UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 (OL)
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

NRC STAFF'S PROPOSED OPINION, FINDINGS OF FACT, AND CONCLUSIONS OF LAW IN THE FORM OF A PARTIAL INITIAL DECISION

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VOLUME ONE OF TWO

TABLE OF CONTENTS

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PARTIAL INITIAL DECISION

I. BACKGROUND AND SCOPE OF DECISION

This is a partial initial decision on an application from the Long Island Lighting Company (LILCO or Applicant) to operate a nuclear power plant. The application is for the operation of one boiling water reactor (BWR) at the Applicant's Shoreham site in Suffolk County, New York. A permit to construct the plant, which has a rated output of 820 megawatts of electric power, was issued on April 12, 1973, and the notice of an opportunity for a hearing on the operating license application was published on March 18, 1976.

In addition to LILCO and the Nuclear Regulatory Commission Staff (NRC staff or Staff), the parties to this proceeding are Suffolk County (SC or County), the Shoreham Opponents Coalition (SOC), the North Shore Committee Against Thermal and Nuclear Pollution (North Shore Committee or NSC), the Oil Heat Institute of Long Island, Inc. (OHILI), and the New York State Energy Office (SEO). On the issues addressed in this partial initial decision, the active participants were LILCO, the NRC staff, and Suffolk County.

This partial initial decision covers the following issues:

Contention Number	Subject
SC/SOC 7B; SOC 19(b)	Systems, Structures, and Components Important to Safety: Classifiction and Systems Interaction
SC 4	Water Hammer
SC 10	ECCS Core Spray
SC 11	Passive Mechanical Valve Failure
SC 16	Anticipated Transients Without Scram
SOC 19(e)	Seismic Design
SC 21	Mark II Containment
SC 22;SC 28(a)(vi)/ SOC 7A(6)	Safety Relief Valve Tests and Challenges
SC 27/SOC 3	Post-Accident Monitoring

Also covered by this partial initial decision are the Board's determinations regarding the impact on Shoreham of certain generic unresolved safety issues (USI's). Proposed findings of fact and conclusions of law concerning USI's were served by the Staff on November 2, 1982.[1/]

Our decision on the remaining issues -- principally quality assurance, environmental qualification, and offsite emergency planning -- will come at a later date.

1/ The Staff's proposed findings of fact and conclusions of law concerning USI's included USI Nos. A-8/A-39, A-10, A-11, A-31, A-36, A-40, A-42, A-43, A-44, A-45, and A-48. USI Nos. A-1, A-9, A-17 and A-47 are subsumed by contentions addressed in this partial initial decision.

II. OPINION

A. Water Hammer (SC 4)

Water hammer is a potential problem that can occur in any electric generating facility. It is a single shock or series of shocks produced by sudden changes in the flow conditions of fluids in pipes that can cause damage to pipes and equipment. Finding 4:3. Water hammer has been designated as an Unresolved Generic Safety Issue. It was accorded this status as a result of a relatively large number of past water hammer events. These events were made the subject of NUREG-0582 which listed this water hammer data and made certain recommendations as to corrective action. After its subsequent designation as an unresolved Generic Safety Issue, the Staff has initiated studies to develop recommended actions to eliminate it or minimize its effects. If the outcome of these studies shows a need for additional design changes or operating conditions not already incorporated into the design of Shoreham, the Staff will require implementation of such additional requirements. Finding 4:4.

Where unresolved generic safety issues are involved in an operating license proceeding, for an application to succeed there must be some explanation why operation can proceed even though an overall solution has not been found. Virginia Electric Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245, 248 (1978). A plant will be allowed to operate pending resolution of these unresolved issues when there is "reasonable assurance" that the facility can be operated without undue risk to the health and safety of the public. Gulf States Utility Co.

(River Bend, Units 1 and 2), ALAB-444, 6 NRC 760, 774 (1977). See also Pacific Gas and Electric Co. (Diablo Canyon Units 1 and 2), LBP-81-21, 14 NRC 107, 118 at 35 (1981). A basis for allowing a plant to operate could exist in a number of ways. See Gulf States Utilities Co., 6 NRC at 775; Virginia Electric Power Co., supra, 8 NRC at 248; and Pacific Gas and Electric Co., supra, 14 NRC at 113 #35.

The record in this proceeding establishes that no undue risk to public health and safety will be caused by the operation of Shoreham pending the generic resolution of the water hammer issue. To achieve this interim goal, measures were taken to eliminate water hammer at Shoreham in the areas of design, stress analysis, operating procedures, training procedures, and testing. Findings 4:5 - 4:13. Most of the measures that are expected to be included in Staff's final generic resolution of water hammer have already been incorporated at Shoreham. These measures are presently the subject of reports by the Quadrex Corporation and the EG&G Corporation who were commissioned by the Staff to report on this issue and make needed recommendations. It is Staff's opinion that the major recommendations which have already been made in a draft Quadrex report which has been issued, will not be changed. Finding 4:14.

Intervenor Suffolk County's concern with water hammer in this proceeding is that preoperational or startup data from other plants has not been considered at Shoreham. Contrary to this assertion, both the General Electric Company (the nuclear reactor system supplier) and Stone & Webster (the architect-engineer) had programs that monitored water hammer events at other nuclear facilities which would have assured that appropriate corrective action would have been taken at Shoreham.

Finding 4:17. The Staff's generic study regarding water hammer assured this same result since this study was based upon past experience at other facilities and the knowledge thus acquired was then applied to the design and operating provisions at Shoreham. Finding 4:18.

As part of his testimony in this proceeding, Intervenor's consultant, Mr. Marc Goldsmith, listed several nuclear facilities that he alleged had water hammer events which were not evaluated at Shoreham. One such plant was a European BWR Mark II that had experienced serious water hammer damage on RHR startup operations. Contrary to Mr. Goldsmith's assertion, that event was not applicable to Shoreham since Shoreham's systems would minimize the type of phenomenon that occurred at the European reactor. In addition, because the causes for the European BWR event were the same as the causes for a number of similar events that had occurred in the United States, there was no need to emphasize that event. Finding 4:19. The other plant Mr. Goldsmith specifically referred to was Commonwealth Edison's LaSalle nuclear unit where a pipe vibration monitoring program had been conducted which the Intervenor believed should be utilized at Shoreham. The evidence of record reveals, however, that the use of the LaSalle monitoring program at Shoreham would have been unnecessary since Shoreham already had a similar monitoring program. Finding 4:20.

To also show that water hammer experience at other nuclear facilities was not utilized at Shoreham, Intervenor Suffolk County cross-examined Staff and Applicant witnesses concerning various water hammer events that were listed in the EG&G Report. However, the testimony revealed that the Applicant or Staff either had taken those experiences into account or there was no need to do so. Finding 4:21.

Based upon the evidence of record that water hammer experiences at other similar BWR facilities would have been taken into account at Shoreham, the Intervenor's water hammer concerns are without merit.

B. ECCS Core Spray (SC 10)

SC Contention 10 asserts that the Shoreham Emergency Core Cooling System (ECCS) has not been demonstrated to meet the requirements of 10 C.F.R. 50.46 and Part 50, Appendix K. The basis for this contention is Japanese test data reported in Board Notification 81-49 concerning maldistribution of ECCS core spray. Finding 10:1. For the purposes of this contention the parties agreed to stipulate that this maldistribution of the ECCS core spray occurred. Specifically we are assuming no direct core spray distribution to the central region of the reactor core. Finding 10:2. This assumption does not affect the conclusion that the Shoreham ECCS does in fact comply with the requirements of 10 C.F.R. 50.46 and Appendix K.

Appendix K specifies a minimum heat transfer coefficient for core spray cooling for reactors with seven by seven fuel assembly arrays. The Appendix K heat transfer coefficient value has been used in the General Electric ECCS Evaluation Model used in the analysis for Shoreham. Finding 10:4. The value has been demonstrated to be conservative and acceptable for use in reactors, such as Shoreham, with eight by eight fuel arrays. Finding 10:5.

Presuming that there is no direct core spray distribution to the central bundles of the reactor core, there are four other cooling phenomena that would take place in the core and would justify the use of the Appendix K heat transfer coefficient. The first cooling mechanism results from counter-current flow limitation (CCFL) at the top of the fuel bundles. Finding 10:7. CCFL is caused by the uprush of steam through the reactor core. The steam imparts a force on water which is flowing down through the core, thereby limiting the flow of water. As a result, a pool of water accumulates at the top of the core. This pool spreads out across the top of the core and makes an even direct core spray distribution unnecessary. Finding 10:8. Water flows down through all the bundles from this pool, because of the dynamic equillibrium created between the downward gravitational force on the water and the upward force exerted by the steam. Finding 10:7.

The indirect flow through the bundles has been shown by General Electric tests to be in the range of 2 to 4 gallons per minute (gpm). The Appendix K heat transfer coefficient can be acheived by a minimum flow to each bundle of only one gallon per minute. Finding 10:8. Therefore, the flow from the pool at the top of the core alone is adequate to justify the use of the Appendix K coefficient in the Shoreham ECCS evaluation.

Although the flow from the accumulated pool is sufficient to assure adequate core cooling, the three other cooling phenomena also provide additional margin. These further cooling mechanisms are not taken credit for in the GE ECCS Evaluation Model. Finding 10:11.

At the same time as water is accumulating at the top of the core and flowing at a rate of 2 to 4 gpm down through the central bundles, a much greater quantity of coolant is flowing down through quenched peripheral bundles. This flow has been shown to reach approximately 100 gpm. This water increases the reflood rate at the bottom of the core. Finding 10:9. The increased reflood alone, with no flow from the top through the central bundles, would provide sufficient cooling capacity to assure that the peak clad temperature required by 10 C.F.R. 50.46 is not exceeded. Finding 10:10.

The third cooling mechanism in the Shoreham core is provided by the steam rushing up through all the bundles of the core. The steam alone may also provide a heat transfer coefficient greater than the Appendix K value. Finding 10:12.

The final cooling effect testified to by the witnesses is caused by the CCFL phenomenon at the bottom of the core. The steam updraft from the lower plenum of the reactor into the bundles will hold cooling water up in the bundles, thereby providing the additional cooling. Finding 10:13.

In conclusion, the Board is able to find that an assumption of no direct core spray distribution to the central fuel bundles does not lead to a conclusion that there will be no cooling in those bundles. Direct core spray distribution is simply not a dispositive factor. Multiple cooling effects are present to assure that the Shoreham ECCS will comply with the requirements of 10 C.F.R. 50.46 and Part 50, Appendix K.

C. Passive Mechanical Valve Failure (SC 11)

SC Contention 11 asserts that there is a possibility that the valves used in the Shoreham safety-related systems could fail in an undetectable or unsafe mode. Finding 11:1.

While there existed some initial confusion over the definition of the term "passive mechanical valve failure" and whether this referred to types of components or types of failures, it was generally established on the record that this term refers to the mechanical failure of an active valve which may remain undetected until the system is called upon to operate. Finding 11:3.

Valves utilized in the Shoreham safety-related systems are in accordance with the approved ASME codes and standards and are the standard valves used throughout the industry, chosen for their high performance reliability. Finding 11:4. Nevertheless, in spite of strict adherence to these codes and standards, it cannot be guaranteed that a valve will never fail. Finding 11:13.

However, the Shoreham reactor coolant pressure boundary is designed to accommodate such a possibility. This is accomplished through the complete redundancy of all valves in the system, with the single exception of the Safety Relief Valves (SRV's). Findings 11:4, 11:10.

The Shoreham safety-related systems are required to be designed against the single failure criterion as outlined in 10 C.F.R. 50, Appendix A. The single failure criterion is considered to be met in fluid and electric systems if neither a single failure of any active component (assuming passive components function properly) nor a single failure of a passive component (assuming active components function properly) results in the system losing its capability to perform its safety function. However, while the failure of passive components should be assumed in designing against a single failure in electric systems, whether this should also be applied to fluid systems has not yet been determined. Finding 11:7.

The analysis performed by the Applicant considered the failure of both active and passive components. Whether the valve failure was active or passive, the consequences of that failure would not be greater than those contained in the analysis. The assumed failure of the entire subsystem in which the valve is located was included in this analysis. Finding 11:8.

The Applicant has submitted a draft valve test plan to provide for the in-service inspection and testing of valves in the Shoreham safety-related systems which is still under review by the Staff. Testimony on the record, however, provides this Board reasonable assurance that this testing plan is in accordance with the ASME Boiler and Pressure Vessel Code, Section XI regarding the frequency of testing. Any relief requests from the testing frequency suggested in the ASME Code will be carefully reviewed by the Staff to assure that longer intervals for testing frequency would not be unsafe and would still meet Staff requirements. Finding 11:9.

The ability to detect valve failures at Shoreham is further assured by the use of monitoring devices, including valve position indicators (on motor operated valves), limit switches (on air operated valves), downstream discharge indicators (on SRV's), and lights indicating electric circuitry (on solenoid valves). Further, the redundancy of valves (excluding the SRV's), provides reasonable assurance that an undetected valve failure, if such should occur, will not cause the system to lose its capability to perform its safety function. Finding 11:10.

Industry operating experience is reviewed by Shoreham's Independent Safety Evaluation Group (ISEG) to assess its applicability to Shoreham systems and oversee the implementation of any modifications of components that may be deemed necessary. Sources of this industry operating experience include notices from manufacturers, other plants, I&E Bulletins and Licensee Event Reports (LER's) issued by the NRC. Finding 11:11. This Board finds the Shoreham program to monitor industry experience adequate and reliable.

The Board also notes that specific concern was raised by the County over the use of the Rockwell-Edward Main Steam Line Isolation Valve (MSLIV) at Shoreham. This valve has experienced undetected failures in the past at two operating plants. These failures involved the separation of the valve stem from the disc. This failure of the MSLIV prompted the manufacturer to analyze the component and subsequently recommend certain modifications of the valve to resolve the problem. The modifications recommended have been ordered at Shoreham and will be installed in place of the existing MSLIV's. Finding 11:12. The record supports a finding that the modifications to the valve satisfactorily resolve the problem with this particular valve, and the record discloses no other specific valve problems.

Hence, this Board finds with regard to this contention, that for Shoreham adequate precautions have been taken to detect valve failures in the system design and analysis, coupled with an in-service inspection and testing program and monitoring devices for the valves. Further, if such a failure did occur, its effects are substantially mitigated by the redundancy in the system which meets the criteria set forth in 10 C.F.R. Part 50, Appendix A.

D. Anticipated Transients Without Scram (SC 16)

SC Contention 16 questions the adequacy of measures taken at the Shoreham plant to reduce the risk of anticipated transients without scram

(ATWS). Finding 16:1. An ATWS is an event in which the reactor trip (scram) system is postulated to fail to operate as required. This subject is currently the subject of a Commission rulemaking proceeding. Shoreham will be required to make any modifications that result from the rulemaking. Finding 16:3. However, SC Contention 16 is specifically addressed to the adequacy of the ATWS measures taken at Shoreham in the interim period of several years prior to the completion of the rulemaking and the implementation of the result at Shoreham. Suffolk County contends that these measures are inadequate and that Shoreham will therefore not meet the requirements of 10 C.F.R. Part 50, Appendix A, GDC 20.

The Applicant and Staff presented a combined witness panel on this contention. Suffolk County presented no witnesses or testimony to support its contention. Finding 16:2. Based on the record before us, this Board is able to conclude that the interim ATWS measures being taken at Shoreham are in keeping with the Commission's position on interim operation, and that GDC 20 will be satisfied for the period prior to the implementation of the final ATWS rule.

The Commission has stated that Shoreham and other plants should be permitted to operate prior to the generic resolution of the ATWS issue. This conclusion was based on such considerations as the estimated low probability of anticipated transients with potentially severe consequences in the event of a scram failure, and favorable operating experience with current scram systems. Finding 16:4. Within the Commission's sphere of knowledge was the fact that the scram systems are highly redundant and highly tolerant of component failures. Finding 16:5. Also, a standby liquid control system (SLCS) is available to inject liquid boron into the reactor to achieve safe shutdown of the plant in the event of a failure to scram. Finding 16:7. At Shoreham an added measure is taken to provide even further redundancy to the scram system. The alternate rod insertion system (ARI) is designed to automatically insert control rods following a normal trip signal and a failure to scram. This system is a diverse means to further reduce ATWS challenges. Finding 16:6.

On top of these considerations the Commission has required that additional interim ATWS measures be taken in order to further reduce the risk of ATWS events during the period prior to a final ATWS rule. These interim measures, which are being taken at Shoreham, are: (1) the installation of a recirculation pump trip (RPT); (2) the development and implementation of an ATWS operator procedure; and (3) training of operators for ATWS events. Finding 16:4. The adequacy of these interim measures, as implemented at Shoreham, is the focus of this contention.

The RPT is an automatic system designed to reduce reactor power on a high reactor pressure signal or a low water level signal. The pump trip provides a reduction in reactor power to less than 40% in less than one minute. This system provides overpressure protection at the beginning of an ATWS event. Finding 16:9. The Board finds no indication that the RPT as installed at Shoreham does not provide the additional measure of ATWS protection intended during the interim period.

The Board also finds that the Shoreham operators are being actively trained to follow an ATWS emergency operating procedure. The training includes classroom instruction and simulator excercises. The NRC's regional inspection program has monitored the training at Shoreham throughout. Moreover, as a part of the training program each operator candidate takes the NRC operator exam, which includes testing on the ATWS operating procedure. Finding 16:10.

The specific Shoreham emergency operating procedure for the ATWS scenario was based upon General Electric ATWS emergency procedure guidance, and has been reviewed by the NRC staff. The Staff review included an assessment against Staff acceptance criteria for ATWS procedures, an analysis of human factors implications, and an observation of the procedure on the Limerick simulator. Based on this review the Staff made comments and suggestions, which were then incorporated by the Applicant into a revision of the Shoreham procedure. The Staff has approved the revision. Findings 16:11-16:12.

The ATWS procedure provides the operator with clear step-by-step instructions to follow in the event of a failure to scram. The first steps of the procedure, the "immediate actions", will be memorized by the operator to ensure his rapid response to the event. Finding 16:13. The primary objective at that stage is to manually scram the reactor, because insertion of the rods is the fastest way to decrease reactor power. Finding 16:14. In the next steps, the operator is given explicit criteria on when to abandon the attempt at manual scram, in order to initiate the SLCS. The time into the transient is not the principal parameter for SLCS initiation. Instead the operator will be watching the power level and the suppression pool temperature. When either the power parameter or the pool temperature meets the criteria in the procedure, the operator will not hesitate to inject the boron. This assures that the SLCS will be initiated in a timely manner. Findings 16:15-16:17.

During the testimony on the details of the ATWS procedure, one concern was highlighted by the Board. This related to the location of the SLCS key. Finding 16:18. The witnesses considered the problems involved in keeping the key in a locker with many other keys, and the Applicant agreed to make the necessary changes to alleviate the Board's concerns. Finding 16:19. In sum the Board is able to find no deficiencies in the Shoreham ATWS procedure, as written.

The BWR Owners' Group is currently revising its emergency procedure guidelines. This Revision 2 will include ATWS guidance. The revision has been reviewed by the NRC staff. Finding 16:24. The Shoreham operating procedures will be modified to reflect any changes. Finding 16:25. The procedures at Shoreham will therefore be kept current with the latest refinements. Operator training, however, will not be undermined by modifying the procedures before the refinements are fully developed in the procedures, and approved by the NRC staff.

In issuing its position on plant operation during the period prior to a final ATWS rule, the Commission understood that the interim ATWS measures may not mitigate all ATWS events. Finding 16:26. However, the RPT system, the ATWS emergency operating procedure, and operator training for ATWS, do reduce a low risk even further. The Applicant has satisfactorily implemented these interim measures, and therefore Shoreham will be in compliance with GDC 20 for the period until the generic ATWS resolution. Accordingly, there is reasonable assurance that the plant can be operated in the interim without posing an undue risk to public health and safety. See Gulf States Utility Co. (River Bend, Units 1 and 2), ALAB-444, 6 NRC 760, 774 (1977).

E. Seismic Design (SOC 19(e))

SOC Contention 19(e) alleged two inadequacies in the Shoreham seismic design. Specifically, SOC charged that the Shoreham design failed to comply with 10 C.F.R. Part 50, Appendix A, Criterion 2, and 10 C.F.R. Part 100, Appendix A, because (1) the design response spectra used were not based on the standard in Regulatory Guide 1.60 and (2) a higher damping value than that identified in Regulatory Guide 1.61 was used. Direct testimony on the contention was provided by the Applicant and the NRC staff; the Intervenors relied on cross-examination by counsel for Suffolk County. Findings 19:1-19:2.

