published. Several alternative approaches have been suggested. The ACRS has advised and the Director of NRR has decided that alternatives be considered and approved if found by the staff to satisfactorily accomplish the functions described by the Task Force for the Shift Technical Advisor. As an aid to evaluating alternatives, a more comprehensive discussion of the purpose and basis of the Task Force recommendation is provided below. The discussion is in terms of the two principal functions intended to be accomplished and the characteristics thought to be necessary to effectively accomplish these functions. It is intended that the licensing review staff make use of this discussion in evaluating alternatives proposed by licensees and license applicants.

Introduction

As stated in NUREG-0578, the Lessons Learned Task Force has concluded that the need for improved operations is the most important lesson learned from the accident at TMI-2. One key element so far identified is the need to improve the capability in the control room to recognize and diagnose unusual events. Over the next several years, improvements in the capability of the reactor operations staff to respond to unusual events can and will be sought through improvements in plant design, operating procedures and the qualification and training of operators. Improvements in plant design are expected to include improvements in the area of human factors, especially improvements in display and diagnostic systems available to aid operators. For example, the Task Force made a short term recommendation for improvement of the means of assessing inadequate core cooling. The Task Force also made short term recommendations for improvements in emergency procedures and preparations by the plant operations organization. The purpose of these recommendations is to assure that the operators and the onsite operational and technical support personnel are organized both administratively and physically in an effective manner. In addition, improvements in the licensing requirements for operators have been recommended to the Commission. Over the coming months, it is likely that further increases in qualification and training requirements for operators will be developed by the industry's recently announced Nuclear Operations Institute for implementation over the next several years. Because these changes are necessary but difficult to achieve rapidly, the Lessons Learned Task Force has recommended the use of Shift Technical Advisors as a method of immediately improving the operating staff capabilities for response to off normal conditions and for evaluating operating experience.

The consensus of the Task Force is that there are two necessary improvements in the capability to assess the status of a plant during unusual conditions such as a transient or an accident, to realize the significance of the available information such as instrument readings, and to take appropriate action. First, there should be an accident assessment capability based on a comprehensive education in engineering and science subjects related to nuclear power plant design and on training and experience in the dynamic response of the specific plant. This capability must be rapidly available in the control room in the event of an accident. Second, there should be a capability to maintain and upgrade safe plant operations through the cognizance and evaluation of applicable operating experience by an engineering group with diverse technical knowledge, experience, and perspective in relevant areas such as electrical, mechanical and fluid systems and human factors. The addition of Shift Technical Advisors to the plant operating staff is an acceptable means of supplying both of these functions. Alternative manning and organizational schemes will be considered and will be evaluated for satisfaction of the qualifications, training and duty assignment criteria discussed below.

Discussion

In developing the recommendation for the Shift Technical Advisor, the Task Force concentrated on the two functions that needed to be provided, namely, an accident assessment function and an operating experience assessment function. The proper performance of these functions requires the provision of certain characteristics described in the following paragraphs.

A. Accident Assessment Function

1. General Technical Education

The technical education of at least one person in the control room under off normal conditions should include basic subjects in engineering and science. The purpose of this education is to aid the operator in assessing unusual situations not explicitly covered in the current operator training. The following is a tentative list of areas of knowledge that are considered to be desirable:

Mathematics, including elementary calculus

Reactor physics, chemistry and materials

Reactor thermodynamics, fluid mechanics, and heat transfer

Electrical engineering, including reactor control theory

These areas of knowledge should be taught at the college level and would be equivalent to about 60 semester hours. Although a college graduate engineer would have many of these subjects and more that would not be essential, some engineers might be deficient in a few of these specific areas, e.g., reactor physics. Although the time to teach these subjects to a licensed senior reactor operator could be as short as two years, depending on the scope and content of the subjects, the selection of a graduate engineer would likely be a more rapid means of fulfilling this characteristic.

2. Reactor Operations Training

All persons assigned to duties in the control room should be trained in the details of the design, function, arrangement and operation of the plant systems. This training is necessary to assure that the meaning and significance of instrument readings and the effect of control actions are known. A licensed operator or supervisor of an operator would not be required to have further training in order to fulfill this characteristic. A graduate engineer not previously licensed or trained as an operator or senior operator would require additional training in order to fulfill this characteristic.

3. Transient and Accident REsponse Training

In addition to the training in normal operations, anticipated transients, and accidents presently required of operators and senior operators, one person in the control room under off normal conditions should be trained to recognize and react to a wide range of unusual situations including multiple equipment failures and operator errors. This training should not be limited to written procedures or specific accident scenarios, but should include the recognition of symptoms of accident conditions such as complex transient responses or inadequate core cooling and possible corrective actions. The purpose of this training is to broaden the ability for prompt recognition of and response to unusual events, not to modify the instinctive, rapid procedural response to transients and accidents provided by reactor operators. The training is required in recognition of the fact that real accidents inherently are initiated and accompanied by unusual and unexpected events. The training is also to emphasize need to focus on the essential parameters that indicate the status of the core and the primary coolant boundary. This additional training would take up to a year to accomplish for a person not already experienced in nuclear plant transient and accident analysis or evaluation. Both inexperienced graduate engineers and currently licensed operators would require additional training to fulfill this characteristic.

4. Detachment from Operations

The plant response assessment function requires a measure of detachment from the manipulation of controls or immediate supervision of operators. This is intended to provide the perspective and the time for assessing plant conditions and advising on appropriate operator actions. It has been called a safety monitor characteristic. Currently only three operators would normally be in the control room at the time an unusual event occurred, and it is allowed that at times there would be fewer. This number is only enough to satisfy the demands for prompt control and supervisory actions under off normal conditions. The time necessary to make a considered assessment and permit independent monitoring of plant safety require one more person in the form of the Shift Technical Advisor or some alternative in the control room.

5. Independence from Operations

In order to provide both perspective in assessment of plant conditions and dedication to the safety of the plant, this function should have a clear measure of independence from duties associated with the commercial operation of the plant. In an accident situation where command authority should not be diluted, complete independence is not desirable and is not necessary to the safety assessment function.

6. Availability

This capability should be readily available in the control room, preferably immediately at all times, but at most within ten minutes. Having this capability on duty for each shift is the best approach.

B. Operating Experience Assessment Function

1. Independence from Operations

A measure of independence is required to provide for effective safety monitoring of operating experience at the individual plant and at plants of like design. The assessment of operating experience at the assigned plant and other similar plants and the routine monitoring of the safety of plant operations is usually compatible with and necessary for efficient operations. However, the demands of commercial operation can sometimes distract from or appear to override safety judgments. An independent monitoring of the safety of plant operations is intended to counter-balance the immediate and pressing needs of commercial operation.

2. Dedication

Personnel should be dedicated to the function of safety monitoring of operating experience as their primary responsibility and duty. Although reactor operating personnel have a commitment to safety that derives from self interest as well as regulatory requirements, it is only one of two primary responsibilities, the other being the continuous production of power. The assignment of safety evaluation of operating experience as a primary responsibility for certain specified individuals will reduce potential conflicts and assure adequate time to discharge the duties.

3. Diversity of Technical Knowledge

The technical knowledge of those assessing operating experience should be diverse and encompass all technical areas important to safety. The types of problems that can affect safety include all areas related to the design and operation of nuclear power plants; e.g., mechanical, electrical and fluid systems and reactor physics, chemistry and metallurgy. Recognition and understanding of a problem and its significance requires some knowledge in the relevant technical specialities and cannot depend solely on the descriptions and judgements of the persons identifying and reporting the problem. Because of the broad scope of possible technical areas and the possible interactions of components, equipment and systems, the people engaged in operating experience review should have experience in areas usually designated as systems engineering. They should also be graduate engineers, or equivalent. In addition, because of the importance of operator actions in the safety of plant operations, familiarity with or routine access to persons with the principles of human engineering or human factors should be provided.