First we will examine the damping issue. Damping is a measure of dissipation of the energy associated with an earthquake. Damping figures are utilized to adjust seismic response spectra; as the percentage of critical damping increases, the response spectra reflect smaller ground motion values. Finding 19:3. The gist of Intervenors' complaint in this area is that Regulatory Guide 1.61 allows Applicants to utilize a 4% damping figure, while the Shoreham design utilized a 5% damping figure.

Intervenors' contention reflects a basic misapprehension of damping figures involved; the figures do not refer to the same types of damping. The 4% damping figure used in Regulatory Guide 1.61 addresses only structural damping.[2/] The 5% damping figure used at Shoreham represents a combination of structural and soil damping[3/] referred to as total

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2/ Structural damping (also known as material damping) is a measure of energy dissipation of a structure under dynamic excitation. Finding 19:3.

3/ Soil damping measures the dissipation of energy associated with the interaction of a structure and the surrounding soil. Finding 19:3.

system damping. The evidence indicated that a 10% soil damping figure would be conservative for Shoreham, and that a combination of 4% structural damping and 10% soil damping would result in an acceptable total system damping value for the site in excess of 5%. Findings 19:4-19:5. The use of 5% damping for total system damping at Shoreham was accepted by the NRC staff. Finding 19:6. We find the 5% figure both appropriate and in no way contradictory with the Staff guidance found in Regulatory Guide 1.61. Suffolk County has conceeded such in its proposed findings of fact.

We now turn to the issue of the Shoreham design response spectra. A response spectrum is defined in 10 C.F.R. Part 100, Appendix A, as "a plot of the maximum responses (acceleration, velocity or displacement) of a family of idealized single-degree-of-freedom damped oscillators against natural frequencies (or periods) of the oscillators to a specified vibratory motion input at their supports." A design response spectrum is a smoothed combination of a number of individual response spectra obtained from the time history records of a number of earthquakes which is then used in structural analysis and design. Finding 19:8.

Intervenors contend that the design response spectra for Shoreham are not based on the standards in Regulatory Guide 1.60, and hence have not been demonstrated to be sufficiently conservative to comply with the applicable Commission regulations. Before determining whether the Shoreham spectra meet the regulations, a word is in order about the regulatory effect of NRC staff regulatory guides.

It is settled law that regulatory guides are "not regulations _per se_ and are not entitled to be treated as such; they need not be followed by Applicants; and they do not purport to represent that they set forth

The procedures followed by the Applicant in developing its seismic response spectrum squarely comply with the requirements of Appendix A to 10 C.F.R. Part 100. A modified Housner spectrum was used as the design spectrum after it was shown that the modified Housner spectrum adequately enveloped the spectra of time history records of the vibratory motion caused by four earthquakes (and one artificial time history). This Board has found that the earthquake records selected were proper, that the amplification analysis used to modify the records for the conditions at Shoreham was proper, and that the modified Housner spectrum used for the Shoreham design adequately enveloped the spectra of the time history records developed for the site. The Board therefore finds that Applicant has met the Commission's requirements as they relate to seismic design response spectra.

F. Mark II Containment (SC 21)

Suffolk County Contention 21 asserts that Shoreham's primary containment, reactor pressure vessel supporting structure, and attached and associated safety-related equipment have not been shown to meet the requirements of 10 C.F.R. 50, Appendix A, GDC 4, 16, 50, 51 and 52. Finding 21:1. As originally admitted, Contention 21 consisted of 5 subparts, each setting forth a separate area of concern. Subpart (b), relating to load definitions associated with SRV actuation, was voluntarily withdrawn by Suffolk County. Finding 21:2.

The Board has examined each of the remaining four subparts which make up Contention 21. For each of these subparts we have examined whether the Shoreham equipment is deficient as Suffolk County alleges. If it was determined that the allegation was valid, we have examined:

the only satisfactory method of meeting a specific regulatory requirement." Gulf States Utilities Company (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760, 772-73 (1977); see also Vermont Yankee Nuclear Power Corp. (Vermont Yankee Station), ALAB-179, 7 AEC 159, 174 n. 27 (1974). An application for an operating license must be judged against (and ultimately must comply with) the Commission's Regulations, not the regulatory guides.

The Commission's seismic requirements are contained in Appendix A to 10 C.F.R. Part 100. Section VI of Appendix A establishes that response spectra be developed for both the Safe Shutdown Earthquake and the Operating Basis Earthquake. Section VI further states:

> In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the response spectra be smoothed design spectra developed from a series of response spectra related to the vibratory motions caused by more than one earthquake.

Findings 19:9-19:10.

The Shoreham design response spectra were developed before Regulatory Guide 1.60. Finding 19:12. To develop the design spectra for Shoreham, the Applicant took time history records from four actual earthquakes (El Centro 1940, Taft 1952, Helena 1935, and Golden Gate 1957) and one artificial time history (having properties in between the Taft and El Centro records), and subjected the records to an amplification analysis in order to reflect the soil conditions at the Shoreham site. Findings 19:13-19:14. It was determined that a .2g Housner spectrum adjusted for frequencies below 2 hertz adequately enveloped the response spectra of the time history records modified for the Shoreham site. For this reason, the .2g Housner spectrum adjusted below 2 hertz ("modified Housner spectrum") was used as the design spectrum for Shoreham. Finding 19:15.

the steps taken by the Applicant to address the concern; the Staff review of the allegation and the Applicant's remedial steps; the safety implications, if any; and whether the requirements of 10 C.F.R. 50, Appendix A, GDC 4, 16, 50, 51 and 52 have been met.

Contention 21(a)

Contention 21(a) asserts that several forces generated during the suppression pool LOCA dynamics have not been adequately considered in the design of the Mark II containment. Suffolk County specified three forces in particular -- steam condensation downcomer lateral loads, steam condensation oscillation loads, and steam condensation chugging loads -- as having received inadequate attention. Based on its independent review of the Applicant's submittals, the 4TCO test data, the Karlstein test data and the JAERI test data, however, the Staff concluded in Supplement 1 of the Shoreham Safety Evaluation Report that the Applicant's specifications for assessing the various suppression pool dynamic loads, including the three loads specified in Contention 21(a), were adequately conservative and therefore acceptable. Finding 21:6-21:7. Subsequently, NUREG-0808 was issued, setting forth the final generic load specifications for Mark II containments, developed from the Mark II reassessment program that has been underway for several years. Id. In response to a Staff consultant's report raising the issue of a possible lack of conservatism in the established chugging load specifications, the Staff and the Mark II owners' group reviewed the generic chugging load definition of NUREG-0808, and concluded that no modification to the load specifications was required as these loads were within loads previously analyzed. Findings 21:8-21:10.

Two further issues regarding the Mark II containment design were raised in this proceeding: a concern raised by the ACRS regarding a potential pool bypass from stuck-open wetwell-to-drywell vacuum breakers resulting from intermittent steam condensation; and concerns related to the Mark III containment design raised by a former General Electric employee.

In response to the ACRS concern the Applicant has implemented a design modification by the blocking of the downcomers on which the vacuum breakers are installed. This will eliminate the dynamic pressure exerted on the vacuum breaker and avoid the potential of a stuck-open breaker. Finding 21:12.

Additional concerns regarding the Mark III design were expressed by Mr. John Humphrey. Most of these Mark III concerns were either inapplicable to the Mark II design or were of no significance because the margins inherent in the Mark II design were sufficient to accommodate the quantifiable potential effect of the applicable concerns. Finding 21:13. The single Mark III-related issue which required detailed consideration for the Mark II was whether any safety-related equipment would be disabled by the effects of discharge into the suppression pool when the residual heat removal (RHR) system is operated in the steam condensation mode. It is not possible, however, to operate the RHR discharge system in the steam condensation mode at 5% or less power. The Staff will complete its analysis of this concern prior to full power operation. In the event that the discharge load is found to be excessive, this problem can be solved simply by adding a quencher at the end of the RHR line. Findings 21:13-21:16. For these reasons it is unnecessary to resolve this issue before deciding on the issuance of a low-power license.

It is the opinion of this Board, therefore, based on the evidence on the record discussed above, that in the design of the Mark II containment for Shoreham, the Applicant has adequately considered the forces generated during the suppression pool LOCA dynamics. In addition, the record demonstrates that the design modification implemented by the Applicant has resolved any concerns related to the pressure exerted on the vacuum-breakers. Finally, the record shows that the issue of the effect of RHR discharge on safety-related equipment does not arise at 5% or less power, and hence need not be considered prior to issuance of a low-power operating license.

Contention 21(a) is, therefore, resolved in favor of the Applicant.

Contention 21(c)

Contention 21(c) asserts the inadequacy of the test procedure used by the Applicant to demonstrate an acceptable leakage rate in leakage paths between the drywell and the wetwell. The evidence on the record concerning the test procedure satisfies this Board that the procedure is adequate to assure that excessive steam bypass will not occur. This procedure consisted of pressuring the drywell and measuring the pressure decay inside the drywell over time. The measured pressure rate of decay was compared with an acceptance criterion equal to ten percent of the leakage rate needed for the pressure inside containment to equal the design base pressure following the most limiting LOCA. Steam under accident conditions will condense inside potential leak paths resulting in a slower leakage rate than that observed in the test procedure. Findings 21:16-21:17.

Based on these findings, this Board concludes that the test procedure used by the Applicant to measure the rate of leakage between the drywell and the wetwell is adequate; hence, Contention 21(c) is resolved in the Applicant's favor.

Contention 21(d)

Contention 21(d) asserts that the Mark II containment design has not been shown to be adequate to ensure, with sufficient margin, that it can accommodate combined loads from transients and LOCA events.

NUREG-0808 sets forth the generic load specifications for Mark II containments, including load considerations of combinations of transients and LOCA events. The Applicant has evaluated the containment design against the loads specified in NUREG-0808, and, as a result, has modified the design of steel structures in the plant, with the result that the design now meets the requirements of NUREG-0808. There is no evidence controverting Applicant's compliance with NUREG-0808. Findings 21:20-21:21. The acceptance criteria used in the Applicant's evaluation are in conformance with the requirements delineated in the Staff's Standard Review Plan, Section 3.8. Finding 21:22.

The evidence on the record demonstrates that the Shoreham Mark II containment design has been shown to be capable of accommodating the combined loads from transients and LOCA events, with sufficient design margin to satisfy the general design requirements of 10 C.F.R. 50, Appendix A.

This Board, therefore, resolves Contention 21(d) in Applicant's favor.

Contention 21(e)

Contention 21(e) asserts that an adequate and properly controlled experimental design verification program has not been performed, as required by Appendix B to 10 C.F.R. 50, Sections III and XI. Specifically, Suffolk County asserts that there is insufficient assurance that the acceptance criteria used by Applicant to evaluate the containment design are suitably conservative. Contrary to these assertions, the evidence on the record demonstrates that a suitable verification program has been performed by the Applicant. The structural integrity of the containment structure for Shoreham was tested in accordance with Regulatory Guide 1.18, thereby meeting the requirements of 10 C.F.R. 50, Appendix B, Section XI. Finding 21:25. The NRC staff, based on its independent review and on the 4TCO, Karlstein, and JAERI test data, found the acceptance criteria used by the Applicant to be acceptable. Finding 21:27. The design testing program used during the Mark II assessment was under the supervision of General Electric, whose Quality Assurance program meets the requirements of 10 C.F.R. Part 50, Appendix B. Finding 21:28. By using acceptance criteria in their evaluation that combined LOCA loads with seismic event loads, Applicant's evaluation took into account adequately adverse environmental conditions. Findings 21:29-21:31. Hence, it has been demonstrated to the satisfaction of this Board that the acceptance criteria used by Applicant in evaluating the Mark II containment design was sufficiently conservative, and that the design verification program was adequate. Contention 21(e) is, therefore, resolved in Applicant's favor.

Having resolved all subparts of SC Contention 21 in Applicant's favor, the Board concludes that Shoreham meets the Commission's regulations as they relate to Mark II Containment design.

G. Safety Relief Valves (SC 22; SC 28(a)(vi)/SOC 7A(6))

Shoreham is equipped with eleven Target Rock two-stage type safety relief valves (SRV's). The primary design function of the SRV's is to relieve excess pressure in the reactor vessel by releasing steam from the vessel to the suppression pool. Finding 22/28:6. There are two contentions in this proceeding which concern the Target Rock SRV's.

The first of these contentions, SC 22, questions the sufficiency of the Shoreham SRV testing to meet the requirements of NUREG-0737, Item II.D.1, which was issued to respond to the TMI accident. The Applicant, as a member of the BWR Owners' Group, participated in a generic test program to respond to the TMI Item. The contention asserts that this test program did not include all the necessary test conditions, most notably ATWS conditions. The contention also asserts that a detailed plant specific analysis of the Shoreham SRV's, piping, and supports, is necessary to demonstrate applicability of the generic test program to Shoreham. Finding 22/28:1.

The second contention, SC 28(a)(vi)/SOC 7.A(6), is based upon NUREG-0737, Item II.K.3.16. To respond to this Item the Applicant took several steps to improve SRV reliability and to reduce the number of SRV challenges. However, the contention asserts that Item II.K.3.16 requires further reductions in SRV challenges, and therefore that the Applicant's response is insufficient. Finding 22/28:2.

Two other issues, unrelated to the two contentions, arose concerning two-stage Target Rock safety relief valves. First, the problem of SRV "set-point drift" was highlighted by Board Notification 82-79, and was discussed by the witnesses. Following this, the Board raised the questions of polymerization of SRV lubricants and its possible relationship to set-point drift. We will briefly discuss our findings on these questions below.

The Applicant and Staff presented a combined witness panel on all the SRV issues. Findings 22/28:3-22/28:4. Suffolk County also presented two witnesses on the subject. Finding 22/28:5. Based upon the record before us, the Board is able to conclude that the two SRV contentions are without merit. Furthermore, Board Notification 82-79 and the existence of set-point drift provide no basis for altering this conclusion, and do not pose a significant safety concern for Shoreham.

SC Contention 22: TMI Item II.D.1

The objective of NUREG-0737, Item II.D.1, "Performance Testing of Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves," was to require testing of the performance of the SRV's for liquid or two-phase flow conditions. Finding 22/28:7. In response to this Item, the BWR Owners' Group, of which the Shoreham Applicant is a member, contracted with General Electric to develop and implement a generic SRV test program. The valve test program was completed and the results submitted to the NRC staff on September 25, 1981. Finding 22/28:8.

The first issue under SC 22 concerns the selection of test conditions to be included in the generic II.D.1 test program. The test conditions are required by Item II.D.1 to be based upon a determination of the "expected valve operating conditions" for which liquid or two-phase flow through the SRV's is reasonably likely. Finding 22/28:9. The BWR Owners' Group identified the alternate shutdown cooling mode as the only operating condition to be tested. Because other transients and operating conditions have such low probability of occurrence and consequences which would not exceed the design basis accident, testing for those conditions was not warranted. Finding 22/28:10. The NRC staff accepted the test condition chosen for the Owners' Group study. Finding 22/28:11.

One of the events analyzed, but not included as a test condition, was a potential high pressure vessel overfill event which would result in a liquid flow through the steam lines and the SRV's. However, Shoreham is equipped with water level 8 injection trips. These trips would shut off the water before the level reaches the steam lines, making a flow through the SRV's extremely unlikely. Furthermore, even if such an event were to occur, the loads on the valves, the piping, and the pipe supports that would result would be bounded by the loads calculated for the design basis steam line break analysis. Findings 22/28:11, 22/28:22. Therefore, a high pressure test condition is not required for the Item II.D.1 test program.

Contention SC 22 specifically asserts that the SRV test program is deficient because it did not include testing for ATWS conditions. However, the record shows that such testing is not required for BWR's. The Staff witness, a member of the committee that drafted Item II.D.1, clarified the slightly ambiguous language of the Item. It was his testimony that ATWS test conditions were only intended to be required for PWR's. Finding 22/28:12. There are sound technical reasons for this distinction. Most notably, it is unlikely in a BWR that the water level will increase during an ATWS event. Therefore water will not reach the steam lines to challenge the SRV's. Furthermore, even if water level did rise, level 7 alarms and level 8 trips virtually eliminate the possibility of water reaching the steam lines. Finding 22/28:13. The Board therefore concludes that ATWS testing is not required for Shoreham in order to comply with Item II.D.1.

The second major issue under SC 22 concerns the applicability of the results of the generic SRV test program to the specific configuration of the Shoreham plant. The issue is whether or not the loads on the Shoreham valves, discharge piping, and piping supports are enveloped by the loads calculated in the test program. The Applicant had considered this issue and submitted a position to the NRC staff on December 9, 1981. Finding 22/28:17. The Staff had some questions on the Applicant's conclusion, and sent them to the Applicant on July 8, 1982. Finding 22/28:18. The questions were discussed by the witnesses and the Staff accepted the Applicant's responses. Findings 22/28:19-22/28:27.

The record supports the conclusion that the generic BWR Owners' Group test results are applicable to the Shoreham plant. The generic report contains the results of pipe and support load measurements performed for both low pressure liquid test conditions and operating pressure steam conditions. In all cases the loads measured for the liquid discharge conditions were considerably lower than under steam conditions. The Shoreham SRV piping and supports have been designed and approved for loads endured during steam flow conditions. Therefore a design margin assures that the Shoreham configuration will be adequate for liquid flow conditions. Finding 22/28:17.

Only one configuration difference between the test facility and the Shoreham plant was highlighted. Finding 22/28:20. Shoreham's SRV discharge lines are supported by spring hangers in conjunction with snubbers and rigid supports. The test configuration did not utilize spring hangers. The Staff required the Applicant to confirm that any increased loads on the Shoreham SRV's due to the piping difference would still be offset by the design margin. The Applicant submitted the stress analysis results on December 15, 1982, and concluded that the loads at Shoreham from liquid discharge events would be lower than for the design basis steam conditions. Id. The Board finds the evidence persuasive, and concludes that the generic Owners' Group load analysis is applicable to the Shoreham plant, and that the plant will safely accommodate the loads of a liquid flow condition.

Based on the record on all of the above issues, the Board is able to conclude that the Applicant has conducted an SRV test program which meets the requirements of NUREG-0737, Item II.D.1.

SC Contention 28(a)(vi): TMI ITEM II.K.3.16

NUREG-0737, Item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves -- Feasibility Study and System Modification," represents an effort to reduce the incidence of stuck-open or spuriously opening SRV events (SORV's). The Item directs that a study be made of the feasibility of various possible measures to achieve this goal, and that those measures which do not compromise the performance of the valves be taken. Finding 22/28:29. In response to the Item the Shoreham Applicant participated in a BWR Owners' Group evaluation of methods available to reduce SRV challenges and stuck-open SRV events. The Applicant then identified three modifications being implemented at Shoreham which, based upon the results of the evaluation, would constitute compliance with Item II.K.3.16. Finding

22/28:30. Contention SC 28(a)(vi)/SOC 7.A(6) questions the adequacy of the measures to be taken at Shoreham to meet the Item.

The three measures selected by the Applicant include: (1) the use of Target Rock two-stage SRV's, (2) the use of an operating procedure providing for manual implementation of low-low set relief, and (3) a lowering of valve reclosure set-point. Finding 22/28:31. The Target Rock two-stage SRV was developed to improve reliability over the Target Rock three-stage design. The two-stage design eliminates the middle stage of the three-stage valve, which was the major cause for many SORV events. However, this change does not reduce the number of SRV challenges. Finding 22/28:32. A reduction in the number of challenges is achieved through implementation of the low-low set relief procedure. The procedure directs the operator to hold open a valve beyond the reclosure set-point, thereby providing an additional depressurization, and reducing the need for subsequent SRV actuations. Finding 22/28:33. Similarly, the lowering of the reclosure set-point allows the valve to automatically remove more heat with the initial SRV actuation, eliminating subsequent challenges. Finding 22/28:34.

The Staff reviewed this submittal from the Applicant and determined that it would be sufficient to meet Item II.K.3.16. However, the Staff held open the generic review of Item II.K.3.16 to consider requiring the additional measure of changing the set-point on water level for main steam isolation valve closure. This procedural change would further reduce the number of SRV challenges. On January 7, 1983, the Applicant submitted a commitment to make this change at Shoreham. Findings 22/28:37-22/28:38. The Staff also noted in its testimony that Shoreham will be equipped with an improved pneumatic supply control system to the SRV's. This improvement will eliminate the small number of challenges caused by pressure regulator malfunctions. The reduction had not been taken credit for by the Applicant. Finding 22/28:31, 22/28:35.

Suffolk County asserts that LILCO has failed to meet the strict requirements of NUREG-0737, Item II.K.3.16. The County argues that the Item calls for a ten-fold reduction in magnitude of SRV challenges. Because the Target Rock two-stage SRV's do not literally reduce challenges, the County would disallow their inclusion as part of the Applicant's response to the Item. The actual reduction in challenges acheived at Shoreham would only be 20 to 30 percent. Finding 22/28:36. The Board, however, finds this construction to be overly restrictive. The language and title of the Item make clear that the goal of the Item is to reduce SORV's. Reducing challenges is only a means to this end with no independent safety significance. The Staff witness, who was a member of the committee that drafted the Item, verifies that improved reliability of the SRV's is the objective of Item II.K.3.16. Finding 22/28:30. The Staff witness estimates that the change to the two-stage SRV alone will result in a reduction by a factor of eight in the number of SORV events, compared to a BWR 4 with three-stage SRV's. Finding 22/28:36. Therefore, the Board finds that the two-stage Target Rock valve is a legitimate response to NUREG-0737, Item II.K.3.16.

The County also argues that the two-stage Target Rock valve should not be considered a proper response to Item II.K.3.16 because the decision to use that valve at Shoreham pre-dates the Item. However, this assertion lacks basis. The BWR Owners' Group study, in calculating reductions in SORV event frequency, properly utilizes the BWR 4 with three-stage Target Rock valves as the benchmark. Finding 22/28:36. The two-stage design represents an improvement over the benchmark, and it is irrelevant when the decision was made to use the valves at Shoreham.

The Board also finds that the number goal of Item II.K.3.16 is not a goal to be strictly construed. While the express language of the NUREG may call for a reduction in SORV's by "an order of magnitude," the Staff properly treats the language as a flexible goal. Finding 22/28:37. The focus of Item II.K.3.16 should not be to acheive a factor of ten reduction, but to identify and implement all modifications to reduce SORV's which can be accomplished without compromising valve performance. Finding 22/28:29. No such modifications have been identified for Shoreham which are not being implemented. See Finding 22/28:39.

The Board concludes that the Applicant has successfully responded to NUREG-0737, Item II.K.3.16, and that Contention SC 28(a)(vi)/SOC 7.A(6) is without merit.

Board Notification 82-79

On July 26, 1982, shortly before the hearing on SRV issues, the NRC staff issued Board Notification 82-79, "Opening Pressure of Two-Stage Target Rock Safety Relief Valves." Finding 22/28:41. The notification recounts a recent event at the Hatch 1 plant in which eight of eleven Target Rock two-stage valves did not open at a pressure exceeding the nominal set-points. Finding 22/28:42. The NRC staff is studying the Hatch 1 event to determine the causes. However, it was the Staff's judgment that a more rapid rate of pressurizing the system, or a higher maximum pressure, would have caused most or all of the SRV's which remained closed to open. Finding 22/28:44.

The Hatch 1 event is an example of a problem known as "set-point drift" which results in a failure of the valve to open at designated pressure. The problem is unrelated to either NUREG-0737, Item II.D.1, or Item II.K.3.16, and therefore does not fall within the scope of either of the two SRV contentions or alter the Board's conclusions on those contentions. Finding 22/28:45.

The evidence heard does indicate that "set-point drift" is a long standing minor problem generic to all SRV's. All valves, after they have been in service for a period, demonstrate a tendency for the opening pressure to vary from the set-point. However, this variance is not considered to be a design problem. Furthermore, when a variance from conservative tech spec limits is noticed in post-service testing, the valves are required to be repaired, reset, and retested prior to reinstallation. Finding 22/28:45. This provides assurance that set-point drift will not result in a significant safety hazard.

Board Notification 82-79 also includes a counter-example to the Hatch 1 experience. At Browns Ferry 2, ten of eleven Target Rock two-stage SRV's successfully opened at pressures within their set-point tolerances. For the one valve that did not open, the pressure never did exceed the nominal set-point. Finding 22/28:43.

SRV Maintenance and Lubricant Polymerization

At Shoreham, station procedures will be implemented for operation, maintenance, testing, and surveillance of the SRV's. Finding 22/28:47. This will aid in assuring long-term reliable performance of the valves, and minimization of set-point drift. Furthermore, the Shoreham SRV's will not use lubricants such as castor oil which are subject to polymerization under reactor operating conditions. Finding 22/28:49. Such lubricants have been suggested as possible contributors to set-point drift on Target Rock two-stage SRV's. Finding 22/28:48.

H. Post Accident Monitoring (SC 27/SOC 3)

SC Contention 27/SOC Contention 3 addresses post accident monitoring instrumentation. Specifically, the Intervenors contended that Shoreham failed to comply with the guidance contained in Revision 2 of Regulatory Guide 1.97 in 11 designated areas. After various agreements were reached between the parties, only four items remained in the contention: Radiation Exposure Rate Monitoring; Secondary Containment Area Radiation Monitor; Drywell Spray Flow and Suppression Chamber Spray Flow; and Standby Liquid Control System Flow. Findings 27:1-27:3.

Regulatory Guide 1.97, Revision 2, published in December, 1980, provides guidance for the design and qualification of instrumentation used to monitor plant environs and systems during and after an accident. The time for implementation of Reg. Guide 1.97, Revision 2 is discussed in SECY 82-111 which was approved by the Commission on July 16, 1982. Finding 27:5.

According to SECY 82-111, the implementation dates of Reg. Guide 1.97 will be established after the Staff finalizes generic requirements for emergency response capabilities. At that time, Applicants for operating licenses and licensees of operating plants will be required to submit a schedule to the Staff for completing actions to comply with the NRC requirements. The Staff and Licensees/Applicants are then to arrive at a mutually agreeable schedule for each individual plant. Findings 27:5-27:6.

In accepting the implementation schedule, the Commission instructed that SECY 82-111 be published as a supplement to NUREG-0737, and that SECY 82-111 items should be accorded the status of approved NUREG-0737 items. Finding 27:7.

In light of the implementation schedule for Reg. Guide 1.97 approved by the Commission, the NRC staff has not reviewed Shoreham's compliance with the Reg. Guide. At the hearing, the Staff took the position that compliance with the Reg. Guide (or provision of an equivalent alternative level of protection) could await the approved implementation date and that reasonable assurance existed that safe operation of the plant could be assured in the interim period. Findings 27:8-27:10. For their parts, the Applicant and the Intervenors both offered their technical positions on how the regulatory requirements identified in the Reg. Guide could best be met. Inasmuch as the Staff has deferred its review of Reg. Guide 1.97 matters, the Staff did not comment on the views put forward by the Applicant and Intervenors. Finding 27:12.

As mentioned earlier, the implementation dates set forth in SECY 82-111 are to be treated as NUREG-0737 items. In its "Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses," dated December 18, 1980, the Commission determined that parties to licensing proceedings could challenge either the necessity for or sufficiency of NUREG-0737 requirements. CLI-80-42, 12 NRC 654, 660; see also Pacific Gas and Electric Company (Diablo Canyon Plant, Units 1 & 2), CLI-81-5, 13 NRC 361 (1981). The Commission added that "[i]t would be useful if the parties in taking a position on [the necessity for or sufficieny of] such requirements stated (a) the nexus of the issue to the TMI-2 accident, (b) the significance of the issue, and (c) any differences between their positions and the rationale underlying the Commission consideration of additional TMI-related requirements." CLI-80-42, supra, 12 NRC at 660.

In this proceeding, the Staff argued that Shoreham meets the guidance of the Standard Review Plan and that implementation of Reg. Guide 1.97 is not necessary for the interim safe operation of the plant. Findings 27:8-27:10. In addition, the fact that the Commission approved the implementation schedule for both operating license applicants and holders of operating licenses indicates that the Commission also believes immediate implementation of Reg. Guide 1.97 is not necessary for safe operation of a nuclear facility. See Finding 27:6.

In arguing that implementation of Reg. Guide 1.97 could not be deferred until after operation, the Intervenors advanced three arguments:

1. the accident at Three Mile Island occurred after the equivalent of only 90 days of operation;

2. Applicant's resolution of the Reg. Guide 1.97 requirements might become permanent; and

3. the Standard Review Plan is too old to be reliable.

Findings 27:11, 27:13.

The Board finds Intervenors' arguments to be without merit. As to the argument that the TMI accident occurred after the equivalent of 90 days of operation, the short answer is that the Commission was well aware of the occurrence at TMI when it approved SECY 82-111. Finding 27:14.

The implementation dates approved by the Commission applied not only to license applications, but to operating reactors as well. Finding 27:6. It is apparent that the Commission was of the belief that, notwithstanding the accident at the Three Mile Island, the implementation dates for Reg. Guide 1.97 could be deferred. The Intervenor has provided no reason to believe that the Commission was incorrect in its belief.

We are not persuaded that Applicant's resolution of Reg. Guide 1.97 - related items will be permanent if not reviewed today. The NRC staff will review Applicant's implementation of these items in accordance with the schedule approved in SECY 82-111. Finding 27:15. There is no reason to believe the NRC staff will not perform a review in good faith of Reg. Guide 1.97 items. Nonetheless, if the Intervenor is dissatisfied in any way with that review, the Intervenor will have the option of applying for an order to Show Cause according to the provisions of 10 C.F.R. § 2.206. The fact that the Staff has not yet conducted its review of Reg. Guide 1.97 items does not mean that a review will never be conducted, nor does it mean the Intervenor will be unable to examine and challenge that review if it so desires.

Finally, as to Intervenor's complaint that the standard review plan is too old to be reliable, the Commission has taken many actions in the post-TMI period to provide additional assurance in the areas of accident prevention and mitigation. <u>See</u> <u>e.g.</u>, NUREG-0737. The very area covered by this contention, post-accident monitoring, has been squarely addressed by the NRC staff in SECY 82-111, a document approved by the Commission in July of 1982. A complaint addressing the age of the Standard Review Plan as it was applied to Shoreham ignores the Commission's recent determination that implementation of the equipment called for by Intervenors need not take place before the Shoreham plant becomes operational.

In sum, we find that the implementation date for Reg. Guide 1.97 is to be treated as a NUREG-0737 item. The sufficiency of such items may be challenged by a party to a license proceeding. CLI-80-42, <u>supra</u>, 12 NRC at 660. But the Commission's findings on post-TMI requirements are entitled to some degree of deference by its Licensing Boards. While an Intervenor may challenge these requirements, we believe it is incumbent upon that Intervenor to make at least some showing that the Commission's position is inadequate. Otherwise there would be no limit to the litigation of TMI-related items, a situation certainly never contemplated by the Commission:

> The Commission believes the TMI-related operating license requirements list as derived from the process described above [NUREG-0737] should be the principal basis for consideration of TMI-related issues in the adjudicatory process. There are good reasons for this. First, this represents a major effort by the staff and Commissioners to address more than one hundred issues and recommendations in a coherent and coordinated fashion. This entire process cannot be reproduced in individual proceedings. Second, the NRC does not have the resources to litigate the entire Action Plan in each proceeding. Third, many of the decisions involve policy more than factual or legal decisions. Most of these are more appropriately addressed by the Commission itself on a generic basis than by an individual licensing board in a particular case.

CLI-80-42, <u>supra</u>, 12 NRC at 660. The Intervenor in this proceeding has provided the Board with no reason to believe that the Commission was wrong in its belief that implementation of Reg. Guide 1.97 need not occur prior to plant operation. We therefore find the contention without merit.

III. FINDINGS OF FACT

A. Water Hammer (SC 4)

4:1 As admitted for litigation, Suffolk County Contention 4 is that:

> Suffolk County contends that LILCO has not demonstrated adequate assurance of the operability of safety-related piping to prevent or withstand the effects of water hammer because the Company has not considered the start-up experience at similar BWR plants. Therefore, Shoreham safety-related piping (e.g., ECCS, Reactor Decay Heat Removal Systems) does not meet 10 C.F.R. 50 Appendix A, GDC 1, 31, and 40.

4.2 In support of this contention, Suffolk County presented the testimony of its consultant, Marc W. Goldsmith, a nuclear engineer and president of the consulting firm of Energy Research Group, Inc.: Goldsmith, ff. Tr. 2381. The Applicant presented the testimony of Raymond E. Fortier, a Lead Power Engineer employed by Stone & Webster Engineering Corporation, and Richard A. Hill, a Systems Evaluation Programs Manager employed by the General Electric Company. Fortier, et al., ff. Tr. 1935.[4/] The Staff presented the testimony of Mr. Marvin W. (Wayne) Hodges who is a Section Leader in its Division of Systems Integration. Hodges, ff. Tr. 1940.

4:3 Water hammer is a single shock or series of shocks (pressure waves) produced by sudden changes in the flow conditions of fluids in a pipe that can cause damage to pipes and equipment. It typically occurs when

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4/ At the Board's request, Messrs. John J. Kreps and Jack A. Notaro of Long Island Lighting Co. also submitted supplemental written testimony on behalf of Applicant which was admitted into the record. Tr. 2681-83; LILCO Ex. 45, ff. Tr. 15,506. However, as a result of stipulation between the parties, neither Mr. Kreps or Mr. Notaro appeared at the hearing or were cross-examined. ff. Tr. 15,504, at 2.

a number of ways: by design of the facility to preclude water hammer where possible; by the implementation of a proper stress analysis to assure that the systems can withstand water hammer type loads; by proper operating procedures and training; and by testing. Tr. 2022, 2338-2339 (Fortier); Fortier, et al., ff. Tr. 1935, at 4-6. Although during cross-examination Intervenor Suffolk County attempted to descredit some of these programs, the record in this proceeding demonstrated their adequacy.

4:6 The design of Shoreham piping prevents or minimizes the effects of water hammer by: (a) Having all steam line piping provide for continuous draining to preclude the formation of water pockets. Fortier, et al., ff. Tr. 1935, at 5; Tr. 2054 (Hodges). (b) Utilizing pipe suppressors in safety related piping systems. Fortier, et al., ff. Tr. 1935, at 5; Tr. 2027-2034, 2040 (Fortier). (c) Using slow opening/closing electric motor operators to open and close automatic valves. Fortier, et al., ff. Tr. 1935, at 5; Tr. 2165-68 (Fortier); Tr. 2170-72 (Hodges). (d) Including high-point vents in water-filled lines to allow system venting to eliminate the formation of air pockets. Fortier, et al., ff. Tr. 1935, at 5; Tr. 2196-97, (Fortier); Tr. 2201, 2253-55, 2262-66 (Hodges). (e) Using vacuum breakers. Fortier, et al., ff. Tr. 1935, at 5; Tr.2024 (Fortier). (f) Using bypass valves to allow slow startup. Tr. 2019-2021 (Fortier).

4:7 In addition to these general design practices, the Shoreham design has various special systems to insure against water hammer. Among these systems is an ECCS loop-level fill system for low pressure core injection (LPCI), including portions of the residual heat removal (RHR), core spray (CS), and high pressure core injection (HPCI) systems.

Fortier, et al., ff. Tr. 1935, at 6; Tr. 2022 (Fortier). These loop-level fill systems preclude the occurrence of water hammer by operating continuously to maintain filled and pressurized water lines. They are electronically monitored by an alarm which will alert operators to malfunctions. There is periodic high point venting to ensure that the system design function is satisfied. Fortier, et al., Tr. 1935, at 6; Tr. 2051, 2253-55 (Hodges).

4:8 Another system which has been added at Shoreham to mitigate water hammer is the HPCI turbine steam supply preheating system which maintains the turbine supply piping at elevated temperatures. Such higher temperatures will reduce condensation so that water hammer is minimized during rapid start up. Fortier, et al., ff. Tr. 1935, at 7; Tr. 2022-23 (Fortier).

4:9 A stress analysis of the overall piping-system at Shoreham was performed which addressed the combination of loads, including the dynamic effects of water hammer. Computer modeling at Shoreham has been incorporated into this analysis and has been used as a basis for designing a support system within the allowable ASME III Code limits. Fortier, et al., Tr. 1935, at 7-9.

4:10 Although no specific procedures for water hammer are written into the Shoreham operating procedures, water hammer concerns are nevertheless taken into account in those procedures. Tr. 2303 (Fortier); Kreps, et al., ff. Tr. 15,505, at 1-3. At the Shoreham facility a Joint Test Group (JTG) has the overall responsibility for development, approval and implementation of preoperation test procedures and a Review of Operations Committee (ROC) serves this function with respect to all other plant procedures. The JTG and ROC include members who are familiar with the design of the system and who review all types of information where water hammer events may be reported. Kreps, et al., ff. Tr. 15,505, at 3-6.

4:11 Water hammer concerns do not need to be specifically spelled out in Shoreham operating procedures since the operators have already gone through a training program where water hammer problems have been dealt with and analyzed. Tr. 2303 (Fortier). In fact, it would not be desirable for Shoreham emergency procedures to specifically refer to water hammer since that would tend to clutter up these procedures by adding extraneous and redundant information and thereby detract from their purpose and utilization. Tr. 2311-15 (Hodges); Kreps, et al., ff. Tr. 15,505, at 2-3.

4:12 Water hammer is addressed in operator training by its inclusion in lectures and discussions in classroom training, by increasing operator awareness of water hammer during system walkdown and procedures training, by having operators participate in system preoperational and startup testing to gain water hammer experience and familiarity, and by disseminating information on water hammer experiences at other plants. Kreps, et al., ff. Tr. 15,505, at 7-8; Tr. 2339-40 (Fortier).

4:13 To confirm that Shoreham safety-related piping systems function properly, preoperational and start up tests will be performed at Shoreham. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operation transients. Fortier, et al., ff. Tr. 1935, at 9-10; Tr. 2059, 2061-62 (Hodges).

4:14 Based upon the above methods that have been utilized to eliminate or mitigate the effects of water hammer at the Shoreham facility, there is reasonable assurance that the Shoreham facility can be safely operated pending final resolution of the generic safety issue for water hammer. The methods utilized by the Applicant to eliminate or mitigate water hammer have been found acceptable to the Staff, at least until Staff's generic review is completed. They were believed to be acceptable because the measures that will be recommended in the Quadrex and EG&G reports (the reports upon which the ultimate Staff position on water hammer is expected to be based) are measures that the Applicant has already taken at Shoreham to mitigate or prevent water hammer. Tr. 2082-84, 2109-2110, 2118, 2127 (Hodges). Furthermore, it is Staff's opinion that the major recommendations which have already been made in a draft Quadrex report which has been issued, will not be changed. Tr. 2085 (Hodges).5/

4:15 Water hammer issues at Shoreham have also been largely resolved because in those instances where it was determined that water hammer could not be avoided, the Applicant remedied the problem by designing the affected equipment to accommodate water hammer loads. Tr. 2117-2122 (Hodges). In addition, an important factor regarding safety

5/ The only aspect of water hammer which Mr. Hodges was not sure was satisfied for Shoreham pertained to operating and training procedures. Tr. 2085, 2113-2114 (Hodges). Mr. Hodges admitted, however, that this was only his own opinion, that these were areas which were outside of his scope of review, and that he had not even looked at all of Shoreham's training manuals or operating procedures. Tr. 2114, 2125-2130 (Hodges).

concerns at Shoreham is that although there have been a number of water hammer events at other nuclear facilities, none have ever incapacitated a system. Tr. 2330 (Hodges); Tr. 2332-34 (Hill).

4:16 Intervenor Suffolk County's concern with water hammer in this proceeding is that preoperational or start up data from other plants has not been incorporated in the NRC's safety review of Shoreham or in the Applicant's preoperational or startup program. However, such comparisons are often not appropriate since many of the problems that might arise at one particular plant may not necessarily apply to other plants, and often the causes of the water hammer have already been analyzed and no new phenomena have occurred. Tr. 2062-2063, 2331 (Hodges); Tr. 2235-A (Fortier).