Alternatives

As discussed in NUREG-0578, several alternative means of providing the accident assessment function were considered by the Lessons Learned Task Force. They were:

1. Upgrade the requirements for reactor operators and senior reactor operators to include more engineering and plant response training.
2. Provide additional on-shift personnel with science or engineering training and specific traning in plant design and response.
3. Provide on-call assistance to the control room by identified personnel in the plant engineering organization having the training described in alternative 2.

Although the Task Force initially assumed that the accident assessment function would be combined with the operating experience assessment function, it is possible that the two functions could be separated. Some have suggested that people with the education, training, and experience required for both the operating experience assessment function and the safety monitoring function would be more easily obtained and retained if not required to work on shift. Others believe that such people can be retained if sufficient incentives are provided. The advantages and disadvantages of these alternatives are discussed below. Although no alternative other than a group of dedicated Shift Technical Advisors has so far been found acceptable, it is possible that innovative improvements in the other alternatives could be found acceptable.

Discussion of Alternatives

1. Upgrade the training and qualifications of the senior reactor operator. This alternative would require no change in the present number or organization of control room operators. The debilitating feature of this alternative is that the senior operator would be busy directing the reactor operators or taking actions himself during an accident and not have sufficient time or perspective to make the desired assessment of plant conditions; i.e., perform the safety monitor function. This arrangement would also not provide a clear independence from commercial operation. However, the capability would be readily available when needed. It is unrealistic to expect the senior operator to fulfill the operating experience assessment function. A separate group could be established to accomplish that function on the day shift when interaction with offsite experts and utility management would be enhanced. If schemes are proposed to accomplish the two functions separately, then they should include mechanisms

for sufficient coupling of the two to assure continuous feedback of and ready access to the knowledge being acquired in operating experience evaluation.

2. Additional on-shift personnel

This alternative would require the addition of one person to the on-shift control room staff. If the person is to be a Shift Technical Advisor, no license would be required, thus making the position easier to fill quickly. However, detachment from first-line commercial operations decisions can be attained by either a line or advisory position. For example, instead of the Shift Technical Advisor proposed by the Task Force, there may be acceptable methods of using a Shift Engineer, who normally has authority over a Shift Supervisor, to perform the accident assessment function. Either approach would utilize people on shift so they would be readily available. Since the Shift Engineer would have normal duties other than operating experience assessment, a separate day shift group would be required to fulfill that function if the shift engineer was found to be an acceptable source of the accident assessment (safety monitor) function.

3. On-call assistance

This alternative would require no additional on-shift personnel. Others have susggested that provision of the recommended technical education and training would be most easily accomplished with this alternative since degreed engineers with intimate knowledge of the plant design basis and accident response characteristics are available in the utility technical staff. Since these personnel would be remote from the control room, a requirement to be licensed does not appear to be consistent. Knowledge of accident response might also be more easily found among vendor personnel who have extensive experience in accident analysis and systems design. This alternative also provides detachment from actual operation and some independence from commercial operation. However, these people would not be readily available when needed. The use of utility or vendor personnel not at the site would increase the difficulties of communication. Although there is need for backup assistance from these other organizations, it is doubtful that they would be able to provide for the prompt response needs of the accident assessment function and they do not have sufficient plant unique experience and familiarity to satisfy the operating experience assessment function.

2.2.1.c Shift and Relief Turnover Procedures

NUREG 0578 POSITION:

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

    a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist).

    c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist);

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transients (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

NRC CLARIFICATION:

No clarification provided.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group agrees that knowledge of plant status, expecially for thos systems required to mitigate the consequences of an accident, should be transferred in a systematic manner from one shift to the next. The Group is also convinced that to be most effective as a means of information transfer in the course of a shift or relief turnover, the information must be limited to that which can be summarized on a single list on a single piece of paper. Furthermore, the information provided by the list should be reviewed not only by the shift supervisor and control room operators, but by other plant personnel (auxiliary operators, technicians, etc.) as appropriate, thus eliminating the need for separate checklists, as apparently required in the Staff's position.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. A checklist will be devised to ensure that control room status of systems that are required to mitigate the consequences of an accident are monitored on a shift turnover basis. This list will include system lineups and alarms located in the main control room. Systems and components in a degraded condition will be identified as required by plant status.

2. The checklist will be kept in the control room at all times.

3. The checklist will be reviewed by personnel other than the shift supervisor and control room as appropriate.

LILCO'S RESPONSE:

The Shoreham Nuclear Power Station has a procedure for operations staff shift relief turnover. This procedure will be reviewed and revised as necessary to assure that the above requirements are addressed.

2.2.2.a Control Room Access

NUREG 0578 POSITION:

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

NRC CLARIFICATION:

No clarification provided.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group agrees that it is necessary to limit access to the control room and to establish a clear line of authority and responsibility in the control room in the event of an emergency.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

Procedures will be developed and implemented which will meet the intent of the staff's position.

LILCO'S RESPONSE:

Appropriate procedures will be prepared for the Shoreham Station to limit access to the Control Room to those individuals responsible for the operation of the plant such as Operations Supervisors, Watch Engineers, Control Room Operators; technical advisors as requested and predesignated NRC personnel. The procedure(s) will clearly establish the following:

a) The authority and responsibility of the person in charge of limiting access to the Control Room;

b) The line of authority and responsibility in the Control Room in the event of an emergency; and

c) The lines of communications and authority for plant management personnel not in direct command of operations, including those who report to stations outside the Control Room.

2.2.2.b Onsite Technical Support Center

NUREG 0578 POSITION:

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plants as necessary to incorporate the role and location of the technical support center.

A complete set of as-built drawings and other records, as described in ANSI N45.2.9-1974, shall be properly stored and filed at the site and accessible to the technical support center under emergency conditions. These documents shall include, but not be limited to, general arrangement drawings, P&IDs, piping system isometrics, electrical schematics, and photographs of components installed without layout specifications (e.g., field-run piping and instrument tubing).

NRC CLARIFICATION:

1. By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

   A. Establish a TSC and provide a complete description,

   B. Provide plans and procedures for engineering/management support and staffing of the TSC,

   C. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC. Provide, between the TSC and the control room, a capability for the transmittal of some data. This requirement could be satisfied by closed circuit television or process computer printout,

   D. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets, or evacuation t the control room),

E. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC,

F. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable, and

G. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size & Staffing

The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4. Activation

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event"; however, the activation of the TSC is discretionary for that class of event.

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be given to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum, data (historical in addition to current status) should be available to permit the assessment of:

Plant Safety Systems Parameters for:
- Reactor Coolant System
- Secondary System (PWRs)
- ECCS Systems
- Feedwater & Makeup Systems
- Containment

In-Plant Radiological Parameters for:
- Reactor Coolant System
- Containment
- Effluent Treatment
- Release Paths

Offsite Radiological
- Meteorology
- Offsite Radiation Levels

8. Data Transmission

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

9. Structural Integrity

A. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

B. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. Habitability

The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

A. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

B. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

C. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

D. Dose reduction measures such as breathing apparatus and potassium iodide tablets cannot be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group agrees that it is important to have a technical support center (TSC) designated where "individuals who are knowledgeable of and responsible for engineering and management supports of reactor operations in the event of an accident" can go to, consistent with the intent to limit access to the control room. Furthermore, it is agreed that it is appropriate that the emergency plants will designate the role and location of the technical support center. There is, however, one area in particular which needs further discussion.

The requirement that the TSC be onsite and in close proximity to the control room is not necessarily the best choice under all circumstances for meeting the intent of the position. The location of the TSC should be dictated by its accessibility to the engineering and management personnel who will occupy it, rather than by its physical proximity to the control room. For example, multi-unit sites which share engineering and management personnel, or so-called outdoor sites which have administrative buildings detached from the plant, may designate locations which may not be judged as in close proximity to the control room, but make sense from a personnel access viewpoint. Furthermore, "close proximity" would only seem to be required as a means of supplementing the transmittal of plant status from the control room to the TSC, and in that sense then becomes inconsistent with the desire to limit access to the control room during emergencies. Thus, the requirements for close proximity could be eliminated on the basis that the plant status must be monitored from the TSC.

The Owners' Group also agrees that monitoring equipment may vary from plant to plant, and that there is no single best way in which to monitor plant status in the TSC. There was agreement that TV monitors which could read and transmit information from the control room panels to the TSC would meet the requirement to display and transmit plant status. It was also agreed that the TSC should have two-way communication links with the control room, other onsite telephones, the offsite Emergency Operations Center, and the NRC. It was further agreed that the existing direct link between the NRC and the control room would be switched over to the TSC upon its activation in accordance with the intent to limit access to the control room. Finally, it was agreed that the staffing and activation criteria for the TSC would be specified in the emergency plan.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA

PHASE I (By January 1980)

1. A location will be designated in the emergency plan. This may be a temporary location.

2. Communication links will be established with the control, the onsite Operational Support Center, the offsite Emergency Operations Center, and the NRC. These may be temporary.

3. The staffing and activation criteria will be specified in the emergency plan.

4. The TSC will have access to the records (system descriptions, arrangement drawings, etc.) in accordance with the revised NUREG 0578 position.

The implementation criteria of Phase II will be issued after further discussions between the Owners' Group and the NRC staff.

LILCO'S RESPONSE:

LILCO will provide a technical support center (TSC) onsite prior to fuel load. A location will be designated in the SNPS emergency plan. The TSC will have communication links with the control room, the onsite Operation Support Center, the offsite Emergency Operations Facility, and the NRC. The TSC staffing and activation criteria will be specified in the SNPS emergency plan. The TSC will also have access to system descriptions, arrangement drawings, and other plant records in accordance with the Staff's position. For a description of the conceptual design currently being implemented refer to Appendix A, enclosed herein. This information was previously submitted to the NRC via LILCO letter SNRC-486 from J. P. Novarro to H. Denton, dated July 21, 1980.

APPENDIX A

J.O. No. 11600.02

July 22, 1980

DESIGN CRITERIA AND DESCRIPTION

TECHNICAL SUPPORT CENTER

SHOREHAM NUCLEAR POWER STATION - UNIT 1
LONG ISLAND LIGHTING COMPANY

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1.0 GENERAL CRITERIA AND DESCRIPTION

1.1 General Criteria

A separate Technical Support Center (TSC) shall be provided for use by plant management, technical, and engineering support personnel. In an emergency, this center shall be used for assessment of plant status and potential offsite impact in support of the control room command and control function. The center should also be used in conjunction with implementation of onsite and offsite emergency plans, including communications with an offsite emergency response center. Provide at the onsite Technical Support Center the as-built drawings of general plant arrangements and piping, instrumentation, and electrical systems. Photographs of as-built system layouts and locations are an acceptable method of satisfying some of these needs.

1.2 General Description

The second floor of the security building will be upgraded to serve as the TSC by the addition of filtered ventilation, computer generated system and radiological parameter displays and a backup power supply.

The TSC staffing and activation criteria and interaction with the Emergency Operations Facilities will be specified in the Shoreham Nuclear Power Station - Unit 1 (SNPS-1) Emergency Plan.

The TSC will be operational by fuel load.

2.0 DESIGN CRITERIA AND DESCRIPTION

2.1 Location/Space

2.1.1 Criteria

The TSC shall be located in proximity to but separate from the control room, and within the plant security boundary. The facility shall be of sufficient size to accommodate those operating the TSC, NRC, and vendor representatives as well as the required equipment and technical data.

2.1.2 Description

The existing security building is a separate structure located on the north side of the plant, as shown on the Site Arrangement Plan, Figure 1. The entire second floor of approximately 4,000 sq ft consisting of lecture and classrooms, an office, a library and toilets will be made available as the TSC on a joint basis. The first floor will continue as the security facility although it will be within the protected (habitable) environment provided for the entire building due to the TSC requirements.

The existing floor plan is shown on Figure 2. It will provide ample space for 25 people.

2.2 Structural/Architectural

2.2.1 Criteria

The TSC need not be designed to seismic Category I requirements. It shall be well built in accordance with sound engineering practice, with due consideration to the effects of natural phenomena which may occur at the site.

2.2.2 Description

The existing security building will be modified as necessary to accommodate the functions of a TSC.

2.2.2.1 Existing Structure

The security building superstructure is of steel framed construction supported on reinforced concrete spread footings. The energy efficient curtain wall design utilizes insulated cavity wall construction. The roof deck and intermediate floor slab are of reinforced concrete construction, with the roofing material comprised of insulated, built-up asphalt and gravel.

2.2.2.2 Building Modifications

The existing roof level HVAC penthouse will be expanded to accommodate additional mechanical equipment. This penthouse expansion will be of a similar construction as the existing security building and will complement the existing architectural style.

Additional building modifications will include the architectural sealing of the building to develop the ability to sustain the positive internal pressure required for TSC occupation.

This will be accomplished by providing existing doors and frames with appropriate weather stripping and gaskets. Should such modifications provide inadequate reduction of air leakage, the interior walls will be coated with an impermeable coating system where necessary.

2.3 Habitability

2.3.1 Criteria

The TSC shall be designed to protect personnel from radiological hazards including direct radiation and airborne contaminants in accordance with General Design Criterion 19 and Standard Review Plan 6.4. Limits of 5 rem whole body, 30 rem thyroid, shall not be exceeded for the duration of the accident considering major sources of radiation.

Monitoring shall be provided for both direct radiation and airborne radioactive contaminants. The monitors should provide warning if the radiation levels in the support center are reaching levels approaching the design limits. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodine tablets, or evacuation to the control room).

2.3.2 Description

The security building meets these criteria, as follows:

1. Credit is taken for mixed mode release; see Attachment 1 for justification, and

2. The TSC atmosphere is filtered through a Charcoal-HEPA filter. See Attachment 2 for a discussion of the analysis. This is achieved by upgrading the security building HVAC system as discussed in Section 2.4.

   The 30 day integrated doses calculated based on the above are:

	Total 30 Day Integrated Dose (Rem)		
	Thyroid	Gamma	Beta
Mixed Mode Release & 95 percent Halogen Filter	16	0.348	3.75

These are within the limits of General Design Criterion 19. Local wall-mounted area radiation monitors will be provided to measure radioactivity within the TSC and a ventilation monitor with an iodine cartridge will measure recirculated airborne levels. Action levels to define when protective measures should be taken (including evacuation) will be designated.

2.4 Heating, Ventilation, and Air Conditioning

2.4.1 Criteria

Permanent ventilation systems, including particulate and charcoal filters, shall be provided. These systems need not be qualified as ESF systems. However, the design and testing guidance of Regulatory Guide 1.52 shall be followed except that the systems need not be redundant, seismic, instrumented in the control room, or automatically activated. In addition, HEPA charcoal filters need not be tested as specified in Regulatory Guide 1.52 nor meet the QA requirements of 10CFR50 Appendix B. Spare parts shall be readily available.

2.4.2 Description

To pressurize the security building atmosphere, 2,000 to 3,000 cfm of filtered outside air will be supplied to the building. Provision has been made in the system design for 3,000 cfm maximum outside air, with recirculation capability of up to 1,000 cfm.

A 3,000 cfm capacity Charcoal-HEPA filter train with booster fan will be installed on the roof of the security building inside an extension of the existing equipment room. This filter train will remove, with 95 percent efficiency, the gaseous iodine, methyl iodine, and any particulates from the outside air, reducing concentrations to within acceptable limits.