4:17 Notwithstanding that such comparison is not always necessary, the evidence in this proceeding has established that such events have in fact been taken into account at Shoreham. General Electric (GE), the nuclear reactor system supplier, has a regular program for BWR product evaluation. GE maintains a crew of experienced field representatives on every construction site who prepare daily and monthly product evaluation reports regarding any adverse product experiences. These reports provide a basis whereby any design problems or incidents that are caused by water hammer at all BWRs can be flagged, evaluated by the engineering department and appropriate changes can be made. Tr. 1997-1998, 2014-2015, 2335-E-2335-F, 2336 (Hill); Tr. 2058-2059 (Hodges). Stone & Webster, the project architect-engineer, also has a program whereby water hammer events at other nuclear facilities would be brought to bear on Shoreham. Tr. 2335-F-2335-G, 2037 (Fortier).

4:18 In its evaluation of Shoreham, the NRC Staff also took into consideration water hammer events at other facilities. The Staff had apprised itself through a detailed generic review (NUREG-0371) of water hammer events that had occurred both prior to and after commercial operation for both boiling water and pressurized water reactors. Based upon information obtained from this review, the Staff requested and received information for Shoreham concerning the ECCS design and operation provisions to prevent and mitigate water hammer. Hodges, ff. Tr. 1940, at 2-3, Attachment B.

4:19 The Intervenor has pointed to various nuclear facilities that it claims had water hammer events which were not evaluated at Shoreham. One such plant (the European reactor), referred to in the testimony of Intervenor's consultant Marc Goldsmith, was a European G.E. BWR Mark II (like Shoreham) that had experienced serious water hammer damage on RHR start up operations. Goldsmith, ff. Tr. 2381, at 3-5. However, that event had in fact been evaluated by the Applicant. An event report was prepared by GE of the European reactor incident and distributed to responsible people at the Shoreham plant including onsite GE personnel. It was determined that no changes should be made in the Shoreham design because Shoreham's system would minimize the type of phenomenon that occurred at the European reactor. Tr. 2000-2003 (Hill). The Staff also confirmed that the event at the European reactor had been adequately assessed. The Staff determined that the causes for that event were the same as the causes for a number of similar events that have been observed at other similar plants in the United States and there was accordingly no need to emphasize that event as opposed to ones that had occurred at domestic plants. Tr. 2070, 2074-2075 (Hodges).

4:20 The other plant which Mr. Goldsmith specifically referred to was Commonwealth Edison's LaSalle nuclear unit where a pipe vibration monitoring program has been conducted which the Intervenor believes should also be utilized at Shoreham. Goldsmith, ff. Tr. 2381, at 10. Contrary to this assertion, the use of the La Salle monitoring program at Shoreham is unnecessary since Shoreham already has a similar monitoring program. Tr. 2337-38 (Hill). Further, there is no need to directly compare LaSalle water hammer experience with Shoreham since the piping arrangements at the two plants are different. Tr. 2062-63 (Hodges).

4:21 Cross-examination also failed to demonstrate that the Applicant did not take into account water hammer experiences at other facilities listed in the EG&G Report. On the contrary, this testimony revealed that the Applicant or Staff either had taken this experience into account or there was no need to do so. Tr. 2163-64, 2176, 2208, 2216-2217, 2226, 2335-B-2335-E (Fortier); Tr. 2178 (Hill); Tr. 2185, 2214, 2231-32, 2236, 2237 (Hodges) 6/

6/ The Board does not view the Intervenor's use of the EG&G report as a reliable method to establish omissions regarding water hammer experiences by the Applicant since no one who participated in its production was available to testify and such testimony would have been necessary to establish that the circumstances surrounding the water hammer events at these other facilities was substantially similar to the situation at Shoreham. Tr. 2146-2148, 2212. Further discrediting the Intervenor's cross-examination with respect to the EG&G report was the fact that testimony revealed that some of the events listed in that report were probably not caused by water hammer Tr. 2235, 2237-2239.

4.22 For the above stated reasons this Board concludes that adequate measures have been taken to eliminate or mitigate water hammer concerns at Shoreham and that this facility can be safely operated pending resolution of the unresolved generic safety issue. This Board further finds that start-up experience for water hammer at similar BWR plants has been adequately considered for Shoreham.

B. ECCS Core Spray (SC 10)

10:1 SC Contention 10 states:

> Suffolk County contends that LILCO and the NRC staff have not adequately demonstrated that the Emergency Core Cooling System (ECCS) for Shoreham meet the requirements of 10 C.F.R. § 50.46 and Appendix K with regard to core spray distribution and counter current flow, as shown by the recent Japanese test data described in BN-81-49.

10:2 Prior to litigating this contention the parties agreed to the following stipulation:

> For the purpose of the testimony concerning SC-10, Core Spray, [assume] no direct core spray distribution to a central 54-inch diameter region of the reactor core.
>
> Tr. 2522.

10:3 Testifying for the Applicant on this contention was Mr. Richard A. Hill, the Manager of Systems Evaluation Programs in the Safety and Licensing Operation for the General Electric Company. Hill, ff. Tr. 2524, at 1. In that capacity, Mr. Hill is responsible for GE's generic licensing programs to resolve ECCS technical issues. Tr. 2545 (Hill). Testifying for the Staff was Mr. Summer B. Sun, a Nuclear Engineer in the Core Performance Branch of the Division of Systems Integration. Sun, ff. Tr. 2527, at 1. Mr. Sun's expertise is in the area of reactor core thermal hydraulics. Id., at Professional Qualifications; Tr. 2533-24 (Sun). Suffolk County presented no witnesses to support its contention, and proceeded only through cross-examination.

10:4 The minimum heat transfer coefficient for core spray cooling as specified in 10 C.F.R. Part 50, Appendix K, § D.6 is 1.5 Btu/hr-ft²-°F. This value is used in the GE ECCS Evaluation Model. Sun, ff. Tr. 2527, at 2. This value is considered to be conservative and there has been no basis for disputing its validity. Tr. 2551 (Sun).

10:5 The value for the convective heat transfer coefficient specified in Appendix K is explicitly acceptable for reactors having fuel rods in a seven by seven fuel assembly array. Shoreham has an eight by eight fuel array. Tr. 2550. (Hill). General Electric Company has performed tests at its two loop test apparatus (TLTA), from 1979-1981, to verify the effectiveness of its ECCS design for an 8 by 8 fuel array. Tr. 2554-2558 (Hill). The NRC Staff has reviewed these tests and has accepted the Appendix K value for the heat transfer coefficient as conservative for 8 by 8 fuel arrays. Tr. 2582-2583 (Sun).

10:6 The Shoreham core spray is delivered from a sparger ring around the side of the reactor vessel, but above the fuel bundles. Tr. 2582 (Sun).

10:7 Counter-current flow limitation (CCFL) is a phenomenon whereby an uprush of steam through the core limits the amount of water that can flow down through the orifices at the top. Tr. 2561-62 (Hill). A dynamic equillibrium is created between the upward force exerted by the steam on the water and the downward gravitational force on the water. Water is injected into the core whenever the weight of the water exceeds the force imparted by the counter-current flow. Tr. 2563-64 (Sun).

10:8 The effect of the CCFL phenomenon is that a pool of water accumulates at the top of the reactor core. Therefore it does not matter whether the direct core spray is distributed evenly or unevenly. An even water level across the top of the core is established by the pool. Tr. 2588 (Hill). From the pool a limited amount of water will flow down through each of the bundles. Tests performed by General Electric Company at its Lynn test facility show that there will be a flow from the pool through each of the bundles in the neighborhood of two to four gallons per minutes. Tr. 2592-2593 (Hill). This indirect flow, even with zero direct core spray flow to the central bundles, provides adequate coolant to justify the Appendix K heat transfer coefficient used in the GE ECCS Evaluation Model. Hill, ff. Tr. 2524, at 4-5. General Electric FLECHT data verifies that the minimum flow to each bundle to achieve the heat transfer coefficient of Appendix K is on the order of 1 gallon per minute. Sun, ff. Tr. 2527, at 2.

10:9 A second phenomenon which has been observed in Japanese tests and in tests at the Lynn facility involves coolant flowing down through quenched peripheral channels to increase the reflood rate. Hill, ff. Tr. 2524, at 5; Sun, ff. Tr. 2527, at 4. This occurs at the same time as the CCFL accumulation phenomenon. Tr. 2594 (Sun) The flow through quenched peripheral bundles has been observed to be approximately 100 gallons per minute per quenched bundle. Tr. 2593 (Hill).

10:10 Even assuming that the water accumulation phenomenon does not occur, such that there is no core spray flow through the central bundles, the peripheral flow through the quenched bundles and the resulting rapid reflood will insure that the peak clad temperature specified in 10 C.F.R.

50.46 is not exceeded. Sun, ff. Tr. 2527, at 4; Tr. 2596-97 (Sun). General Electric has performed and the Staff has reviewed a sensitivity study which verifies that with a core spray heat transfer coefficient equal to zero, the clad temperature remains less than 2200°F. Tr. 2633 (Sun). The flow distribution is not a critical factor from a thermal hydraulic point of view. Tr. 2598 (Sun).

10:11 In the General Electric ECCS Evaluation Model used for the Shoreham ECCS analysis, the flow down peripheral channels was not taken credit for. Tr. 2618 (Hill).

10:12 Additional cooling will be provided by the uprush of steam through the core. Steam cooling effects will provide a heat transfer coefficient greater than 1.5. Tr. 2597 (Sun); Hill, ff. Tr. 2524, at 5.

10:13 CCFL can also occur at the bottom of the fuel bundles. Tr. 2644 (Sun). This effect is caused by a steam updraft from the lower plenum into the bundles. It causes a slower drainage rate out of the bundles, thereby providing additional cooling Tr. 2644-45 (Sun/Hill); Hill, ff. Tr. 2524, at 5. This CCFL phenomenon at the bottom of the bundles will not block the flow of water into the bundles during the reflood stage. Tr. 2644 (Sun).

10:14 In the Shoreham ECCS there is a high pressure coolant injection system (HPCI), two low pressure coolant injections (LPCI), an automatic depressurization system, and two low pressure core sprays. Staff Ex. 2A, § 6.3.1, at 6-41. Under the single failure event criteria the Staff requires an analysis of a worst case single failure. For Shoreham the worst single failure event is a failure to open of one low pressure coolant injection valve in the unbroken recirculation loop. There will remain one

LPCI available. Tr. 2602-2603 (Sun). In this limiting case the Staff has concluded that peak clad temperature will not exceed 2200°F, even assuming a core spray convective heat transfer coefficient equal to zero, if credit is taken for the fast reflood phenomenon due to flow down the peripheral channels. Tr. 2631 (Sun); Staff Ex. 2C, § 6.3, at 6-2.

10:15 In conclusion, an assumption that direct core spray distribution to the central bundles is zero does not prevent Shoreham from complying with the requirements of 10 C.F.R. 50.46 and Appendix K. Applicant and Staff witnesses testified to several cooling phenomena present in the core which assure adequate cooling. This evidence was uncontroverted by Suffolk County.

C. <u>Passive Mechanical Valve Failures (SC 11)</u>

11:1 SC Contention 11 states:

> Suffolk County contends that LILCO has not demonstrated that the valves used in the safety-related systems at Shoreham will not fail in an undetectable or unsafe mode, thereby jeopardizing the safe operation of Shoreham and violating 10 C.F.R. 50, Appendix A, GDC 23, 34, 35, 37 & 40.

11:2 Applicant's witnesses on this contention were Mr. Raymond E. Fortier, Senior Power Engineer in the Power Division of Stone & Webster Engineering Corporation, and Mr. John J. Kreps, Startup & Test Engineer, NUS Corporation. Fortier, ff. Tr. 3629. Testifying for the Staff was Mr. Robert Kirkwood, Principal Mechanical Engineer in the Mechanical Engineering Branch of the Division of Engineering. Kirkwood, ff. Tr. 3741. Suffolk County presented Mr. Gregory C. Minor and Mr. Dale G. Bridenbaugh of MHB Technical Associates to testify on this contention. Bridenbaugh, <u>et al</u>., ff. Tr. 3545.

11:3 As indicated by the County's witnesses during cross-examination, the basic thrust of this contention is concern over the possibility of undetectable valve failures. Tr. 3692 (Minor). However, there developed some confusion at the hearing over the definition of a "passive mechanical valve failure." Tr. 3565, 3571 (Minor). 10 C.F.R. 50, Appendix A, Definitions and Explanations, uses the terms active and passive to refer to types of components, not types of failures. Passive mechanical valves are those valves that require no mechanical movement to perform their safety function, while active valves do require mechanical movement to perform their safety function. Tr. 3640 (Fortier). In the context of this contention, however, a "passive" failure is interpreted to mean "undetectable" failure whether of an active or passive valve. Tr. 3561-62 (Minor). Hence, a passive mechanical valve failure usually refers to the mechanical failure of an active valve, such as the separation of the stem from the disc in a main steam line isolation valve (MSIV). Tr. 3645 (Fortier).

11:4 Shoreham has been designed to prevent passive mechanical failures by utilizing valves in the safety-related systems that are in accordance with approved codes and standards -- specifically ANSI B.31.1.0. Power Piping, ASME Code Section III, Class 1, 2 & 3, and Draft ANSI Code for Pumps and Valves for Nuclear Plants, Class 1, 2 & 3. The valves used at Shoreham are the standard valves used throughout the industry and they are chosen for their high performance reliability and they meet the quality guidelines. Kirkwood, ff. Tr. 3741, at 2; Fortier, ff. Tr. 3629, at 4. Additionally, these valves are designed against undetectable failures in that there are position indicators or other monitoring devices to detect valve failures. Further, there is redundancy built into the systems to satisfy the single failure criteria in the event of a passive mechanical valve failure. There will also be an in-service inspection and testing program. Fortier ff. Tr. 3629, at 3; Kirkwood, ff. Tr. 3741, at 3.

11:5 The Intervenors raised four basic concerns with regard to passive mechanical valve failures: Applicant's design basis analysis does not meet the single failure criterion for passive mechanical valve failure in fluid systems as interpreted by the County; Applicant's testing program is inadequate; not all safety-related valves have monitors; and Applicant has not taken sufficient notice of industry operating experience. Bridenbaugh, et al., ff. Tr. 3545, at 2-8.

11:6 The single failure criterion as outlined in 10 C.F.R. 50, Appendix A, Definitions and Explanations, provides:

> A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety function.2/

2/ Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

11:7 The County has interpreted this definition to mean there must be consideration of passive (undetectable) failures in passive fluid systems along with the assumed active component failure. Tr. 3711. (Minor). However, as noted in the footnote, the question of whether passive component failures must be assumed in this analysis is still under development. Thus, there is no code or document to support the county's interpretation. Tr. 3714. (Bridenbaugh).

11:8 The analysis done by the Applicant does meet the single failure criterion outlined above in that Shoreham's fluid systems are designed against the single failure of active valves and the analysis included assumed failure of passive components such as pump seals, valve seals, and measuring devices (pressure connection piping). Tr. 3632 (Fortier). Whether the failure was active or passive, the consequences of that failure are bounded by the Applicant's active component analysis. Tr. 3634 (Fortier). Further, the assumed failure of the entire subsystem bounds the failure analysis of all valves in that system. Tr. 3697 (Minor).

11:9 The in-service inpsection and testing program for Shoreham is in accordance with the ASME rules for in-service inspection. Kirkwood, ff. Tr. 3641, at 3. The ASME Boiler and Pressure Vessel Code, Section XI, generally provides guidance for in-service inspection and testing of safety-related valves every 3 months. However, certain valves are exempted from this testing requirement by the ASME Code: manual vent valves, instrument valves, drain valves, and maintenance valves. Tr. 3779 (Fortier). Further, the ASME Code provides guidance for establishing the frequency of testing, but does not set out requirements. Applicant has sought relief from the 3 month interval for certain of the valves included in its draft valve test plan. Tr. 3635 (Fortier). However, this draft plan is still under review by the Staff and not all relief from testing requested may be granted. Tr. 3745 (Kirkwood). The Staff generally requires compliance with the ASME Code regarding the frequency of testing except in those cases where such testing would place the plant in an unsafe condition; and in those instances, a longer interval between tests, i.e. until cold shutdown or refueling, would not be unsafe and would meet Staff requirements. Tr. 3928-29 (Kirkwood). In those cases a period ranging from 3 months up to 2 years (normal refueling) is adequate. Tr. 3902 (Kirkwood). If the plant is shutdown for long periods, such that the testing interval might actually exceed 2 years, the entire system must be tested prior to startup. Tr. 3911 (Kirkwood).

11:10 Nearly all safety-related valves in the Shoreham systems have monitors. Motor operated valves have position indicators and air-operated valves have limit switches to detect stem movement. Tr. 3775 (Fortier). The remaining valves which do not have these monitors include the safety-relief valves (SRV's), which have downstream discharge indicators; the solenoid valves, which have lights detecting the electrical circuitry; and check valves. The reliability of these three types of valves in industry operating experience is such that position indicators on these valves are not called for. Tr. 3786 (Fortier). Further, all valves in the reactor coolant pressure boundary, with the single exception of the SRV's, are redundant and the system satisfies the single failure criterion.

Tr. 3772-3 (Fortier). Thus, if an undetected valve failure did occur, its consequences will be mitigated by this redundancy.

11:11 The Applicant has procedures in place for review and evaluation of industry standards and operating experience. Information from various sources, including the manufacturer, G.E., other plants, and NRC notices such as I&E Bulletins and Licensee Event Reports (LER's), goes to the Nuclear Operating Services Division. From there is it disseminated to the Technical Support Division which forwards such information to the Plant Manager and Chief Engineers. Final review and analysis is performed by the Independent Safety Evaluation Group (ISEG) for Shoreham. Tr. 3636 (Kreps).[7/] These procedures, along with the existence of the ISEG, assure industry information is reviewed for its applicability to Shoreham systems. If modification of components is required, such modifications can be implemented.

11:12 Specific concern was raised by the County over undetectable valve failures regarding the Rockwell-Edward Main Steam Line Isolation Valve (MSLIV). This particular valve has suffered failure in the past at the Brunswick Unit 2 and Hatch Unit 1 plants. Tr. 3791-92 (Fortier). The failures at those plants related to separation of the stem from the disc. As a result of this experience the manufacturer undertook an analysis of the component. Tr. 3796(Fortier). The manufacturer has recommended certain modifications of the valve to resolve the problem, and the Applicant

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7/ Further findings regarding the ISEG will be made at the time the parties submit proposed findings on QA/QC (SC Contentions 12-15).

has ordered the recommended modifications and will install them. Tr. 3934 (Fortier). No other specific valve problems were identified by Suffolk County.

11:13 It cannot be guaranteed that a valve will never fail. However, the use of highly reliable valves, the redundancy of the system, valve monitoring devices, a testing, inspection and surveillance program, and the fact that the system is designed using a single failure analysis which bounds the possibility of a passive mechanical valve failure, all assure that the consequences of a passive mechanical valve failure, if such occurs, will not jeopardize the safe operation of Shoreham.

D. Anticipated Transients Without Scram (SC 16)

16:1 SC Contention 16 states:

> Although the anticipated transients without scram issue is generically before the Commission in a rulemaking proceeding, Suffolk County contends that LILCO and the NRC Staff have not adequately demonstrated that Shoreham meets the requirements of 10 C.F.R. Part 50, Appendix A, GDC 20, regarding correction of the ATWS problem in the interim period of several years pending completion and implementation of the result of the rulemaking for Shoreham. This is because the interim measures to be taken at Shoreham, including operational procedures and operator training, will not compensate for the lack of an automatically initiated and totally redundant standby liquid control system (SLCS) which meets the single failure criterion.

16:2 Testifying for the Applicant on this contention were the following individuals: Leonard J. Calone, Chief Technical Engineer for the Shoreham Nuclear Power Station; Harry T. Carter, Plant Engineer for Operations at the Shoreham Nuclear Power Station; Eugene C. Eckert, Manager, Plant

Transient Performance Engineering for the General Electric Company; Henry C. Pfefferlen, Manager of BWR Licensing Programs for the General Electric Company; John A. Rigert, Lead Nuclear Systems Engineer for the Shoreham Project; and William P. Sullivan, Technical Leader in the Nuclear Energy Engineering Division of the General Electric Company. Calone, et al., ff. Tr. 8870, at 2-3. Testifying for the NRC staff was Marvin W. Hodges, a Section Leader in the Reactor Systems Branch of the Division of Systems Integration. Hodges, ff. Tr. 8872, at Professional Qualifications. Suffolk County presented no witnesses or testimony to support its contention, and proceeded only through cross-examination.