In order to use the outside air of 2,000 to 3,000 cfm for pressurization only, exhaust from the second floor lecture hall, toilets, and locker area will be eliminated by shutting down roof fans and securely closing dampers. In addition, the main exhaust damper will be closed securely. Procedures shall be provided to ensure all necessary actions are completed upon manning the TSC.

Existing system controls will be modified to suit the new design requirements and to maintain positive pressure following a DBA. A central control center will be provided for remote manual operation of the HVAC system during an accident. This will include push buttons for all the manual-controlled, power operated dampers, startup of the filter booster fan, and direct expansion air conditioning.

A conceptual study sketch (Figure 3) showing a schematic of the existing security building HVAC system and proposed modifications is attached.

Provisions for a direct expansion (DX) refrigeration system have been made to meet the requirements of heat gains due to equipment, outside air, lights and power, and personnel occupany, and loss of office and service building chilled water supply. During an accident, the TSC HVAC will be self contained and its power source will be from a backup power supply as discussed in Section 2.6.

Regulatory Guide 1.52, Design Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units, will be followed as required to meet the criteria as stated in Section 2.4.1. Spare parts will be readily available as will procedures for replacing failed components.

2.5 Instrumentation

2.5.1 Criteria

The TSC shall have the capability to display plant parameters and equipment status to technical and management personnel responsible for engineering and support of reactor operations (control room activities) following an accident. The TSC capability to assess plant parameters shall be independent from actions in the control room. The TSC equipment is not required to be safety grade or redundant.

The data between the beginning of the accident (t=0 defined as initial event, e.g., reactor scram or turbine trip) and the time of activation of the TSC shall not be lost and shall be available at the TSC.

The instrumentation in the TSC shall not degrade plant installed safety-grade instrumentation and equipment.

2.5.2 Description

The TSC data display will be entirely computer based and will be provided by way of an enhancement of the existing process computer (PCS) and digital radiation monitoring computer (RMS) systems. Refer to FSAR Sections 7.5.1.6 and 7.5.2.7 on the process computer system and to Sections 11.4 and 12.3.4 on the digital radiation monitoring system.

The selection of parameters will be based on capabilities to:

1. Diagnose initial event/accident,
2. Evaluate performance of safety related systems,
3. Ensure that the plant is in a stable shutdown condition following an accident,
4. Monitor offsite (portable) and onsite radiological data, and
5. Monitor meteorological data.

2.5.2.1 In-plant System Parameters

Presentation of in-plant system parameters will be provided at the TSC by the process computer system. Data will be presented by a color graphics CRT display with keyboard access. Three high speed typers will be provided for hard copy record. One typer, a KSR (Input/Output) type, will provide user demand request capabilities and will receive outputs from existing NSSS and BOP post-trip logs as well as significant in-plant system alarms. The other two typers will be of the RO (Receive Only) type; one will provide the same alarms being presented on the main control room alarm typer and the other will provide output or the process computer TSC data log (Section 2.5.2.3).

The entire process computer system data base will thus be available, on demand, for TSC display. Attachment 3 provides a listing of specific data points which will be available at the TSC as part of the PCS TSC historical data file and log (Section 2.5.2.3).

Additional data will be provided to the process computer to ensure plant safety parameter availability at the TSC. All Class 1E signals required at the TSC will be isolated prior to input to the process computer to ensure that these signals are not jeopardized or degraded by the operation of, or failure of, the process computer system. Qualified isolation devices will, in these cases, be inserted into those required Class 1E circuits to provide this assurance. A new TSC termination cabinet will be located in the relay room and will be designed as a central gathering point for all required additional data prior to process computer system input.

2.5.2.2 Radiological Data

Presentation of in-plant radiological parameters and meteorological data will be provided at the TSC by the digital radiation monitoring computer system. Data will be presented by a color graphics CRT display with keyboard access. One high speed typer, a KSR (input/output) type will be provided for hard copy records by way of user demand requests and outputs from the RMS TSC data log and historical file (Section 2.5.2.3).

The entire RMS computer system data base, including off-site dose calcuations, will be available, on demand, for TSC display. Attachment 4 provides a listing of specific data points which will be avilable at the TSC as part of the RMS TSC historical data file and log (Section 2.5.2.3).

Additional data, including wide ranges on effluent process monitors, will be provided to the radiation monitoring computer to ensure radiological and meteorological parameter availability at the TSC.

2.5.2.3 TSC Logs and Historical Data Files

Two logs and historical data files will be provided, one by way of the process computer system (PCS) and the other by way of the radiation monitoring system (RMS) computer. Pre-event historical data files and post-event logging of data will provide TSC personnel with the capability to diagnose the initiating event and its radiological consequences, as well as provide an immediate evaluation of safety systems performance and plant status.

2.5.2.3.1 TSC Log/Historical File - In-plant System Parameters

The process computer TSC historical data log will initiate upon receipt of an external event signal (t=0) a printout, on a TSC typer, of those in-plant system parameters assigned to this log (Attachment 3). The log will continue printing out data (the frequency of a data point printout will depend on its assigned scan rate) until manually terminated. A 5 minute pre-event data file (i.e., history) of these selected TSC log parameters will be stored by the process computer to be recalled to the TSC, on demand, using the TSC KSR typer.

2.5.2.3.2 TSC Log/Historical File - Radiological/Meteorological Parameters

The RMS computer log and historical files provided will be similar to the process computer TSC log and data file (Section 2.5.2.3.1) thus providing both pre- and post-event radiological/meteorological data record.

2.6 Electrical Power Supply

2.6.1 Criteria

An emergency power supply shall be provided for the permanent ventilation system of the TSC. The power supply to the TSC instrumentation need not meet safety grade requirements, but shall be reliable and of a quality compatible with the TSC instrumentation requirements. The power supply for instrumentation shall be continuous once the TSC is activated.

2.6.2 Description

The security building facilities are presently supplied from a 300 kVA 480-120/208 V transformer through an automatic transfer switch which receives power from buses 11C and 12C. Black power from an on-site diesel will be connected to the alternate side of the automatic transfer switch instead of the feed from bus 12C. The normal supply to the transfer switch still has access to both sources of offsite power by virtue of the transfer scheme on the 4 kV switchgear and manually through the tie-breaker in the 480 V double-ended load center. This arrangement allows access to two sources of offsite power and a diesel generator and will carry existing and added HVAC, lighting and other necessary loads.

The power requirements for the added computer points and the peripheral equipment in the TSC will be supplied from the existing computer inverter which is connected to the safety diesels. The inverter serves as an isolation device so that the computer does not have to be tripped on a LOCA signal. Power for power supplies associated with isolation devices for added instrumentation will be fed from the appropriate safety related buses.

2.7 Communications

2.7.1 Criteria

Communication links shall be established with the control room, the onsite Operational Support Center, the offsite Emergency Operations Facility, and the NRC.

2.7.2 Description

Bell System phone lines will be used for communication with the NRC, the onsite Operational Support Center, and the offsite Emergency Operations Facility, with optional lines to the Nuclear Steam Supplier and the Architect Engineer. An existing line to NAWAS will be available.

Communication to the control room will be by page/party, and the plant pbx phone systems.

2.8 Records

2.8.1 Criteria

A complete set of as-built drawings and other records, as described in ANSI N45.2.9-1974, shall be properly stored and filed at the site and accessible to the TSC under emergency conditions. These documents shall include, but not be limited to, general arrangement drawings, P&IDs, piping system isometrics, electrical schematics, and photographs of components installed without layout specifications (e.g., field-run piping and instrument tubing).

2.8.2 Description

Critical documents such as Emergency Procedures, System Descriptions, and General Arrangement, Flow, Logic and Elementary Schematic Drawings will be available in the TSC and the balance will be available in the plant Records Center.

ATTACHMENT 1

SECURITY BUILDING

TSC X/Q CALCULATIONAL TECHNIQUE

Murphy and Campe identifies the technique that is to be utilized to evaluate X/Q values to be used in plant habitability calculations (see Standard Review Plan 6.4). The technique identified applies to a Design Basis Accident (DBA) release emanating from some wall of the containment structure. Historically, DBA X/Q calculational techniques have been conservatively limited to ground level release criteria except for releases from stacks 2 1/2 times the height of the nearest adjacent building (see Regulatory Guide (RG) 1.145).

The release from the Shoreham DBA is unique in that the reactor building standby ventilation vent fulfills all seismic criteria. Instead of the release leaking through portions of the primary and secondary containments, it is confined to exit through the vertical vent atop the secondary containment structure. This vent is higher than any adjacent building in the plant. Thus, a more appropriate approach to consider X/Q calculation would be to utilize the mixed-mode release concept identified in RG 1.111, Revision 1, Position C2b. To this end, the governing Murphy and Campe equation can be married with the RG 1.111 Position C2b concept (which was developed from atmospheric tracer tests sponsored by the Atomic Industrial Forum at Millstone) to produce the following working equation:

$$\chi/Q = \frac{E_T}{\bar{u}\,(\pi\sigma_y\sigma_z + \frac{A}{K+2})} + \frac{(1-E_T)\,\mathrm{Exp}\left[-\frac{1}{2}\left(\frac{h_e}{\sigma_z}\right)^2\right]}{\bar{u}\pi\sigma_y\sigma_z}$$

Where:

E_T = entrainment coefficient

E_T = 1 for $W_{o}/\bar{u} \le 1$

E_T = 2.58-1.58 ($W_{o}/\bar{u}$) for $1 < W_{o}/\bar{u} \le 1.5$

E_T = $0.3-0.06\ (W_o \bar{u})$ for $1.5 < W_o \bar{u} \leq 5.0$

E_T = 0 for $W_o \bar{u} > 5.0$

$\bar{u}$ = wind speed at 10-m level (m/sec)

σ_y = horizontal dispersion coefficient (m)

σ_z = vertical dispersion coefficient (m)

A = containment building area (m^2)

K = $\frac{3}{(S/d)^{1.4}}$

where:

S = source to receptor distance (m)

d = containment diameter (m)

h_e = effective stack height (m)

where:

$h_e = h_s + h_{pr} - h_t$

h_s = height of vent release (m)

h_{pr} = nonbuoyant plume rise (m)

h_t = height of TSC roof above plant grade (m)

W_o = stack exit velocity (m/sec)

It is conservatively assumed that the 10-m wind speed applies to the elevated portion of the release.

RG 1.145 also identifies a fumigation condition as limiting for elevated (or partially elevated) releases during accident conditions. Seabreeze fumigation occurs only when the winds are blowing onshore. Examination of the relative locations of the shoreline, containment structure, and TSC, clearly shows the fumigation from the containment can only occur in the opposite direction from the TSC. Thus, this condition yields a zero X/Q at the TSC.

The final consideration is to identify a 5 percent worst condition for this type of release. In order to introduce more conservatism into the calculational techniue, the meteorological condition producing the highest (worst) X/Q value (i.e. 0.01 percent) was assumed to occur for the first 8 hr of the accident (0-8 hr period). An additional conservatism, nonbuoyant plume rise, due to the momentum of the release, was presumed to be zero, even though RG 1.111, Revision 1 recommends its consideration.

The X/Q's shown below were determined as being the highest (worst) values for the TSC. For the mixed-mode release, the 0-8-hr X/Q value maximized during Pasquill stability Class D (neutral) with a wind speed of 10.73 m/sec. For the ground release scenario, the 5 percent X/Q resulted from a Pasquill stability Class F (stable) and a wind speed of 1 m/sec.

Security Building*	0-8 Hr	8-24 Hr	1-4 Day	4-30 Day
Mixed Mode Release X/Q (sec/m³)	8.58×10^{-5}	5.32×10^{-5}	1.89×10^{-5}	3.43×10^{-6}
Ground Release X/Q (sec/m³)	1.02×10^{-3}	6.32×10^{-4}	2.24×10^{-4}	4.08×10^{-5}

*Distance 107 m from source to receptor

ATTACHMENT 2

SECURITY BUILDING

TSC INTEGRATED DOSE CALCULATION

Regulatory Guide 1.3 identifies the technique that is to be utilized to evaluate the integrated dose. The TSC integrated dose analysis was done based on a LOCA release from the primary containment at a rate of .5 percent volume per day, 10 gph ECCS leakage into the secondary containment, and MSIV leakage corresponding to a Technical Specification value of 11.5 scfh per valve. All releases are discharged via the RBSVS System.

The thyroid doses are computed using the conversion factors given in TID 14844 and a breathing rate of $3.47x10^{-4}$ m^3/sec (1.25 m^3/hr). The gamma doses are computed based on a finite cloud model in the TSC plus a semi-infinite cloud surrounding the building which has an equivalent 4 inch concrete structure. The beta doses are based on the simi-infinite cloud model suggested by the NRC, Regulatory Guide 1.3.

The total 30-day integrated LOCA doses from the airborne activity in the TSC plus gamma penetrating the building are indicated below. The doses are calculated based on mixed-mode releases with atmospheric dispersion factors (X/Q's) as described in Attachment 1 and providing a HEPA-charcoal HVAC system as delineated in Section 2.4.2 of the TSC Design Criteria and Description.

	Total 30-Day Integrated Dose (Rem)		
	Thyroid	Gamma	Beta
Mixed-Mode Release			
95% Halogen Filter	16.	1.73	3.70
No filter	319.	1.74	3.75

ATTACHMENT 3

TSC LOG AND HISTORICAL DATA FILE
IN-PLANT SYSTEM PARAMETERS

1.0 Core Parameters

1. Control Rod Postion (Core map graphic display)
2. Neutron Flux Levels (APRM, TIP)

1.1 Reactor Coolant System Parameters*

1. Reactor pressure
2. Reactor water level
3. Safety and relief valve position

1.2 Power Conversion System Parameters

1. Feedwater flow
2. Feedwater temperature
3. Condensate storage tank level
4. Main condenser pressure
5. Circulating water pumps disch. pressure

1.3 Safety System Parameters*

1. RCIC pump disch. flow
2. RHR system flow
3. RHR HX inlet/outlet temperatures
4. HPCI pump disch. flow
5. Core spray system flow
6. RHR HX - RHR SW outlet temperature
7. RHR HX - RHR SW flow
8. RBCLCW HX outlet temperatures
9. RB flood level

1.4 Containment Parameters*

1. Drywell pressure
2. Drywell temperature
3. Suppression chamber pressure
4. Suppression pool water temperature
5. Suppression pool water level
6. Drywell hydrogen conc.
7. Suppression hydrogen conc.
8. Drywell oxygen conc.
9. Suppression chamber oxygen conc.
10. Reactor Bldg. pressure

1.5 Service Air

1. ADS air header pressure

* Redundant signals in these categories will be provided in this log.