16:3 Anticipated transients without scram (ATWS) are events in which the scram system (reactor trip system) is postulated to fail to operate as required. This subject has been under generic NRC staff review as Unresolved Safety Issue (USI) A-9 for several years. Staff Ex. 2A, at B-8. The resolution of the generic issue will result from the current Commission ATWS rulemaking proceeding, and Shoreham will be required to make any modifications specified in that resolution. Staff Ex. 2A, § 15-3, at 15-6, 7.

16:4 The Commission has decided to permit Shoreham and other plants to operate prior to resolution of the generic ATWS issue. This conclusion was based on several factors, including: (1) the estimated low probability of anticipated transients with potentially severe consequences in the event of scram failure; (2) the favorable operating experience with current scram systems, and (3) the number of operating reactors. However, in order to further reduce the risk of ATWS events during the period prior to a final ATWS rule, the Staff has required that interim measures be taken. The interim measures being taken at Shoreham are: (1) the installation of a recirculation pump trip (RPT) system to reduce reactor power on a high vessel pressure or low water level signal; (2) the use of an ATWS operating procedure based upon emergency procedure guidelines developed by the BWR Owners' Group and reviewed and accepted by the NRC staff; and (3) the implementation by LILCO of operator training for ATWS events. Hodges, ff. Tr. 8872, at 3.

16:5 The Shoreham scram system consists of 137 individual control rods. Each rod is driven by two separate hydraulic pressure sources. Each control rod drive is scrammed as an individual unit. Hot shutdown can be accomplished if at least 50% of the control rods are inserted in a checkerboard fashion. These design features assure that the Shoreham scram system is highly redundant and highly tolerant of component failures. Calone, et al., ff. Tr. 8870, at 6-7.

16:6 The reactor protection system is designed to prevent fuel damage by initiating a scram if variables such as reactor pressure, power level, and water level exceed specified limits. The reactor protection system utilizes redundant and diverse sensors to monitor these variables. Id., at 7-8. The Shoreham plant is also equipped with an alternate rod insertion (ARI) system. This is a redundant and diverse system for initiating control rod insertion by actuating dedicated backup scram valves. Id., at 15. The ARI system is designed to insert the rods 15 to 20 seconds after the normal trip signal and a failure to scram. If ARI functions properly there will be no need for the operator to attempt to manually insert the rods. Tr. 8978-79 (Hodges). Although the ARI system will help reduce ATWS challenges it will not foreclose them. Tr. 9080-81 (Hodges).

16:7 A standby liquid control system (SLCS) is available at Shoreham to inject liquid boron solution into the reactor to achieve safe shutdown of the plant in the event of failure to scram. Calone, et al., ff. Tr. 8870, at 18. The system consists of two redundant pumps designed to pump singly at a rate of 43 gpm. Following an event at Browns Ferry, reported in U.S.N.R.C. I.E. Bulletin 80-17, July 3, 1980 (SC Ex. 38), LILCO and Stone and Webster performed a conceptual design review of the possibility of operating both pumps at once to increase the flow rate. The conclusion of the review was that extensive and costly modifications would be required in the plant piping to accommodate the increased flow. Tr. 9057 (Rigert); Tr. 9289-90 (Rigert).

16:8 Related to each of the redundant SLCS pumps is a squibb valve. The squibb valves fire off a safety-related switch. As long as the squibb valve fires, the pump can be manually operated from the location of the SLCS, as well as from the control room. If both squibb valves fail to fire, there is no way to inject the boron. Tr. 9104 (Calone).

16:9 The recirculation pump trip (RPT) at Shoreham is a feature which is automatically initiated by a high reactor pressure signal or a low water level signal. Calone, et al., ff. Tr. 8870, at 10. The system is designed to provide overpressure protection at the beginning of an ATWS event. The pump trip provides a reduction in reactor power to less than 40% in less than one minute. Tr. 9108 (Eckert). This rapid change in power is one of the design basis cases for fuel thermal limits, and therefore does not present an unusual problem for fuel structures. Tr. 9105-6 (Eckert).

16:10 Reactor operator training includes instruction and simulator experience with the Shoreham ATWS procedure. Tr. 9035-6 (Calone). As part of the training, operators must memorize immediate actions, and are liable to be tested on the ATWS procedures in the operator examinations. Tr. 9183-4 (Calone). The NRC review of Shoreham training in general is conducted by Region I (formerly I&E). The inspectors have made plant visits to observe training, interview instructors, and review schedules and exam results. Tr. 9236 (Calone). At the time of the hearing LILCO had just passed a five day review of the training program by the NRC. Tr. 9238 (Calone). The NRC also conducts the operator exams, which effectively test the training program. Id.

16:11 The LILCO emergency procedure which addresses the ATWS scenario is SP 29.024.01, "Transient with Failure to Scram," Calone, et al., ff. Tr. 8870, Attachment 1. This is the only ATWS procedure currently in existence for Shoreham, and is the procedure currently being used to train Shoreham operators. Tr. 8937 (Carter). It was based upon General Electric ATWS emergency procedure guidance of July, 1980. The current Shoreham ATWS procedure, SP 29.024.01, was reviewed by the NRC staff against the criteria established in a memo from Frank Schroeder to Darrell Eisenhut. On the basis of the comparison the ATWS procedure was deemed acceptable. Tr. 8972 (Hodges).

16:12 The NRC staff also reviewed the Shoreham ATWS procedure for human factors. A series of human factors comments were developed by the Staff and its consultants based upon Revision C of the procedure. These were forwarded to LILCO in approximately August 1981. Then on October 17, 1981, LILCO demonstrated their ATWS procedure, along with the rest of their emergency procedures for representatives of the NRC staff at the Limerick simulator. Staff Ex. 2C, § 13.5.2.6, at 13-4; Tr. 8991 (Carter). The NRC staff utilized this exercise to discuss and modify its August 1981 human factors comments on the ATWS procedure, Revision C. Although the simulator exercise was cut short because the simulator stalled two or three minutes into the transient, much of the procedure was demonstrated, including initiation of the SLCS. Tr. 9016 (Carter). Following the simulator exercise, the finalized Staff human factors comments were incorporated by LILCO into Revision E of the Shoreham ATWS procedure. Tr. 9008-9012 (Hodges/Calone). Revision E of SP 29.024.01 was approved by the Staff and written up in the Shoreham SER. Tr. 9008 (Hodges); Staff Ex. 2C, § 15.3. Following Staff approval of Revision E, a draft procedure, LILCO issued the final procedure Revision 0 (numerical). This final version is identical to Rev. E, except for one change on page 1, step 3.1.2, which replaced the word "refuel" with "shutdown." Tr. 9009 (Calone).

16:13 An ATWS is recognizable within 10-15 seconds of its occurrence. Tr. 9065 (Calone); Tr. 9182 (Hodges/Calone). The operator's first step is to perform the immediate actions of section 3.0 of the procedure. The operator is trained to have these steps memorized and to perform them without reference to the procedure. Subsequent actions are taken with the procedure in front of him. The first step of the subsequent operator actions, section 4.0 of the procedure, is to verify that the immediate actions have taken place. Tr. 9183-4 (Calone); Calone et al., ff. Tr. 8870, Attachment 1.

16:14 The first immediate action of the operator is to attempt to manually scram the reactor. This is specified in step 3.0 of the Shoreham procedure. See Calone et al., ff. Tr. 8870, Attachment 1. This is proper because insertion of the rods, contrasted with initiation of SLCS, is the fastest way to decrease reactor power level. Tr. 9203 (Hodges). Even if this attempt fails, and the operator does utilize SLCS, it is the opinion of the Staff witness that the operator should and will go back and continue attempting to insert the rods. Tr. 9205 (Hodges).

16:15 The witnesses during cross-examination discussed a limiting worst case ATWS scenario. The assumptions made included the following: the reactor is operating at full power, there is a total failure to scram, the manual scram does not work, the ARI does not work, the RPT does work, that the ATWS involves an MSIV closure, that the suppression pool temperature begins at 90°F., that service water temperature was 75°F., that both RHR systems functioned, and that HPCI and RCIC auto-started. Tr. 9164; Tr. 9174-76 (Calone/Hodges/Letsche). Step 3.6 of the emergency procedure provides the criteria for initiation of SLCS. After a failure to manually scram the reactor the operator is instructed to initiate SLCS if reactor power is above 6% or suppression pool temperature reaches 110°F. Time is not the key parameter. Tr. 9065-9068 (Hodges/Calone).

16:16 The 6% power parameter used in the procedure refers to the neutron level, which is being used as an indication of reactor power. The LILCO witnesses state that this will not be ambiguous to an operator. Tr. 9069 (Calone). More importantly, the procedure required some explanation as to when the operator would look for the 6% indication. The witnesses explained that as soon as the operator puts the mode switch to shutdown for a manual scram, he will scan the average power range monitors (APRMs). In a normal shutdown, power would decrease almost immediately below 6%. Therefore, if power stays above 6% after about 5 seconds the operator will know that he has his SLCS initiating event. Tr. 9068-69 (Calone/Hodges). In any event, even if the 6% parameter were unclear, the 110°F. parameter is not, and the operator will have clear basis to take action if the suppression pool temperature is more than 110°. Tr. 9160 (Hodges).

16:17 If an operator activates the SLCS and injects the sodium pentaborate into the reactor, the plant will be required to shut down for approximately 12 to 14 days to remove the boron. Tr. 9150 (Carter). This should not affect the operator's decision to inject or not to inject because the emergency procedure presents clear instructions to inject Tr. 9156 (Calone). This has been reiterated in the training in the procedure for Shoreham. Tr. 9157 (Carter). If an operator deviates from a procedure he will face the sanctions of the utility and the NRC. Tr. 9160 (Hodges).

16:18 To start the SLCS the operator must use a key to turn the switch. The key is located in a locker in the watch engineer's office, roughly 35 feet from the switch. The SRO, who is required to always be in the control room, has the key for the locker. The SRO would open the locker to get the SLCS key, and would give it to the operator at the switch. Tr. 9211-12 (Calone). Inside the locker the SLCS key is one of 35 keys on the first page of keys Tr. 9216 (Calone).

16:19 The LILCO witnesses considered the possibility of leaving the SLCS key in the switch on the panel. This would reduce the time for injection by about 15-20 seconds versus the key being in the locker. However, it is the opinion of the LILCO witnesses that this would not be sufficient benefit to outweigh the cost of defeating the purpose of the key lock switch, i.e. to prevent inadvertent operation. Furthermore, all key lock switch keys are kept in the locker. To do otherwise with the SLCS key would be an exception to the standard rule. Tr. 9257-8 (Calone). Therefore to prevent problems in (a) opening the locker containing the SLCS key, and (b) selecting the right key from the locker, LILCO has committed to put a breakable glass door on the key locker, and to have the standby liquid control key uniquely colored for easy identification. The key control procedures will also be modified to include a provision to ensure proper placement of the keys inside the key locker, and to periodically verify the placement. Tr. 9258 (Calone).

16:20 There are no SLCS flow meters at Shoreham. The operator's principal indication that boron is flowing into the reactor is the level indicator on the standby liquid control tank. Tr. 9028 (Hodges). The operator is trained to verify a level drop to determine that standby liquid control is injecting. Tr. 9030 (Calone). Furthermore, under step 4.1 of SP 29.024.01, the operator has explicit instruction to verify his immediate actions -- the last of which, in step 3.6, is to initiate SLCS. Calone, <u>et al</u>., ff. Tr. 8870, at Attachment 1.

16:21 The operator will also have an indication that the SLCS pumps are injecting boron from the pressure indicators on the upstream side of the squibb valves. If the pump does not start, the indicator will read zero.

If the pump starts but the valve fails to fire, the pressure will read "relieve valve pressure." If injection is achieved, pressure will read approximately 150 above reactor pressure. If the pressure is lower than that, the operator will have an indication of pipeline failure. Tr. 9222 (Calone).

16:22 To achieve hot shutdown, boron is needed in a concentration of 480 parts per million, based upon water density at hot conditions. Tr. 9097 (Eckert). This means that with water level in the normal range the operator will need to inject roughly 1,154 gallons of sodium pentaborate. Injection of the 1,154 gallons at a 43 gpm rate will take approximately 26.8 minutes. Tr. 9098 (Hodges). The number of gallons is irrespective of injection point, but does depend upon proper mixing in the reactor vessel. Mixing tests indicate that due to natural circulation, mixing efficiency will approach a value of one as water level increases. Improper mixing will only occur if water level is maintained down near the top of the fuel. It is the opinion of the Staff witness that the operator will normally raise the water level above that point after the boron is injected. Tr. 9099-9101 (Hodges).

16:23 The reactor water cleanup system would remove the boron from the water. Therefore, when the SLCS is activated the cleanup system is automatically isolated. One of the steps in the ATWS procedure is for the operator to verify isolation. The operator can do this visually by checking a green light on the control panel about two feet from the reactor panel. Tr. 9102-3 (Calone). The witnesses could think of no other systems or components which would adversely affect the performance of the SLCS. Tr. 9103 (Calone).

16:24 The BWR Owners' Group's generic emergency procedure guidelines are currently being revised into a Revision 2 which will incorporate the ATWS guidance. This Revision 2 was submitted by the Owners Group to the NRC staff for review. Tr. 8938 (Carter). Following Staff approval of Revision 2 of the guidelines, the Shoreham procedures will be rewritten to reflect the changes in the guidelines. Tr. 8937 (Carter). The Staff does not expect the changes to the Shoreham procedures to be made prior to commercial operation. Tr. 8940 (Hodges). The current ATWS procedure has been approved by the NRC staff for Shoreham, and unless changes to the procedure are truly significant the Staff does not want to undermine operator training by incorporating changes before they are fully developed. Tr. 8958 (Hodges).

16:25 Several changes to the Shoreham emergency operating procedure for ATWS may result from the Revision 2 to the BWR Owners' Group emergency procedure guidelines. First, the revised guidelines would incorporate the ATWS control procedures into the reactor control procedures. Tr. 8987 (Hodges). Any revision to Shoreham's procedures based on Revision 2 of the guidelines would likely incorporate the organizational change. Tr. 8946 (Hodges). Second, a substantive change may be made in the procedure for initiation of SLCS. Under Revision 2 the "or" in step 3.6 may be replaced by an "and." This would call for injection of boron only if power on the APRM scale is above 6% <u>and</u> suppression pool temperature reaches 110°F. Tr. 9203-4 (Hodges/Calone). Staff witness Hodges believes the "and" statement is preferable, Tr. 9203 (Hodges), but the Shoreham procedure is considered to be technically more conservative. Tr. 9203 (Calone). A third change in the procedures which may result from the revision of the guidelines is the addition of an extra step. The additional step will require the throttling of HPCI and RCIC to aid in decreasing the water level to control reactor power. Under the current Shoreham procedure, the feedwater pumps are tripped. This has the effect of lowering water level. However, throttling HPCI and RCIC will decrease the water level more quickly. Tr. 8946-51 (Hodges/Carter). A fourth minor potential change identified by the witnesses would add a step to secure boron injection if control rods are inserted after initiation of SLCS. The present procedure is more conservative in that the operator will inject the full content of the boron tank regardless of subsequent rod insertion. Tr. 9205-6 (Calone/Hodges). Any changes to the procedure guidelines will need to be approved by the NRC staff.

16:26 Staff witness Hodges discussed a "worst case" ATWS scenario which included the assumption that there is no RHR cooling. In this case, with one pump injecting boron at 43 gpm, and initiated at a suppression pool temperature of 110°F., it was his belief that suppression pool temperature would reach 220° to 230° before the reactor achieved shutdown. Tr. 9060-62 (Hodges). There is no data to support the safety of a temperature above 210° to 220°. Tr. 9064; 9071 (Hodges). The witness speculated that in this scenario there could be some clad melting or cracking. Tr. 9273 (Hodges). However, these facts were within the body of knowledge behind the Commission's establishment of the interim ATWS measures. The interim measures will aid in the mitigation of most ATWS events, but they may not help in the very worst case. This is the reason for the generic rulemaking. Tr. 9071-2 (Hodges).

E. Seismic Design (SOC 19(e))

19:1 SOC Contention 19(c) stated:

> A major contributing factor in the TMI-2 accidnet was that operating plants were not required by the NRC Staff (Staff) to be in compliance with current regulatory practices (i.e., Regulatory Guides, Branch Technical Positions, and Standard Review Plans). The TMI-2 accident also demonstrated that the current regulatory practices, practices similar to those being applied by the Staff in their safety evaluation of Shoreham, were in a number of cases not suitably conservative to properly protect the health and safety of the public (i.e. hydrogen generation, radiation shielding, source terms, and single failure criterion).
>
> SOC contends that the NRC Staff has not required LILCO to incorporate measures to assure that Shoreham conforms with the standards or goals of safety criteria contained in recent regulatory guides. As a result, the Staff has not required that Shoreham structures, systems, and components be backfit as required by 10 C.F.R. § 50.55a, § 50.57, and § 50.109 with regard to:
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> * * * * * * * * * * * * * * * * * *
>
> (e) Regulatory Guides 1.60 and 1.61. -- The design response spectra for the seismic design of Shoreham are not based on the standards in Regulatory Guide 1.60. Thus, the spectra have not been demonstrated to be sufficiently conservative to comply with 10 C.F.R. Part 50, Appendix A, Criterion 2, and 10 C.F.R. Part 100, Appendix A. In addition, LILCO did not use the Regulatory Guide 1.61 value of damping (4%) for the operating basis earthquake analysis of Category I reinforced concrete structures, but rather utilized a higher value of damping (5%), thereby also violaing the regulations just cited.

19:2 Witnesses for SOC 19(e) were provided by Applicant and the NRC staff. Direct testimony for the Applicant was provided by Dr. Albert Y. C. Wong, a Senior Structural Engineer with Stone & Webster. Wong, ff.

Tr. 3970. In addition to Dr. Wong, Dr. A. Stanley Lucks, Chief Geotechnical Engineer at Stone & Webster, took the stand for the Applicant. Direct testimony for the NRC staff was provided by Sang Bo Kim, a Senior Structural Engineer in the Structural Engineering Branch, Division of Engineering, NRR, and Dr. Robert L. Rothman, a Seismologist in the Geosciences Branch, Division of Engineering, NRR. Kim and Rothman, ff. Tr. 3979. None of the Intervenors filed direct testimony; they relied solely upon cross-examination of Applicant and Staff witnesses by counsel for Suffolk County.

19:3 Damping is a measure of the dissipation of seismic energy associated with an earthquake. Kim and Rothman, ff. Tr. 3979, at 3; Tr. 3986, 3988 (Wong). Damping figures are used to adjust seismic response spectra. As the damping factor increases, the response spectra will decrease. Tr. 4000 (Kim). Structural damping (also known as material damping) is a measure of energy dissipation of a structure under dynamic excitation. When a structure is deformed or strained, there is a certain amount of loss or dissipation of the strained energy. Kim and Rothman, ff. Tr. 3979, at 3; Tr. 3986 (Wong). Soil damping is the dissipation of seismic energy by means of the interaction of a structure and the surrounding soil. Kim and Rothman, ff. Tr. 3979, at 3; Tr. 3988-89, 4004 (Wong). Total system damping is a weighted combination of structural damping and soil damping. Kim and Rothman, ff. Tr. 3979, at 3-4; Wong, ff. Tr. 3970, at 7; Tr. 3986 (Wong).

19:4 The 4% damping figure contained in Regulatory Guide 1.61 addresses only structural damping. The 5% damping figure used in the Shoreham design addressed total system damping, a combination of structural and soil damping. Kim and Rothman, ff. Tr. 3979, at 3-4; Wong, ff. Tr. 3970, at 7-8; Tr. 3986-90 (Wong).

19:5 A 10% soil damping figure is conservative for the soil conditions at the Shoreham site. Tr. 3989-90, 3998, 4008-09 (Wong). A combination of 10% soil damping and 4% structural damping would result in a figure of 8% total damping, a figure greater than the 5% used in the Shoreham design. Wong, ff. Tr. 3970, at 8; Tr. 3889-90, 3998 (Wong).

19:6 The 5% total system damping figure used at Shoreham was found to be acceptable by the NRC staff. Kim and Rothman, ff. Tr. 3979, at 4; Tr. 3998-99 (Kim).

19:7 The Board finds the 5% total system damping figure used at Shoreham to be acceptable and in no way contradictory with the 4% structural damping figure contained in Regulatory Guide 1.61.