ATTACHMENT 4

TSC LOG AND HISTORICAL DATA FILE
RADIOLOGICAL/METEOROLOGICAL PARAMETERS

Meteorological Parameters

1. Wind direction
2. Wind speed
3. Temperature 10 meters elevation
4. Vertical temp difference between 10 meters and upper levels

Radiological Parameters

1. Main steam line radiation level
2. Containment area radiation high range
3. RHR service water discharge radioactivity
4. RBCLCW system radioactivity level
5. Control room ventilation activity level
6. Radiation levels in essential equipment areas
7. Release paths activity (Station vent exhaust and RBSVS)
8. Gaseous effluent flow rates

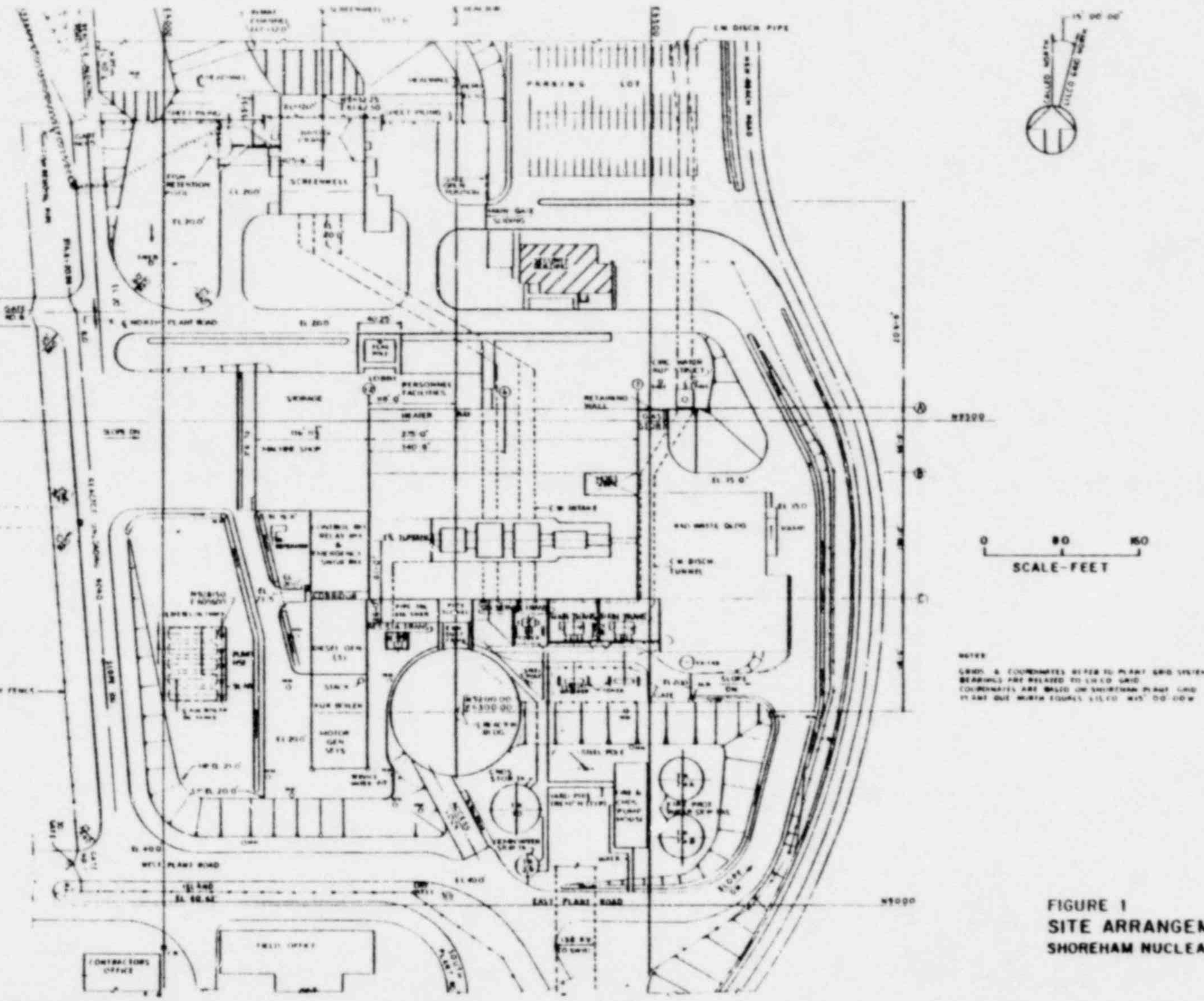


FIGURE 1
SITE ARRANGEMENT PLAN
SHOREHAM NUCLEAR POWER STATION-UNIT 1

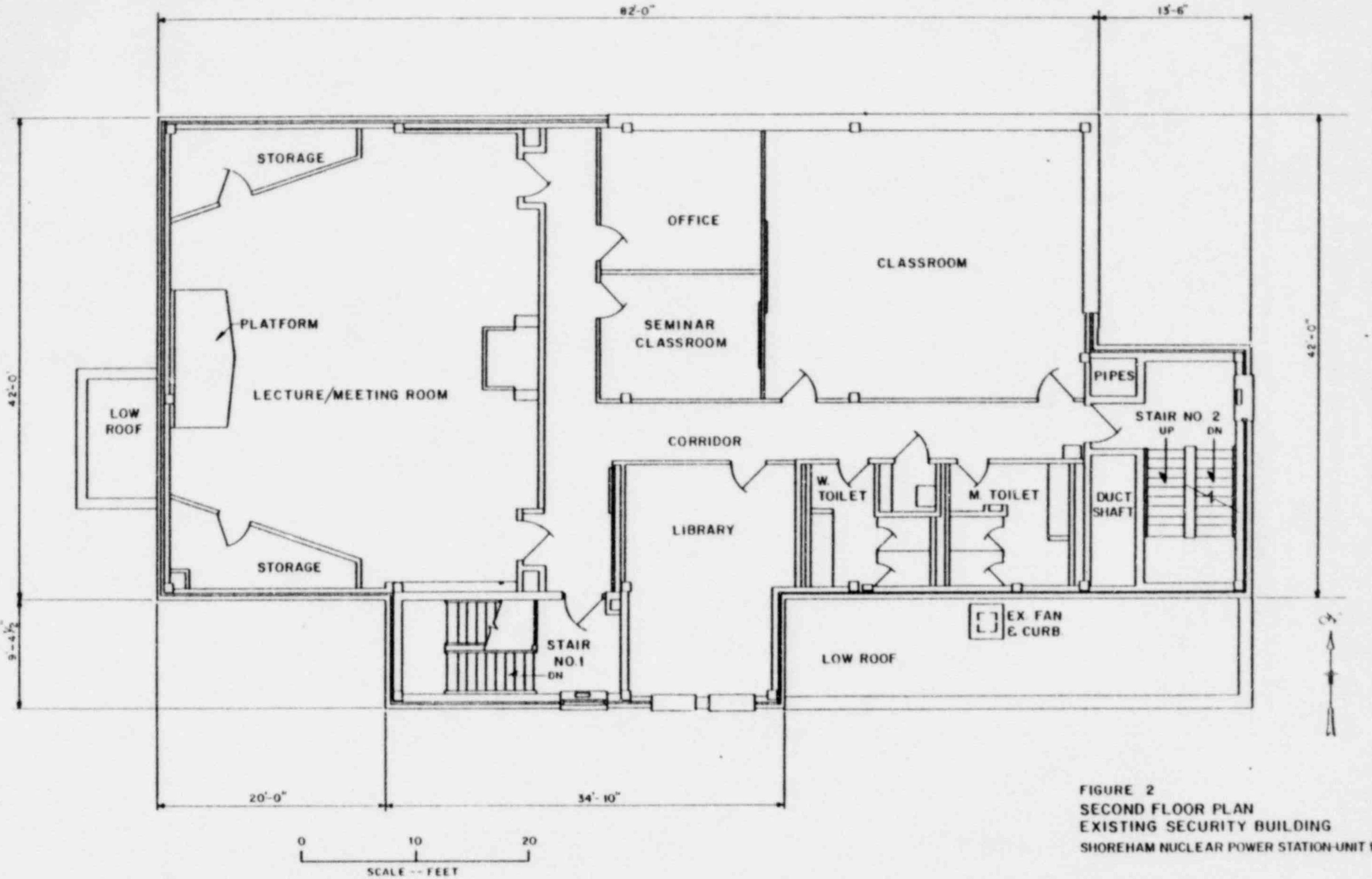
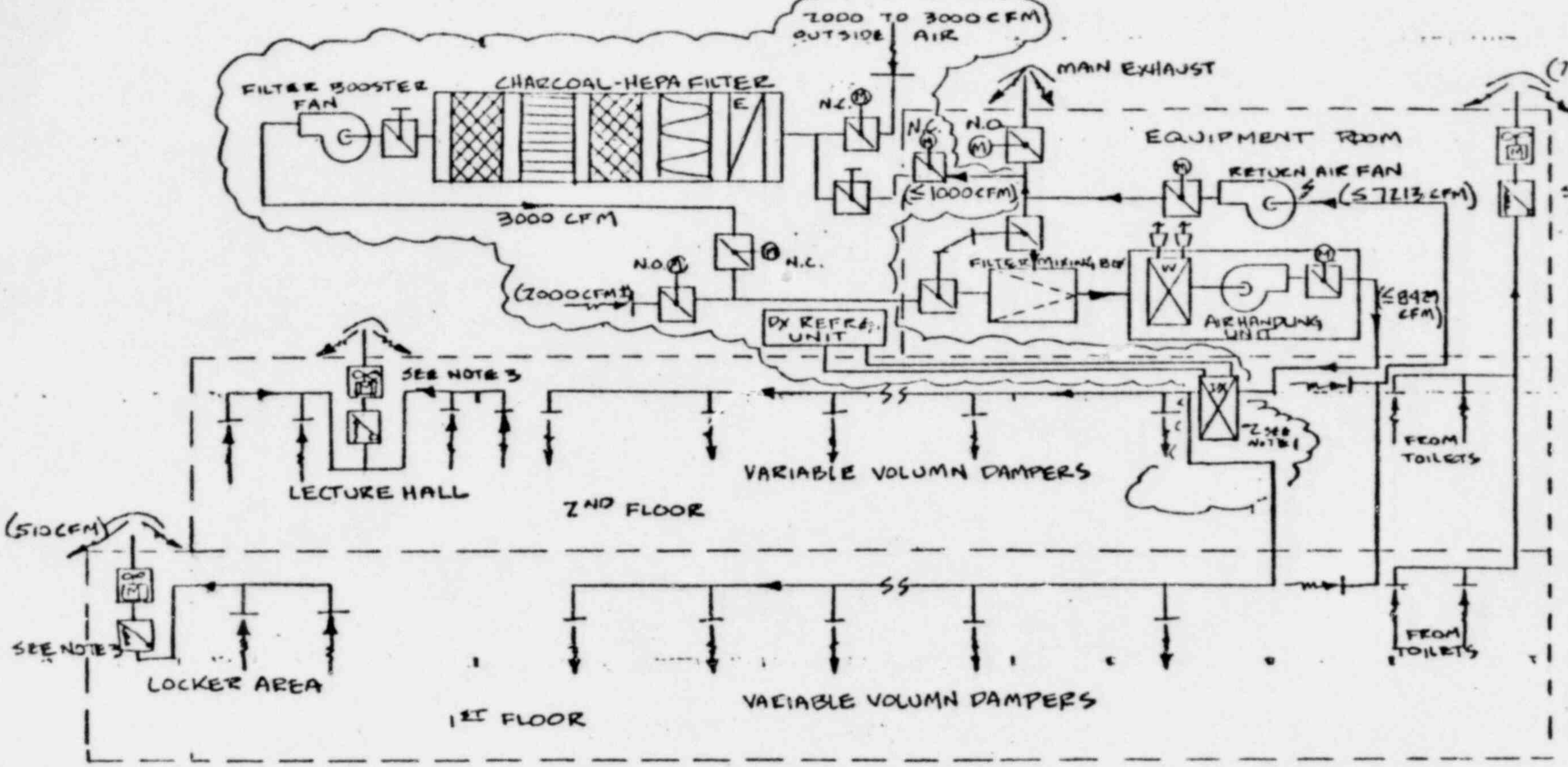


FIGURE 2
SECOND FLOOR PLAN
EXISTING SECURITY BUILDING
SHOREHAM NUCLEAR POWER STATION-UNIT 1



NOTES: 1. ADDITIONAL COOLING CAPACITY FOR TSC HEAT GAIN CAN BE PROVIDED BUT MAY NOT IN FACT BE NEEDED.

2. ALL EQUIPMENT IN (cloud) TO BE ADDED

3. ROOF EXHAUST DAMPERS MAY REQUIRE UPGRADING TO REDUCE LEAKAGE

4. FREE VOLUMN OF SECURITY BUILDING = 105,000
1 AC PER HOUR = 1750 CFM

FIGURE 3
HVAC SCHEMATIC DIAGRAM
TECHNICAL SUPPORT CENTER
SHOREHAM NUCLEAR POWER STATION-UNIT 1

2.2.2.c Onsite Operational Support Center

NUREG 0578 POSITION:

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the lines of communications and management.

NRC CLARIFICATION:

No clarification provided.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group agrees with the position as stated, with the clarification that there may be plant unique situations where it may be more appropriate that more than one location be designated in the emergency plan. As long as these locations are known and the "methods and lines of communication and management" are specified in the emergency plan, the intent of the position will have been met.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

The Staff's position will be implemented as stated and subject to the clarification on location stated above.

LILCO'S RESPONSE:

LILCO endorses the BWR Owners' Group position and will implement the Staff position as stated above prior to fuel load.

2.2.3 Revised Limiting Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability

NUREG 0578 POSITION:

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the necessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4 and 5 operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours.

2. Within 24 hours, bring the plant to cold shutdown.

3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established.

4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function.

5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office. or his designee.

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of steps 3 and 4, above, along with a basis for allowing the plant to return to power operation. The senior corporate executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power.

NRC CLARIFICATION:

No clarification provided.

BWR OWNERS' GROUP DISCUSSION:

None

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

None

LILCO'S RESPONSE:

In accordance with NRC letter from D. B. Vassallo to all pending operating license applicants, dated September 27, 1979, the proposed rule making on limited condition for operation of nuclear power plants has been delayed and no action is required at this time.

Containment Pressure Indication

POSITION:

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

The containment pressure measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy and testability.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group concurs with the ACRS recommendations for additional instrumentation for containment pressure monitoring.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. The Owners' Group intends to implement containment pressure monitoring which will be designed and installed to meet Engineered Safety System criteria.

LILCO'S RESPONSE:

Currently installed instrumentation provides continuous indication of containment pressure in the control room. The existing pressure transmitters and associated instrumentation will be replaced in order to provide the capability to measure three times the design pressure of the primary containment. The range of the pressure instrumentation will be from -5 to +150 psig. The components provided will meet the design criteria outlined in the proposed Revision 2 to Regulatory Guide 1.97 to the maximum extent possible.

Containment Hydrogen Monitors

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.

The containment hydrogen concentration measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group concurs with the ACRS recommendations for additional instrumentation for containment hydrogen monitoring.

It is the Owners' Group's current interpretation that the hydrogen monitoring requirement is associated with ECCS performance and core degradation, rather than with containment atmosphere control.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. The BWR Owners' Group intends to implement containment hydrogen monitoring which will be designed and installed to meet Engineered Safety System criteria.