19:8 A "response spectrum" is "a plot of the maximum responses (acceleration, velocity, or displacement) of a family of idealized single-degree-of-freedom damped oscillators against natural frequencies (or periods) of the oscillators to a specified vibratory motion input at their supports." 10 C.F.R. Part 100, Appendix A; Suffolk County Ex. 15, ff. Tr. 4123, at 160-3. A design response spectrum is "a relatively smooth relationship obtained by analyzing, evaluating, and statistically combining a number of individual response spectra derived from the records of significant past earthquakes." Suffolk County Ex. 15, ff. Tr. 4123, at 160-3. Design response spectra are used in structural analysis and design. Wong, ff. Tr. 3970, at 2.

19:9 The Commission's Regulations require that design response spectra be developed for both the Safe Shutdown Earthquake and the

Operating Basis Earthquake. 10 C.F.R. Part 100, Appendix A, VI(a). The Safe Shutdown Earthquake (somtimes referred to as the Design Basis Earthquake) is that earthquake which is based upon an evaluation of the maximum earthquake potential at a site. The Operating Basis Earthquake is that earthquake which could reasonably be expected to occur at a site during the operating life of the plant. Id.; at III(c) and (d).

19:10 In providing that design response spectra be developed, the Commission's regulations state:

> In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the response spectra be smoothed design spectra developed from a series of response spectra related to the vibratory motions caused by more than one earthquake.

Id.; at VI(a).

19:11 One manner of developing design response spectra is discussed in Regulatory Guide 1.60. Suffolk County Ex. 15, ff. Tr. 4123. However, an Applicant for an operating license is not required to establish compliance with regulatory guides. Id., at 1; Kim and Rothman, ff. Tr. 3979, at 5.

19:12 The design response spectra for Shoreham were not developed in the manner described in Regulatory Guide 1.60. Regulatory Guide 1.60 was published after the response spectra for Shoreham were developed, and the Staff did not intend that the Guide be applied to Shoreham. Tr. 4183 (Rothman); 4184 (Lucks); Kim and Rothman, ff. Tr. 3979, at 6.

19:13 The development of the Shoreham response spectra is described in the Shoreham FSAR. LILCO Ex. 10, ff. Tr. 4121. The spectra were developed from the records of four actual earthquakes (El Centro 1940, Taft 1952, Helena 1935, and Golden Gate 1957), and an artificial earthquake having properties intermediate between Taft and Helena. Id., at 3.7-3; Tr. 4201-4210 (Wong and Lucks). The records selected were primarily from stiff sites that contained a broad frequency content of motion. Tr. 4182 (Lucks). All four earthquake records represent strong motions of magnitude 5.3 and up. Tr. 4237 (Lucks). The artificial earthquake was developed to provide additional input. The Taft and El Centro records were rich in all frequencies and were long in duration. Helena, on the other hand, was a short burst of high frequency. The artificial earthquake was given duration and frequency contents in between these groupings, to ensure that all frequencies were adequately considered. Tr. 4182, 4237-38, 4268-69 (Lucks).

19:14 In developing the Shoreham response spectra, the time history records from each of the four earthquakes used were modified to reflect the site conditions at Shoreham. Shoreham is considered a deep cohesionless soil site with a soil depth of 1100 feet. Tr. 4012 (Wong). In modifying the four earthquake time history records, an amplification analysis was performed by entering the original records at the bedrock 1100 feet under Shoreham and then amplifying or deamplifying (depending upon the frequencies involved) the records up through the Shoreham soil profile. Tr. 4181-82, 4232-33, 4241 (Lucks); 4251 (Rothman); LILCO Ex. 10, ff. Tr. 4121, at 3.7-3. To accurately reflect the vibratory ground motion, the records were scaled to give peak velocity at the surface of eight inches per second, the peak velocity associated with an intensity VII earthquake (the Safe Shutdown Earthquake for the Shoreham site). Tr. 4242 (Lucks); LILCO Ex. 10, ff. Tr. 4121, at 3.7-3.

19:15 After the time history records modified for the Shoreham site were developed, a spectrum that adequately enveloped the spectra of the time history records was selected to be used for the actual design for Shoreham. The Applicant settled upon a .2g Housner spectrum adjusted for frequencies below 2 hertz as an appropriate enveloping spectrum. Tr. 4213-15 (Wong); 4222-23 (Lucks); LILCO Ex. 10, ff. Tr. 4121, at 3.7.3. The .2g Housner spectrum modified below 2 hertz completely enveloped the spectra of the time history records except for a few peaks of short range. In developing smooth curved response spectra, infrequent peaks exceeding the spectra do not affect the spectra's overall validity for design purposes. Tr. 4271-74 (Wong and Lucks). The .2g Housner spectrum modified below 2 hertz was used as the design spectrum for Shoreham because it was a known spectrum that adequately enveloped the spectra of the time history records. Tr. 4226-28 (Wong and Lucks). The NRC staff reviewed the earthquakes selected and the amplification analysis used and agreed that the approach taken by Applicant in its seismic design was acceptable. Tr. 4240-41, 4245 (Rothman).

19:16 The Board finds that the design response spectrum developed by the Applicant for use at Shoreham meets the requirements of 10 C.F.R. Part 100, Appendix A. Records were used from more than one earthquake, these records were modified to reflect the actual site conditions at Shoreham, and a smoothed spectrum enveloping the spectra of the modified records was then used for the Shoreham design. The original records used were appropriate, the amplification analysis used to modify the records was appropriate, and the spectrum used for the Shoreham design adequately enveloped the spectra of the records.

F. Mark II Containment (SC 21)

21:1 Contention 21 asserts there is inadequate demonstration that the Mark II containment design for Shoreham meets the requirements of general design criteria 4, 16, 50, 51 and 52 set forth in 10 C.F.R., Appendix A. Specifically, Contention 21 states:

> LILCO and the NRC Staff have not adequately demonstrated that Shoreham's primary containment, reactor pressure vessel supporting structure and attached and associated safety-related equipment meet the requirements of 10 C.F.R. 50, Appendix A, GDC 4, 16, 50, 51 and 52. The specific concerns are as follows:
>
> (a) Forces generated during the suppression pool LOCA dynamics have not been completely and adequately determined and taken into account. Of the numerous Mark II Containment loads assessed on a generic and on a plant unique basis under the Mark II reassessment program underway for the past six years, several LOCA forces have not yet been shown to have been suitably handled in the design of the structures, systems, and components important to plant safety. Included in this category are the forces due to Steam Condensation Downcomer Lateral Loads (Loads I.B.1.a & b in Table 6-1, NUREG-0420, Supp. No. 1), Steam Condensation Oscillation Loads (Loads I.B.2.a in Table 6-1, NUREG-0420, Supp. No. 1), and Steam Condensation Chugging Loads (Loads I.B.2.c in Table 6-1, NUREG-0420, Supp. No. 1).
>
> (b) Forces generated during safety relief valve (SRV) actuation, continuing SRV blowdown, and those due to suppression pool heatup resulting from such extended blowdowns have not been demonstrated to be adequately accommodated. Concern specifically remains for the Quencher Air Clearing Loads (Loads II.B in Table 6-1, NUREG-0420, Supp. No. 1), Steam Condensation Submerged Drag Loads (Loads III.C in Table 6-1, NUREG-0420, Supp. No. 1), and proper specification and accommodation of the suppression pool temperature limit (phenomenon II.A in Table 6-1, NUREG-0420, Supp. No. 1).

(c) The capability and adequacy of the test procedure to periodically demonstrate an acceptable leakage rate of the drywell floor seal and downcomer vacuum breakers and other leakage paths that could lead to excessive steam bypass of the suppression pool has not been demonstrated.

(d) Adequacy of the design to insure, with sufficient margin, that the primary containment and associated safety-related structure can accommodate the simultaneously applied loads of transient and LOCA events has not been demonstrated.

(e) Suffolk County further contends that the extent of the deficiencies resulting from the Mark II containment design program may be further exacerbated by the fact that an adequate and properly controlled experimental design verification program as required by 10 C.F.R. 50, Appendix B, Sections III and XI has not been performed. The verification of the design adequacy of the primary containment, reactor pressure vessel supporting structure, and associated safety-related systems and components is deficient with specific regard to testing under the most adverse design conditions, performance of tests under suitable environmental conditions, documentation and evaluation of test results, and use of test data developed under a non-controlled (foreign) test program. There is, therefore, lack of assurance that the acceptance criteria used by LILCO in evaluating the Shoreham design contains suitable conservatism.

21:2 Although Contention 21 as originally admitted consisted of all five of the above subparts, Suffolk County voluntarily withdrew its concerns relative to the load definitions associated with SRV actuation - Contention 21(b). Hence, this subpart of the contention was not addressed.

21:3 The Applicant presented Mr. Hancock Chau, Manager of the Nuclear Licensing Division of LILCO, Mr. Charles A. Malovrh, Lead Engineering Mechanics Engineer of Stone & Webster Engineering Corporation, and William M. Davis, Project Manager for Containment of General Electric Company to testify on this contention. Chau, _et al_., ff. Tr. 9735. The Staff witnesses are outlined in each subpart of the contention. Suffolk County presented no witnesses in support of this Contention and proceeded only by cross-examination.

Contention 21(a)

21:4 Contention 21(a) asserts that several forces generated during the suppression pool LOCA dynamics have not been completely and adequately determined and taken into account. Of the Mark II containment loads assessed on a generic and plant unique basis, the contention states that several have not been adequately handled, including forces due to Steam Condensation Downcomer Lateral Loads, Steam Condensation Oscillation Loads, and Steam Condensation Chugging Loads.

21:5 Testifying for the Staff was Mr. Farouk Eltawila, Senior Containment Systems Engineer in the Containment Systems Branch of the Division of Systems Integration. Eltawila, et al., ff. Tr. 9741.

21.6 In the Shoreham Safety Evaluation Report, § 6.2.1.8, the Staff concluded that, based on an assessment of the Shoreham load specifications in terms of the generic acceptance criteria set forth in NUREG-0487, the dynamic loads utilized by the Applicant were conservative and therefore acceptable except in a few areas where the generic criteria had not been finalized or the staff review had not been completed. These areas included the Steam Condensation Downcomer Lateral Loads, the Steam Condensation Oscillation Loads, and the Steam Condensation Chugging Loads. Eltawila, et al., ff. Tr. 9741, at 4.

21:7 However, in Supplement 1 of the Shoreham Safety Evaluation Report, NUREG-0420, dated September 1981, it was concluded that the Applicant's specifications for assessing all the suppression pool dynamic loads were conservative and therefore acceptable. The Staff based the conclusion on its independent review of the Applicant's submittals, the 4TCO test data, the Karlstein test data and the JAERI test data. Furthermore, NUREG-0808 was issued setting forth the final generic load specifications for Mark II containments, developed from the Mark II reassessment program that has been underway for several years. Id., at 5. There was no testimony on the record to contest the validity of the Staff resolution of the Steam Condensation Downcomer Lateral loads and the Steam Condensation Oscillation loads as set forth in NUREG-0808.

21:8 A report prepared by a Staff consultant suggested that there was an area which required further review: the generic chugging load definition set forth in NUREG-0808. Id., at 5. Following a pool swell transient, there will be a period of high steam flow through the main vent system. At these high steam flow conditions, the water/steam condensation interface oscillates due to bubble growth and collapse. These condensation oscillations result in an oscillatory load on the pool boundary. At low vent flow rates, the water/steam condensation interface can oscillate back and forth in the vents causing "chugging." The chugging action results in loads on both the downcomer vents and the containment boundaries. Staff Ex. 2A, § 6.2.1.8, at 6-19.

21:9 The concern raised by the Staff consultant was a possible lack of conservatism in the established chugging load specifications due to the random selection process for the individual vent chug initiation times of both symmetric and asymmetric loading. SER Supplement, ff. Tr. 9744, at Attachment 1.

21:10 Further review of the specifications for this chugging load was consequently initiated by Staff and the Mark II owners' group. This review consisted of a two-step approach: First, they showed that containment response to the asymmetric chugging load specification was not significantly different from that for the symmetric specification. In fact, the comparison showed them to be remarkably similar. Id. Second, they applied the symmetric specification to the JAERI facility with 20 different sets of start times and showed that the calculated wall pressures were for the most part greater than the pressures recorded during some of the biggest chugs in the JAERI facility. Id. Hence, the Staff and its consultant concluded that the Mark II Owners' Group approach toward resolution of the chug start time concern was sound and no modification to the load specifications (both generic and plant unique) was required. Id. Thus, the loads specified in NUREG-0808 (against which Shoreham is assessed) are sufficiently conservative and adequate based on this test data. Tr. 9795 (Eltawila).

21:11 A further concern raised by the ACRS regarding the Mark II containment design capability was a concern regarding the potential pool bypass from stuck open wetwell-to-drywell vacuum breakers that might be caused by repeated and strong dynamic underpressure in the vent pipe due to intermittent steam condensation. Eltawila, et al., ff. Tr. 9741, at 6.

21:12 To address this concern, the Mark II owners, including the Shoreham applicant, engaged in a joint qualification test program to demonstrate the operability of the vacuum breaker under this intermittent steam condensation loading. Further, the Applicant has implemented a design modification involving the blocking of the downcomers on which the vacuum breakers are installed. Id., at 7. This design modification will eliminate the dynamic pressure exerted on the vacuum breaker and, hence, the concern over a potential stuck open breaker is resolved. This modification is therefore acceptable to resolve the concern. Tr. 9806 (Eltawila).

21:13 One other issue relative to the adequacy of the Mark II containment was brought out in the record, even though not specifically referenced in the contention. Mr. John Humphrey, a former employee of General Electric, raised a number of concerns related to the Mark III containment design. Testifying on this issue for the Staff was Mr. Mel B. Fields, a Containment Systems Engineer in the Containment Systems Branch of the Division of Systems Integration. Most of the Humphrey concerns were inapplicable to the Mark II design or posed no issue of significance for Mark II design considerations because the effect of the concern could be roughly quantified and the margins inherent in the Mark II design were adequate to accommodate the potential effect of the concern. Tr. 9857 (Fields). The only issue raised by Mr. Humphrey which required detailed analysis for the Mark II design involves the residual heat removal discharge mode when in the steam condensation mode. Tr. 9855 (Fields).

21:14 The Staff analysis of this issue requires that if the system should be operated in the steam condensing mode, the effects of the discharge into the suppression pool must not disable any safety-related equipment. Tr. 9858 (Fields). However, it is not possible to operate the RHR in the steam condensation mode at 5% or less power and the Staff will complete its confirmation analysis of this concern prior to full power operation. Tr. 10,019 (Fields). Further, while the Staff does not believe the design margins will be eroded in this analysis, the simple solution of adding a quencher at the end of the RHR line will take care of the load if the load is found to be excessive. Tr. 10,020 (Fields).

21:15 This Board finds, therefore, that evidence on the record shows forces generated during suppression pool LOCA dynamics have been adequately determined and taken into account. The Mark II reassessment program and the Staff review, coupled with the modifications implemented by the Applicant as a result of this review, have confirmed that the specifications used by the Applicant for assessing these loads are sufficiently conservative and therefore acceptable. Additionally, we find the record shows the issue of the RHR discharge mode is inapplicable at 5% or less power and there is no reason to consider this an issue for a low power license.

Contention 21(c)

21:16 Contention 21(c) asserts that the test procedure to demonstrate an acceptable leakage rate of leakage paths between drywell and the wetwell may be inadequate. Testifying on this issue for the Staff was Mr. Mel B. Fields. These test procedures consist of pressuring the drywell and measuring the pressure decay inside the drywell over time. In the single preoperational high pressure leakage test, the inlet of the downcomers will also be capped. The leakage rate is then compared with the acceptance criterion to determine if the test results are acceptable. The drywell floor has been designed to accommodate thermal and pressure loads under LOCA conditions without cracking and forming new leak paths. The acceptance criterion for the leakage tests is set equal to 10% of the leakage rate needed for the pressure inside containment to equal the design base pressure following the most limiting LOCA. These factors, plus the fact that steam under accident conditions will not leak as fast through a leak path as the air used in the test procedure (due to the condensation of steam inside potential leak paths), make the test procedures adequate and capable of performing the function of assuring that excessive steam bypass will not occur. Eltawila, et al., ff. Tr. 9741, at 9.

21:17 These tests have been performed at an operating Mark II, La Salle, and the results indicated a much lower leakage rate than that specified in the acceptance criteria. Tr. 9865 (Fields).

21:18 Hence, this Board finds the test procedures to demonstrate an acceptable leakage rate for leakage paths between the drywell and the wetwell are adequate and capable of assuring that excessive steam bypass will not occur.

Contention 21(d)

21:19 Contention 21(d) asserts that the Mark II containment design has not been adequately shown to accommodate combined transient and LOCA events loads. Both Applicant and Staff addressed this concern in their direct testimony. Testifying for the Staff was Mr. Chen P. Tan, Senior Structural Engineer in the Structural Engineering Branch in the Division of Engineering.

21:20 As discussed in Contention 21(a), the generic load specifications for Mark II containments is contained in NUREG-0808, which includes load combinations of transient and LOCA events. In Revision 5 to Shoreham DAR, December 1981, the Applicant evaluated the design against the loads identified in NUREG-0808. Tr. 9846 (Eltawila).

21:21 The Applicant committed to using the final load specifications in NUREG-0808, and these have been implemented in the design. Tr. 9787 (Eltawila). Specifically, as a result of this evaluation, design modifications to steel structures in the plant were made. These modifications included the installation of steel and concrete shear rings along the reactor pedestal; the downcomer bracing system was lowered and strengthened; structural steel in the drywell was strengthened; and steel framing in the secondary containment was strengthened. Chau, et al., ff. Tr. 9735, at 27-28. These modifications and loads, addressed in Revision 5 of the DAR, meet the requirements of NUREG-0808. Tr. 9788 (Tan).

21:22 The acceptance criteria used in the assessment by Applicant are based on the ACI 318-71 code for concrete structures, on the ASME Section III, Division 1 and Division 2 for the containment steel liner, and on the AISC 1969 specification for steel structures. This is in conformance with the requirements delineated in the Staff's Standard Review Plan, Section 3.8. SER Supplement, ff. Tr. 9744, at Attachment 1.

21:23 This Board finds therefore that the Shoreham Mark II containment design has been adequately assessed to accommodate these combined loads and we conclude from this assessment that sufficient design margin exists to establish that the general design criteria requirements of 10 C.F.R. 50, Appendix A have been met.

Contention 21(e)

21:24 Contention 21(e) asserts that an adequate and properly controlled experimental design verification program, as requried by 10 C.F.R.

50, Appendix B, Sections III and XI, has not been performed. Additionally, Intervenor Suffolk County asserts that the verification of the design adequacy of the Mark II containment is deficient with regard to testing under the most adverse design conditions, performance of tests under suitable environmental conditions, documentation and evaluation of test results, and use of test data developed under a foreign test program. Hence, the County argues that the acceptance criteria used in evaluating the Shoreham design may not be sufficiently conservative.

21:25 Testifying for the Staff was Mr. Chen P. Tan, Senior Structural Engineer in the Division of Engineering. Testimony on the record shows that the Shoreham containment structure was subjected to a structural integrity test at 1.15 times the design pressure and the test was in accordance with Regulatory Guide 1.18 "Structural Acceptance Test for Concrete Primary Reactor Containments." Tr. 9877-78 (Tan). Thus, the requirements of 10 C.F.R. 50, Appendix B, Section XI have been met.

21:26 Additionally, the design verification program is addressed by the Applicant's commitment to a confirmatory evaluation in which it will perform a 100% reevaluation of the piping attached to the primary containment to assure that acceptance criteria utilized are sufficiently conservative. Tr. 9888 (Terao).

21:27 As to the adequacy of the acceptance criteria used in evaluating the Shoreham design, the Staff testified that the criteria used by the Applicant were acceptable, based on the Staff's independent review, the 4TCO test data, the Karlstein test data and the JAERI test data. Tr. 10,026 (Eltawila).

21:28 Furthermore, the experimental design testing program used during the Mark II assessment program was under the supervision of General

Electric whose Quality Assurance program meets the requirements of 10 C.F.R. Part 50, Appendix B. This has been audited by all the utilities involved in the Mark II containment program, including the Applicant. Tr. 10,004 (Davis).

21:29 With regard to the rest of the contention in connection with adverse design conditions, the Applicant's witnesses testified that the acceptance criteria utilized in their evaluation combined LOCA loads with seismic event loads and the probability of such a combination is so low as to render this analysis inherently conservative. Tr. 9919 (Malovrh); Eltawila, et al., ff. Tr. 7941, at 12. Applicant's witnesses further stated that this combined load analysis (seismic with LOCA events) set forth in Revision 5 of the DAR uses acceptable criteria for a worst case scenario. Tr. 9919 (Malovrh). Sufficient margin exists to accommodate the difference between the design basis and the confirmatory evaluation response spectra for the Staff to accept the criteria set by the Applicant for these combined loads. Tr. 9971 (Tan).

21:30 As it is mechanically impossible for a safe shutdown earthquake to cause a LOCA since the containment is seismically qualified for such an event, the requirement for the design analysis to address this combination is conservative. Tr. 9983 (Tan).

21:31 This combined LOCA load definition constitutes suitable environmental considerations. Tr. 9990 (Davis). Tests performed in Japan controlled by the GE QA program utilized in the Mark II design verification program produced results that confirmed the acceptability of load definitions derived from the 4TCO test data. Tr. 9991 (Davis).

21:32 Accordingly, this Board finds that the acceptance criteria used by the Applicant in evaluating the Mark II containment design is sufficiently conservative. We further find that the record shows the design verification program has taken into account a worst case scenario, environmental considerations, documented test results, and foreign test data in defining loads. There is no evidence that such definitions lack conservatism.

G. Safety Relief Valves (SC 22; SC 28(a)(vi)/SOC 7A(6))

22/28:1 SC Contention 22 states:

> Suffolk County contends that LILCO has not adequately demonstrated that the safety/relief valves to be used at Shoreham meet the requirements of 10 C.F.R. 50, Appendix A, GDC 14 and 30 and 10 C.F.R. 50, Appendix B, Sections III and XI, in that the functionability of the valves, as installed, has not been established by the generic test program results. Specifically, NUREG-0737, item II.D.1, performance testing of BWR relief and safety valves, requires that BWR SRV valves be tested to demonstrate that the valves will open and reclose under the expected flow conditions. It additionally requires that ATWS testing be considered.
>
> LILCO has not yet provided a detailed plant-specific evaluation of the Shoreham safety and relief valves, piping, and supports in accordance with the NUREG-0737 requirements. Additionally, no commitment has been made on ATWS testing. Therefore, it has not been demonstrated at this time that the specific requirements have been met.

22/28:2 SC Contention 28(a)(vi) states:

> Suffolk County/SOC contend that the NRC Staff has not adequately assessed and LILCO has not adequately resolved, both singularly and cumulatively, the generic unresolved issues applicable to a BWR of the Shoreham design. As a result, the Staff has not required the Shoreham structures, systems, and components to be backfit to current regulatory

practices as required by 10 C.F.R. § 50.55(a), § 50.57, and § 50.109, with regard to the following:

A. LILCO has failed to resolve adequately certain generic safety items identified as a result of the TMI-2 accident and contained in NUREG-0737, Clarification of TMI Action Plan Requirements

* * *

(6) LILCO hopes to accomplish a reduction in challenges to safety/relief valves (NUREG-0737, Item II.K.3.16) by procedural techniques, rather than by system modifications. But the reliability of the SRV's chosen for Shoreham has been historically poor. Thus, LILCO has not demonstrated SRV compliance with 10 C.F.R. Part 50, Appendix A, Criterion 30.

22/28:3 Testifying for the Applicant on these two contentions were the following individuals: John J. Boseman, a Senior Engineer and Technical Leader for General Electric who has been involved in the design and evaluation of SRV's for BWR's; Raymond M. Crawford, Vice President of SAI and a participant in the BWR Owners' Group SRV Performance Testing Subgroup; Jeffrey L. Smith, Manager of Special Projects, LILCO Construction and Engineering Department, who also has worked with the BWR Owners' Group SRV Performance Testing Subgroup; Fred Hayes, a licensing engineer for General Electric who has been involved with the BWR Owners' Group Program to respond to NUREG-0737, Item II.K.3.16; and John J. Kreps, a licensed operator and a startup and test engineer with EDS Nuclear assigned to Shoreham for coordination and preparation of plant operating procedures. Boseman, et al., ff. Tr. 7959, at 1-3, and Professional Qualifications. Also on the Applicant's combined panel on the contentions were: Steven J. Stark, Manager of BWR Evaluation

Programs for General Electric, who has worked with the BWR Owners' Group Program to respond to NUREG-0737, Item II.D.1; and Charles A. Malovrh, a Senior Structural Engineer at Stone and Webster, who has been responsible for analyses of piping system fluid transients. Crawford et al., ff. Tr. 7954, at 1-2, and Professional Qualifications.

22/28:4 Testifying for the Staff on these contentions was a combined panel of the following individuals: Robert J. Wright, a Mechanical Engineer from the Equipment Qualification Branch responsible for evaluation of the functional capability of safety-related equipment under all normal, abnormal, and accident loading conditions. Wright, ff. Tr. 7964, at Professional Qualifications. M. Wayne Hodges, a Section Leader from the Reactor Systems Branch. Hodges, ff. Tr. 7966, at Professional Qualifications. Frank C. Cherny, a Section Leader in the Mechanical Engineering Branch who is a member of the ASME Working Group on Safety and Relief Valves. Professional Qualifications, ff. Tr. 7967.

22/28:5 Testifying for Suffolk County on these contentions were Dale G. Bridenbaugh and Gregory C. Minor of MHB Technical Associates. Bridenbaugh Professional Qualifications, ff. Tr. 3543; Minor Professional Qualifications, ff. Tr. 1113.

22/28:6 The Shoreham plant is equipped with eleven Target Rock two-stage 6x10 type Safety Relief Valves (SRV's), Model 7567 F. The primary design function of the SRV's is to relieve excess pressure in the reactor vessel by releasing steam from the vessel to the suppression pool. The SRVs perform this function in two possible ways: (1) the valves provide automatic overpressure protection by opening when pressure reaches a designated set-point; (2) the valves are part of the automatic depressurization system (ADS), which is used in the event there is a small-break LOCA coupled with the unavailability of the high pressure cooling systems, in order to depressurize the vessel to allow low pressure emergency core cooling systems to operate. Boseman et al., ff. Tr. 7959, at 4-5; Crawford et al., ff. Tr. 7954, at 3-4.

SC Contention 22: TMI Item II.D.1

22/28:7 NUREG-0737, Item II.D.1, "Performance Testing of Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (NUREG-0578, Section 2.1.2)" states that all reactor licensees and applicants must "conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents." Crawford, et al., ff. Tr. 7954, Attachment 1. The objective of this Item is to test the performance of the SRV's under a liquid or two-phase flow conditions. Tr. 8223 (Hodges).

22/28:8 In response to Item II.D.1, the BWR Owners' Group, of which the Shoreham Applicant is a member, contracted with General Electric to develop and implement a generic SRV test program. The test conditions proposed by the Owners' Group were submitted to the NRC by letter dated September 17, 1980, D.B. Waters to R. Vollmer. The valve test program itself was completed and the final results submitted to the NRC as General Electric Report NEDE-24988-P, by letter dated September 25, 1981, T.J. Dente to D. Eisenhut. The Shoreham Applicant also submitted to the NRC, by letter dated December 9, 1981, a statement that the generic test results are applicable to Shoreham. Wright, ff. Tr. 7964, at 3-4.

22/28:9 NUREG-0737, Item II.D.1 requires that the test conditions for the generic test program be based upon a determination of the "expected valve operating conditions" for which liquid or two-phase flow through the SRV's is reasonably likely. The conditions were to be selected from the accidents and anticipated operational occurrences listed in Regulatory Guide 1.70, Revision 2. In the analysis to select the test conditions, a single failure is to be applied to the Reg. Guide 1.70 events such that dynamic forces on the SRV's are maximized. Crawford, et al., ff. Tr. 7954, Attachment 1; Tr. 8223-5 (Hodges).

22/28:10 The BWR Owners' Group Study identified the alternate shutdown cooling mode as the only single failure liquid discharge event to be tested under NUREG-0737, Item II.D.1. Crawford, et al., ff. Tr. 7954, at 6. The study took each of the Reg. Guide 1.70 accidents or occurrences and superimposed the single failure of a rupture of the valve or discharge piping. The study determined that, except for the alternate shutdown cooling event, the likelihood of the occurrence of the Reg. Guide 1.70 events was low, and that their consequences did not exceed those of the design basis accident. Tr. 8220 (Crawford). Therefore, it was concluded that testing of valves for any of the events in Reg. Guide 1.70 other than alternate shutdown cooling was not warranted. Id.

22/28:11 The NRC staff accepted the test condition chosen by the Owners' Group for the generic test program. Staff Ex. 2B, § II.D.1, at 22-44; Tr. 8172 (Hodges). The thrust of Item II.D.1 was to require analysis of the performance of SRV's under water or two-phase flow conditions for transients and operating conditions which are reasonable to expect. The alternate shutdown cooling mode event was the only one which met these criteria. Tr. 8223-25 (Hodges). A test under this condition does not include the worst stresses or the worst loads on the valve, but this is not required by the Staff for II.D.1. The worst stresses and loads would not be encountered under "expected operating conditions." Tr. 8234-35 (Hodges). The NRC staff did have a generic question as to whether the test conditions should be expanded to include liquid flow under high pressure. Tr. 8171-2 (Hodges). However, based upon the Applicant's response to Staff questions, particularly concerning the water level 8 injection trip at Shoreham, the Staff found high pressure test conditions to be unnecessary for Shoreham. Id.; See also Finding 22/28:22.

22/28:12 No tests were performed for ATWS conditions in the Owners' Group study. However, Item II.D.1 does not require ATWS testing of the SRV's for BWR's. Crawford, et al., ff. Tr. 7954, at 12. Staff witness Hodges was a member of the committee which drafted the TMI Item. It was his testimony that the intent of II.D.1 was to require ATWS test conditions only for PWR's, which is why Section (c) of the requirement uses PWR pressure conditions and the PWR compliance dates. Tr. 8445-6 (Hodges); See also, Crawford et al., ff. Tr. 7954, Attachment 1.

22/28:13 ATWS testing is not required for BWR valves for sound technical reasons. In BWR's most ATWS events would not result in an increase in water level in the vessel. The water level will tend to decrease and therefore there is no reason to expect the water to reach the steam lines. There are a minority of ATWS events which can be postulated in which there would be an increase in water level. These would require a failure of feedwater control. However, even in these cases it is still unlikely that water will reach the steam lines for several reasons. First, the reactor is still producing full power. Second, there are level 7 alarms which will alert the operator and give him a chance to correct the situation. Third, there is a level 8 trip to automatically cut off feedwater. Therefore, even in the unlikely event that the water level increases during the ATWS, it is very unlikely that the ATWS event would challenge the relief valves with a two-phase or water flow. Tr. 8444-5 (Hodges).

22/28:14 The NRC staff does not require steam testing as part of the BWR Item II.D.1 SRV test program. Tr. 8380-81 (Cherny). The GE test program did include a full flow steam test for 5 seconds. Tr. 8354 (Boseman). This was not done to verify performance of the valves at ATWS conditions, but to expose the valve to the temperature conditions that the valve would encounter before entering the alternate shutdown cooling mode. Following the 5 seconds of steam flow the valve was tested for the liquid discharge in the alternate shutdown mode. Tr. 8381 (Cherny/Smith).

22/28:15 An ATWS event occurring after main steam isolation valve closure is the ATWS event which would cause the greatest challenge to the SRV's. In this scenario, the valves would pass a full flow of steam for approximately 20 seconds. There would also be subsequent actuation to remove decay heat. The number and duration of the actuations would depend upon whether ARI is successful. Tr. 8357-61 (Stark). The NRC staff has not required that this scenario be tested for under Item II.D.1 of NUREG-0737. There are other assurances that the valves will properly perform under the conditions. Apart from testing under Item II.D.1 the valves are subjected to life cycle testing and routine in-service testing. Furthermore, the SRV's have successfully actuated many times in service conditions very similar to the worst case ATWS scenario. Tr. 8379-80 (Cherny). The difference between a 20 second full steam test and the 5 second test used by GE in also not significant. Once the valve has opened on set point, it will remain open until pressure falls below set point, independent of the time the valve is open. Tr. 8376-77 (Smith/Boseman).

22/28:16 The environmental conditions in an ATWS scenario are bounded by the limiting non-ATWS event identified in the Shoreham FSAR. Pressure would reach approximately 1300 psig. The Shoreham SRV's are permitted by ASME Code Section III to operate at a peak pressure of 1375 psig during 1% of their service time. Crawford, et al., ff. Tr. 7954, at 12-13. For Shoreham all the valve electronic control systems are qualified for the limiting LOCA conditions. Tr. 8393-4 (Smith). Under the Item II.D.1 test program the valves and control systems were tested in an environmental cage to simulate these conditions. Tr. 8394 (Boseman).

22/28:17 The Applicant has considered the applicability of the BWR Owner's Group test of SRV's to the Shoreham specific configuration of valves, discharge piping, and supports. NEDE-24988-P contains the results of pipe and support load measurements performed for both operating pressure steam and low pressure liquid test conditions. In all cases the loads measured for water discharge conditions were considerably lower than under steam conditions. The NRC staff has accepted the design of the Shoreham SRV piping and supports for steam flow conditions. Therefore, the Applicant concluded that the generic test program is sufficient to demonstrate the adequacy of Shoreham's piping and supports for liquid flow conditions. Wright, ff. Tr. 7964, at 4.

22/28:18 The NRC staff made a Request for Information, A. Schwencer to M.S. Pollock, July 8, 1982, consisting of six questions focusing on the applicability of the General Electric test program to Shoreham. SC Ex. 34 for identification, ff. Tr. 8312. The Applicant responded to these questions on July 29, 1982, in a written submittal on the hearing docket. ff. Tr. 8402. The Staff witnesses reviewed the submittal and stated their position on the record.

22/28:19 Question 1: In the GE test program a "ram's head" discharge pipe configuration was used, while Shoreham and most BWRs use "tee" quenchers at the end of the discharge line. The Staff was concerned about possible effects of the piping difference on valve loads. Tr. 8404-5 (Cherny). The Staff found acceptable the response that the loads placed on the valves in the generic test are greater than those which will be placed on the Shoreham valves, regardless of the discharge piping configuration. Tr. 8407-8 (Wright). This is because the loads on the first segment of piping downstream from the valve are the most significant to loads on the valve. The load on that segment is directly related to the length of the segment. The length of the segment at Shoreham is substantially shorter than that in the test facility. Given that the discharge and the pressures in the test configuration are the same as those anticipated for Shoreham, the loads at Shoreham will be less than those experienced in the generic test. Tr. 8559-61 (Malovhr/Wright). This question is resolved. Tr. 8405 (Cherny).

22/28:20 Question 2: Shoreham's SRV discharge lines are supported by spring hangers in conjunction with snubbers and rigid supports. The test configuration did not utilize spring hangers. The Staff question, therefore, concerned whether or not this piping support difference will impact valve operability. The Applicant responded that analyses had been performed on the Shoreham discharge line configuration to verify that the Owners' Group generic test results hold true for Shoreham. The Staff found this response acceptable. The analyses showed that the loads on pipe supports from a water discharge event at Shoreham are lower than the loads from a high pressure steam discharge event. Tr. 8409-8410 (Cherny). The Applicant also committed to submit to the Staff the results of a stress analysis to confirm that any increased loads on unpinned spring hangers due to dead water weight in the pipes during the liquid discharge transient, are offset by the fact that the loads in a liquid discharge event are less than in the design basis high pressure steam transient. Tr. 8410-12 (Cherny). The question was considered resolved pending confirmation by the stress analysis. Tr. 8410 (Cherny). The Applicant submitted the stress analysis results in SNRC-812, J.L. Smith to H.R. Denton, December 15, 1982.

22/28:21 Question 3: The Staff requested information as to the existence of any valve functional deficiencies during the generic Owners' Group test program. The Staff accepted the Applicant's response that there were no anomalies encountered in the program which indicated a valve operability failure. The Applicant also clarified the Owners' Group's criteria for selection of data to be presented in the report. Tr. 8425-26 (Wright). The Staff has also reviewed a series of EPRI tests on SRV's. These tests, however, were designed for PWR valves and do not indicate any anomalies which would be expected for BWRs. Tr. 8427-9 (Cherny); Tr. 8430 (Crawford). This question is resolved. Tr. 8425 (Wright).

22/28:22 Question 4: The generic SRV test included only liquid flow during the alternate shutdown cooling. This Staff question concerned the need to test for other anticipated conditions at Shoreham -- specifically a high pressure vessel overfill event. Tr. 8431 (Hodges). Such an event would cause higher dynamic forces than the alternate shutdown cooling mode event. Tr. 8246 (Hodges). The Staff accepted the Applicant's response that testing under conditions other than alternate shutdown cooling is unnecessary. Tr. 8431-2 (Hodges); see also Finding 22/28:11. The high pressure test condition was one of the events considered from the Regulatory Guide 1.70, Rev. 2 list. However, the test condition is not required. Shoreham is equipped with a water level 8 water injection trip which contributes to a low probability of the overfill event. Tr. 8433-4 (Hodges). Furthermore, the consequences of such an event would be bounded by a steam line break analysis. Tr. 8594-5 (Hodges). This item was considered resolved pending only a submittal of verification. Tr. 8432-33 (Hodges). The Applicant included a submittal on this question in SNRC-812, December 15, 1982.

22/28:23 The Shoreham plant includes level 8 trips on HPCI, RCIC, and feedwater. Injection of water from these sources is automatically tripped when water level reaches level 8. The level 8 trip does not cover the control rod drive (CRD) hydraulics and there is a direct injection from the CRD into the vessel. However, it is not a high capacity system -- approximately 100 gpm. There is no concern that the CRD hydraulics would cause a vessel overfill and a high pressure liquid flow through the SRV's. Tr. 8596-8600 (Hodges).

22/28:24 Question 5: This question concerned the possible cycling of the valves during depressurization, and the need to test the SRVs for that condition. The Staff concluded that the question was irrelevant to the Item II.D.1 test program, because during a depressurization the valve will only experience steam conditions. Tr. 8281 (Hodges). Furthermore, based on the operating procedures and the emergency procedure guidelines, the valves would not be cycled in the alternate shutdown cooling mode which was being simulated in the SRV test program. Tr. 8439 (Hodges). In the alternate shutdown cooling mode the procedure calls for only one SRV to be open and for it to remain open. Tr. 8823 (Minor). This question is resolved. Tr. 8439 (Hodges).

22/28:25 In the generic test an orifice plate was used in the SRV discharge line, before the discharge into the suppression pool. This was done to maximize the steady-state backpressure. Tr. 8565 (Crawford). The orifice plate was located above the water level of the pool. There were no tests initiated with water in the discharge line above the orifice plate. This was done because upon entering the alternate shutdown cooling mode there will not be water in the discharge line. Tr. 8565-7 (Crawford). However, if an operator, contrary to procedures, did cycle a valve in the alternate shutdown cooling mode it is conceivable that the valve will reopen with the discharge line filled with water. In the engineering judgment of the Staff witness, such an event will not produce significantly different loads on the valve. The loads on the valves will not be determined by the water in the discharge line, but by the pressure upstream from the valve. Tr. 8572 (Hodges); Tr. 8568-9 (Hodges).

22/28:26 Suffolk County witnesses also postulated an event in the alternate shutdown cooling mode in which an operator does cycle a valve resulting in a flow of water through the valve. Tr. 8813 (Bridenbaugh). The procedures call for the operator to stabilize pressure between 100 and 184 psig, by opening a second SRV if pressure rises above 184 psig. The County witnesses suggest that the operator may cycle the second valve as pressure varies around 184 psig. Tr. 8818-20 (Minor). However, even this operator error will not result in a water flow through the valves unless the valve is cycled open while there is still water in the line. Water will drain out of the line less than 10 seconds after the SRV is closed. Tr. 8813-14 (Bridenbaugh).

22/28:27 Question 6: Through this question the Staff sought the methodology used to calculate the discharge flow coefficient for the Shoreham SRVs. The Staff found the methodology acceptable. Tr. 8440 (Wright). The flow coefficient is used to develop the operator procedures. Based on the coefficient, the procedures direct the operator on the proper pressure to be maintained for the number of open SRVs in order to ensure adequate core cooling. The operator makes no calculations. Tr. 8605-6 (Hodges). This question is resolved. Tr. 8440 (Wright).