LILCO'S RESPONSE:

The hydrogen concentration in the primary containment atmosphere will be continuously monitored by the hydrogen analysis system. This system consists of two redundant sub-systems, each including two hydrogen analyzers to sample the drywell and the suppression chamber atmospheres. Refer to FSAR Figure 6.2.5-1 included with Section 2.1.5.a and footnote on page 2.5.1.a-1. Each analyzer is provided with dedicated instrument penetrations to ensure continuous monitoring. The range of the analyzer will be from 0 to 10 percent hydrogen concentration by volume over a pressure range of -2 to +60 psig. Monitoring units will be qualified for the environment expected during normal and accident conditions. A dual recorder is currently installed in the main control room for each subsystem. These recorders are seismically qualified in accordance with IEEE-344-1971, QA Category I and in conformance with IEEE-323-1971. The hydrogen analysis system is powered from redundant emer-

Containment Water Level Indication

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. Also for PWRs, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 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 narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

BWR OWNERS' GROUP DISCUSSION:

The Owners' Group concurs with the ACRS recommendations for additional instrumentation for containment water level monitoring.

For practical reasons, it is not desirable to monitor suppression pool water level all the way to the bottom of the suppression pool. This is because an instrument tap at the very bottom could become obstructed by sludge and small debris. The Owners' Group believes that water level monitoring down to the elevation of the lowest ECCS pump suction is more practical and fully satisfies the intent of the requirement.

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. The BWR Owners' Group intends to implement containment water level monitoring which will be designed and installed to meet Engineered Safety System criteria.

2. The lowest suppression pool water level monitored will be at or below the elevation of the lowest ECCS pump suction.

LILCO'S RESPONSE:

LILCO concurs with the Owners' Group position. Accordingly, taps to measure water level to the bottom of the suppression pool will not be provided. For Shoreham the lower limit will remain unchanged 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 feet above the normal water level, the currently installed instrument taps will be relocated in order to increase the upper limit from 26'6" to 31'6".

Installation of Remotely Operated High Point Vents in the Reactor Coolant System

NRC POSITION[(1)]:

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform

Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

BWR OWNERS' GROUP DISCUSSION:

Domestic BWRs are provided with a number of power operated safety grade relief valves which can be manually operated from the control room to vent the reactor pressure vessel. The point of con-

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(1) Enclosure 4 to NRC letter, dated October 10, 1979, from D. B. Vassallo to all Licensees of Plants under Construction.

nection of the vent lines from the vessel to these valves is such that accumulation of gases above that point in the vessel will not affect natural accumulation of gases of the reactor core.

These power operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves has been provided to the individual plant Safety Analyses Reports.

The Owners' position is that the requirement of single failure criteria for prevention of inadvertent actuation of these valves, and the requirement (stated in the October 11 topical meeting) that power be removed during normal operation, are not applicable to BWRs. These valves serve an important function in mitigating the effects of transients and in many plants provide ASME code overpressure protection. Therefore, the addition of a second "block" valve to the vent lines could result in a less safe design and in some cases a violation of the code. Also, inadvertent opening of relief valve in a BWR is a design basis event and is a controllable transient. (This is discussed in our position of NUREG-0578, item 2.1.2.)

In addition to the power-operated relief valves, operating BWRs include various other means of high-point venting. Information on which plants are equipped with which features has been provided in individual plant Safety Analysis Reports, and may be summarized by individual licensees in their NUREG-0578 implementation letters. Among these are:

1. Normally closed reactor vessel head vent valves, operable from the control room, which discharge to the drywell;

2. Normally open reactor head vent line, which discharges to a main steam line;

3. Main steam-driven Reactor Core Isolation Cooling (RCIC) System turbines, operable from the control room, which exhaust to the suppression pool;

4. Main steam-driven High Pressure Coolant Injection (HPCI) System turbines, operable from the control room, which exhaust to the suppression pool;

5. Isolation condenser primary side vent valves, operable from the control room, which discharge to containment or a main steam line.

Although the power-operated relief valves fully satisfy the intent of the requirement, these other means also provide protection against the accumulation of noncondensables in the reactor pressure vessel.

In the October 11, 1979, topical meeting on this subject, three procedural questions were raised:

1. Where to vent to (suppression pool vs. containment);

2. When to vent;

3. When not to vent.

Under most circumstances, there would be no choice as to where to vent to or when to vent, since the relief valves (as part of the Automatic Depressurization System), HPCI and RCIC will function automatically in their designed modes to ensure adequate core cooling, and these will provide continuous venting to the suppression pool. The current assessment is that it would not be desirable to interfere with emergency core cooling functions in order to prevent venting, but the matter will be studied further.

The result of a break in the safety/relief valve discharge line, or any of the other systems enumerated above, would be the same as a small steam line break. A complete steam line break is part of the plants' design basis, and smaller-size breaks have been shown to be of lesser severity. A number of reactor system blowdowns due to stuck-open relief valves (also equivalent to a small steam line break) have confirmed this in practice (see Owners' Group position on Requirement 2.1.2). Thus no new analyses to show conformance with 10 CFR 50.46 are required.

Because the relief valves, HPCI and RCIC will vent the reactor continuously, and because containment hydrogen calculations in normal safety analysis calculations assume continuous venting, no special analyses are required to demonstrate "that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment".

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA:

1. The Owners' Group believes that adequate reactor coolant system venting is provided by the existing plant design.

2. Plant procedures will be provided to govern the operator's use of the relief valves for venting the reactor pressure vessel.

3. No new 10 CFR 50.46 conformance calculations or containment combustible gas concentration calculations are required, since systems in the plant's original design and covered by the original design bases are used.

4. In response to a request from the October 11, 1979 topical meeting, the use of isolation condenser tube side vents will be considered.

5. In response to a request from the October 11, 1979 topical meeting, the effect of noncondensables in HPCI/RCIC turbine steam will be addressed.

LILCO's RESPONSE:

LILCO endorses the BWR Owners' Group position. Presented below is a discussion[1] of the features provided for Shoreham, which provide protection against the accumulation of noncondensables in the reactor pressure vessel.

1. Safety Relief Valves

   The Shoreham facility is provided with eleven power operated safety/relief valves (S/RVs), which can be manually operated from the control room to depressurize (vent) the reactor pressure vessel (RPV). Seven of the eleven S/RVs comprise the automatic depressurization system (ADS) and are automatically actuated under certain conditions as described in Chapter 15 of the Final Safety Analysis Report. The S/RVs are connected to the four main steam lines which in turn are connected to the RPV above the fuel. Each S/RV discharge is piped to a quencher discharge device located at the bottom of the suppression pool. Position indication is provided in the main control room for each S/RV. The S/RVs, Steam Line and ADS are safety grade and conform with Appendix A to 10 CFR 50 General Design Criteria including the single failure criterion and the requirements of IEEE 279, as applicable.

2. Reactor Core Isolation Cooling (RCIC) and High Pressure Core Injection (HPCI) Systems

   The RCIC and HPCI, installed at Shoreham, are provided with steam turbine driven pumps. The RCIC and HPCI turbines are supplied with steam from the RPV through the main steam lines. The exhaust steam from these turbines is discharged to the suppression pool. The equipment required for initiation of the RCIC and HPCI are completely independent of auxiliary A-C power; they require D-C power, derived from the station battery. These systems are automatically started upon a RPV low water level signal. Controls are provided for remote manual operation from the main control room.

[1] This discussion includes the plant specific information required in Attachment 1E to the conference report of the BWR Owners'/NRC topic meeting on NUREG 0578 implementation, held on October 11, 1979.

3. Normally Open Reactor Head Vent Line

A normally open reactor head vent line is provided in the Shoreham design. This line discharges to one of the main steam lines and vents the portion of the RPV above the main steam nozzles. The head vent line is provided with a safety-related motor operated valve powered by an emergency bus and operable from the main control room. This line conforms to the same design requirements as the reactor coolant pressure boundary.

We consider that the power operated S/RVs, as described in one (1) above, fully satisfy the intent of the reactor coolant system venting requirement. The alternative path of venting the RPV described in two (2) and three (3) above, however, provide additionally installed protection against the accumulation of noncondensable gasses in the reactor pressure vessel.

Procedures for the proper operation of the S/RVs, RCIC and HPCI will specify operator action to vent the RPV.