22/28:28 In conclusion, the Applicant has satisfied the requirements of NUREG-0737, Item II.D.1. The generic SRV test program included all the necessary test conditions. Testing under ATWS conditions is not required. Furthermore, the Applicant has demonstrated, and the Staff has accepted, that the generic test program is applicable to the Shoreham plant.

SC Contention 28(a)(vi): TMI Item II.K.3.16

22/28:29 NUREG-0737, Item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves -- Feasibility Study and System Modification," seeks to reduce the incidence of stuck-open, or spuriously opening, relief valve events. The NUREG directs that the Applicant make an investiation into the feasibility of reducing challenges to the SRVs by the use of 13 suggested methods, and others. Item II.K.3.16 states: "Those changes which are shown to reduce relief-valve challenges without compromising the performance of the valves or other systems should be implemented." Boseman, et al., ff. Tr. 7959, Attachment 1.

22/28:30 In response to the TMI Item the Applicant participated in the BWR Owners' Group evaluation of the methods available to reduce challenges and failures of SRVs. The results of that study were compiled and submitted to the NRC by letter dated March 31, 1981, D.B. Waters to D.G. Eisenhut. Id., at 7-8, Attachment IV. The evaluation noted that the intent of Item II.K.3.16 was to reduce the number of stuck-open or spuriously opening SRV events (SORV's), not just to reduce the number of SRV challenges. Id., at 8. Staff witness Hodges, who was a member of the committee that drafted the Item, verified that reduction of SORV's was the goal of II.K.3.16. Tr. 8490-10, Tr. 8509; Tr. 8615 (Hodges).

22/28:31 Based on the evaluation, the Applicant selected the methods to be applied at Shoreham to reduce SORVs. These include: (1) the use of the Target Rock two-stage safety relief valve, (2) the use of an operating procedure providing for manual implementation of low-low set relief, and (3) a lowering of reclosure set point on the Shoreham SRVs. Boseman, et al., ff. Tr. 7959, at 10-11. Shoreham will also be provided with an improved pneumatic supply control system to the SRV's. Hodges, ff. Tr. 7966, at 3.

22/28:32 The Target Rock two-stage SRV was developed to improve performance and reliability over the Target Rock three-stage design. By eliminating the second stage of the three-stage design, the two-stage SRV has eliminated the major cause for spurious plant blowdowns and stuck-open relief valve events. Boseman, et al., ff. Tr. 7959, at 12-13. The change to the two-stage Target Rock valve alone does not reduce the number of SRV challenges. Tr. 8497 (Hodges).

22/28:33 A reduction in the number of challenges to the SRVs is achieved through implementation of the low-low set relief procedure. By this procedure the operator is directed to manually hold open a valve beyond the reclosure set point. The technique depressurizes the reactor such that the number of second and subsequent SRV openings is reduced. Boseman, et al., ff. Tr. 7959, at 15; Hodges, ff. Tr. 7966, at 2-3.

22/28:34 The design improvements of the two-stage Target Rock valve makes possible a lowering of the reclosure set point. At Shoreham the reclosure set point will be approximately 70 psi below the opening set point. Because of this setting, the valve will automatically remove more heat with the initial SRV actuation, thereby reducing the number of subsequent actuations of SRV's. Boseman, et al., ff. Tr. 7959, at 16.

22/28:35 The improved pneumatic supply control system to the SRV's, to be applied at Shoreham, will prevent spurious openings caused by pressure regulator malfunctions. By eliminating this specific problem a small reduction of challenges is achieved. Tr. 8503-4 (Hodges).

22/28:36 The BWR Owners' Group study shows that the first three modifictations listed for Shoreham will result in approximately a ten-fold reduction in SORV event frequency when compared to a benchmark BWR 4 with three-stage Target Rock SRVs. Boseman, et al., ff. Tr. 7959, at 16, and Attachment IV at Table 5.1. The Owners' Group did not take credit for any reduction due to the improved pneumatic supply control. Tr. 8640 (Hayes). The NRC Staff could not confirm the numbers in the report. Tr. 8503 (Hodges). However, witness Hodges testified that based on operating data for the two-stage valve there would be a reduction by a factor of 8 in the number of failures to close, based only on the change from the three-stage SRV. Mr. Hodges also estimated a 20 to 30 percent reduction in challenges to the SRV's from the other modifications. Tr. 8505-6 (Hodges).

22/28:37 The NRC Staff found these modifications to be sufficient to meet the II.K.3.16 requirements. The express language of the Item calls for a reduction in SRV challenges by "an order of magnitude." The Staff considers this to be a flexible goal which would be met by a factor of eight reduction in SORV's. Tr. 8490-92 (Hodges). However, the focus of Item II.K.3.16 is not to achieve a number goal, but to identify all modifications which reasonably could be implemented. For this reason the Staff has held open its generic review of the Owners' Group evaluation, to determine whether any further measures can be taken. Tr. 8612-13 (Hodges).

22/28:38 One measure the Staff wanted to study was a possible change on the set-point for water level for main steam isolation valve closure. Tr. 8489 (Hodges). On January 7, 1983, the Applicant submitted SNRC-816, J.L. Smith to H.R. Denton, committing to make this procedural change to further reduce the number of SRV challenges.

22/28:39 One other modification suggested by Suffolk County witnesses is an automatic low-low set relief, as opposed to the manual set implemented in Shoreham's operating procedures. Tr. 8800 (Minor). However, the manual function is highly reliable, given the explicit operator instructions. Tr. 8642 (Hayes). Furthermore, there would be the disadvantage of the loss of flexibility introduced by the automatic mode. Tr. 8643 (Smith). The County's witnesses also suggested an automatic low-low set relief with manual override. Tr. 8802 (Minor). However, the witnesses did not know whether such a system existed, or whether it would introduce overly complex logic. Tr. 8806 (Bridenbaugh); Tr. 8804 (Minor).

22/28:40 In conclusion, the Applicant has satisfied the requirements of NUREG-0737, Item II.K.3.16. The Applicant has taken adequate action to reduce the number of SRV challenges, and more importantly to reduce the number of stuck-open and spurious opening relief valve events.

Board Notification 82-79

22/28:41 On July 26, 1982, the NRC Staff issued Board Notification 82-79, "Opening Pressure of Two-Stage Target Rock Safety Relief Valves." The notification recounts operating experience of variability in the opening pressure of the two-stage Target Rock SRVs, with some valves not opening at designated set-point pressure. Tr. ff. 7968.

22/28:42 On July 3, 1982, at Hatch 1 only three of eleven two-stage Target Rock SRVs opened when the reactor system pressure rose to approximately 1200 psig. The nominal set-points for all eleven SRVs were in the 1080-1100 psig range. Id. The valves were subsequently removed and tested at Wyle labs. In the tests eight of the eleven valves opened within 1 percent of the set pressure, while the other three opened within about 2 percent of the set-point. Tr. 8051 (Cherny).

22/28:43 On July 19, 1982, Browns Ferry 2 experienced a transient in which reactor coolant pressure rose from 1050 psig to 1125 psig. In this transient, ten of eleven two-stage Target Rock SRVs opened. ff. Tr. 7968. The only valve that did not open had a set-point of 1125 psig, plus or minus a tolerance. Tr. 8067 (Boseman). This was considered a counter example to the Hatch 1 event. Tr. 8064 (Cherny).

22/28:44 In both the Hatch 1 and the Browns Ferry 2 events the pressure increase was very gradual. Tr. 8053; Tr. 8064-5 (Cherny). Based on data from Item II.D.1 performance testing and from PWRs, it is the Staff's judgment that a more rapid rate of pressurizing the system or a higher maximum pressure would have caused most or all of the valves which remained closed to open. ff. Tr. 7968; Tr. 8069-70 (Cherny). The NRC staff will continue to study the event at Hatch 1 to attempt to determine the causes. Tr. 8052-3 (Cherny). The investigation will include such possible contributors as the pressure ramp rate and the fact that the valves at Hatch had been unchallenged for eleven months. Tr. 8052 (Cherny); Tr. 8065 (Hodges).

22/28:45 "Set-point drift" is a long standing minor problem with SRVs. Both the NRC and General Electric have been aware that for all relief valves the opening pressure may vary from the design tolerance for the valve, after the valve has been in service for a period. Tr. 8118-9 (Hodges). This variance is observed in post operation SRV tests. ff. Tr. 7968, Attachment. The Shoreham technical specifications and the ASME code require a tolerance of less than 1 percent. This tolerance is conservative and is based on a fresh valve. Tr. 8120-3 (Hodges/Cherny). When this value is exceeded in the post-service testing, the tech specs require that the valve be repaired, reset, and retested prior to installation. Tr. 8127 (Hodges). Set-point drift may be due to such factors as service conditions, and is not considered a design problem for two-stage Target Rock Valves. Tr. 8149-51 (Boseman).

22/28:46 Neither the events recounted in BN-82-79 nor the problem of set-point drift alter the conclusions reached as to Shoreham's compliance with NUREG-0737, Item II.D.1 and Item II.K.3.16. Item II.D.1 is addressed to SRV response to a liquid or two phase flow. Item II.K.3.16 is addressed to stuck-open, or spurious opening relief valve events. These concerns are distinct from those related to a failure to open at designated set-point.

SRV Maintenance and Lubricant Polymerization

22/28:47 At Shoreham, station procedures will be implemented relating to operation, maintenance, testing, and surveillance of the Target Rock two-stage SRV's. These procedures will aid in assuring long-term reliable performance of the valves. LILCO Ex. 46, at 2-3, and Attachment 2. In addition, the procedures will be revised on a continuing basis to include the latest industry experience and improvements. This information will be provided to Shoreham by General Electric Company and Target Rock Corporation through "Service Information Letters" (SILs) and vendor instruction manual changes. Id., at 4. Shoreham procedures are provided to assure that the vendor or valve manufacturer recommendations for SRV operation and maintenance are incorporated into the Shoreham operation and maintenance program. Id., at 8.

22/28:48 Judge Carpenter posed a question concerning the possible polymerization of castor oil or other lubricants used in SRVs. Tr. 8483. There is evidence that castor oil used as a lubricant on SRVs can polymerize during normal reactor operation, eventually altering the castor oil into either a sticky or varnish-like substance. A small amount of castor oil has been used as a lubricant in 2-stage Target Rock valves, on an O-Ring located near the top of the pneumatic actuator. Polymerization of the castor oil has been suggested as a possible cause or contributor to high set-point drift on Target Rock 2-stage SRVs. LILCO Ex. 47, at 2-3. However, polymerization of the lubricant would only affect the valve's operation if the lubricant were to migrate to the labyrinth seal area. Because the amount of castor oil used is small, and the length and complexity of the migration path, it is considered unlikely that such migration would occur in sufficient quantities to affect performance. LILCO Ex. 48, at 4-8. Furthermore, following the Hatch 1 SRV set-point drift event, inspection of the valves at Wyle Lab failed to disclose the presence of castor oil, or other foreign substance, in the labyrinth seal area. LILCO Ex. 48, at 2-3.

22/28:49 Castor oil will not be used on the Shoreham SRVs. Although some of the Shoreham valves were manufactured with castor oil as a lubricant in the O-Ring area, SIL 196, Supplement 10, recommends removing the oil and replacing it with the lubricant DAG Dispersion No. 156. The SIL will be implemented at Shoreham. A second lubricant, Nickel Never-Seez, will also be used on the Shoreham valves. Investigations indicate that neither of these two lubricants will degrade or polymerize under the conditions they will be subject to. LILCO Ex. 47, at 4.

H. Post Accident Monitoring (SC 27/SOC 3)

27:1 SC Contention 27/SOC Contention 3 states:

> The recent Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident" details needed devices and qualifications of instruments. Shoreham is deficient in the following areas:
>
> (a) Radiation Exposure Rate Monitoring (Item 18, Table 1; Items 20 and 41, Table 2);
>
> (b) Radioactivity Concentration or Radiation Level in Circulating Primary Coolant (Item 11, Table 1; Item 14, Table 2);
>
> (c) Continuous On-Line Monitoring of Halogen in Effluent (Item 39, Table 1; Item 43, Table 2);
>
> (d) Secondary Containment Area Radiation Monitor (Item 36, Table 1; Item 17, Table 2);
>
> (e) Reactor Coolant System Soluble Boron Concentration (Item 3, Table 1; Item 4, Table 2);
>
> (f) Analysis of Primary Coolant (Gamma Spectrum) (Item 12, Table 1; Item 15, Table 2);
>
> (g) Drywell Spray Flow and Suppression Chamber Spray Flow (Items 21 and 24, Table 1; Items 23 and 23A, Table 2);
>
> (h) Standby Liquid Control System Flow (Item 28, Table 1; Item 37, Table 2);
>
> (i) Plant and Environment Radiation Monitoring (Item 40, Table 1; Item 45, Table 2);
>
> (j) Post-Accident Sampling Capability (Item 42, Table 1; Item 47, Table 2); and
>
> (k) BWR Core Thermocouples (Item 5; Table 1; Item 13, Table 2).

27:2 Prior to litigating this Contention, the parties agreed on a resolution of parts (b), (e), (f), (i) and (j). The resolution also reaffirmed the prior agreement of the parties dated June 11, 1982 wherein part (c) was withdrawn from this contention without prejudice to the right to submit this as an emergency planning contention.8/ Additionally, the parties agreed to the following stipulation:

> The parties agree that Item E-11 of Regulatory Guide 1.97, Revision 2 and SC 27(k)/SOC 3(k) will not be pursued during the litigation of SC 27/SOC 3, but rather will be pursued during the litigation of emergency planning contentions and SC 3/SOC 8, respectively. This agreement is without prejudice to LILCO's or the NRC Staff's right to argue during that litigation that the regulatory guide no longer requires fixed radiation monitors or BWR thermocouples.9/

27:3 The only remaining parts of this contention are (a) Radiation Exposure Rate Monitoring (with the exception of fixed offsite radiation monitors referred to in the above stipulation); (d) Secondary Containment Area Radiation Monitor; (g) Drywell Spray Flow and Suppression Chamber Spray Flow; and (h) Standby Liquid Control System Flow.

27:4 Testifying for the Applicant on this Contention were Mr. John A. Rigert, Systems Engineering Section and Mr. John F. Schmitt, Radiochemistry Engineer of LILCO, and Dr. Josph F. Baron, Power Engineer of Stone & Webster Engineering Corporation, and Mr. John J. Kreps, Startup and Test Engineer of EDS Nuclear Inc. Rigert, ff. Tr. 9455. Testifying for the Staff were Jerry L. Mauck, Reactor Engineer in the Instrumentation & Control Systems Branch and Mr. Charles E. Rossi, Section Leader in the Instrumentation & Control Systems Branch of the

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8/ Partial Resolution of SC Contention 27/SOC Contention 3 -- Regulatory Guide 1.97 ff. Tr. 11,677 at 1.

9/ Stipulation Regarding SC 27/SOC 3 -- Regulatory Guide 1.97 ff. Tr. 11,677 at 3.

Division of Systems Integration. Mauck, ff. 9462; Rossi, et al., ff. Tr. 9462. The Shoreham Opponents Coalition (SOC) and Suffolk County (SC) presented testimony from Richard B. Hubbard and Gregory C. Minor of MHB Technical Associates. Hubbard, et al., ff. Tr. 9587.10/

27:5 Regulatory Guide 1.97, Revision 2, published in December 1980, provides guidance for the design and qualification criteria for the instrumentation used to monitor plant environs and systems during and after an accident in a light water cooled nuclear power plant. Mauck, ff. Tr. 9462, at 2-3. The schedule for implementation of Reg. Guide 1.97, Rev. 2 is contained in SECY 82-111 dated March 11, 1982 and approved by the Commission on July 16, 1982. Chilk Letter to Dircks, attached to Rossi, et al., ff. Tr. 9462. The Commission specifically provided that the scheduling provisions of SECY 82-111 supercede the schedule in NUREG-0737:

> These recommended requirements are, therefore, to be accorded the status of approved NUREG-0737 items as set forth in the Commission's "Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses" (45 Fed. Reg. 85236, Dec. 24, 1980). In this connection, the provisions for scheduling set forth herein supersede any schedules with respect to such items contained in NUREG-0737. Accordingly, the recommended requirements should be used by the Staff and by adjudicatory boards as appropriate clarifications and interpretation of the related NUREG-0737 items.

Enclosure A to Chilk letter to Dircks, attached to Rossi, et al., ff. Tr. 9462. The scheduling provisions are as follows:

> When the basic requirements for emergency response capabilities and facilities are finalized, they

10/ Although the written testimony was jointly authored, only Mr. Minor, the primary author of the testimony, was available at the hearing for cross-examination. Tr. 9580.

> should be transmitted to licensees by a generic letter from NRR, promulgated to NRC Staff, and incorporated as regulatory requirements (e.g., in the Standard Review Plan or by regulation or Order as appropriate). The letter to licensees should request that licensees submit a proposed schedule for completing actions to comply with the basic requirements. Each licensee's proposed schedules would then be reviewed by the assigned NRC Project Manager, who would discuss the subject with the licensee and mutually agree on schedules and completion dates. The implementation dates would then be formalized into an enforceable document.

Enclosure to SECY 82-111, NRC Staff Recommendations on the Requirements for Emergency Response Capability, attached to Rossi, et al., ff. Tr. 9462; see also SECY 82-111, at 2. The generic letter from NRR referenced was transmitted in December 1982.

27:6 SECY 82-111 recommeded the establishment of a number of requirements for emergency response capability. SECY 82-111 further recommended that these requirements "be applicable to licensees of operating nuclear power plants and holders of construction permits for nuclear power plants." Enclosure to SECY 82-111, NRC Staff Recommendations on the Requirements for Emergency Response Capability, at 1, attached to Rossi, et al., ff. Tr. 9462.

27:7 At the time it approved the adoption of the recommendations contained in SECY 82-111 as a supplement to NUREG-0737, the Commission intended to publish a Policy Statement explaining its action. A draft policy statement enclosed by the Commission with its approval of SECY 82-111 made it clear that these new supplemental items to NUREG-0737 "should be accorded the status of approved NUREG-0737 items as set forth in the December 24, 1980 Statement of Policy. Litigation of the recommended requirements set forth in NRC Staff Recommendations on the Requirements for Emergency Response Capability should be permitted in operating license proceedings under the same conditions as those applicable to NUREG-0737 items in accordance with the December 24, 1980 Statement of Policy." Enclosure C to Chilk Letter to Dircks, at 2, attached to Rossi, et al., ff. Tr. 9462.

27:8 The Shoreham plant has been reviewed in accordance with the Standard Review Plan which contains the analyses of design basis events in Chapter 15, to insure that sufficient indications are available for the operator to cope with design basis events. The analyses included in Chapter 15 have been developed over a number of years, are reviewed by the Advisory Committee on Reactor Safeguards, and contain input of experience from actual events that have occurred. Hence, the design basis events analyzed in Chapter 15 are considered to be very conservative events and if the plant is designed to handle such events there is reasonable assurance operation of the plant is sufficiently safe from the standpoint of public health and safety. Tr. 9488-89 (Rossi).

27:9 The four specific instruments in dispute in this contention would be useful, but not necessary, for coping with design basis events. Tr. 9490, (Rossi). The inherent conservatism in design basis events lowers the probability of a situation arising (a severe accident scenario beyond a design basis event) where the instruments recommended in Reg. Guide 1.97 would be needed. Tr. 9492 (Rossi).

27:10 Additionally, a review to the Standard Review Plan assures that a plant does meet the basic requirements of GDC 13 and 64. Reg. Guide 1.97, Rev. 2 provides for further improvements that can be made, but this goes beyond the basic GDC requirements. Tr. 9531 (Rossi). Hence, there is reasonable assurance the plant can be operated without undue risk to the health and safety of the public until the implementation of Reg. Guide 1.97. Tr. 9534 (Rossi).

27:11 The County raised a concern that the Standard Review Plan against which Shoreham was assessed was too old to be reliable and would not have the same content as a current standard review plan. Tr. 9621 (Minor).

27:12 Both the Applicant and the Intervenor Suffolk County addressed the current status of the Applicant's commitment to the requirements contained in Regulatory Guide 1.97 and noted that the Applicant has proposed deviations or alternatives with regard to the four items in this contention. The Staff has not reviewed the Applicant's position in this regard since such review is premature under the terms of SECY-82-111. Tr. 9618 (Rossi).

27:13 The County was concerned that delay in implementation of Reg. Guide 1.97 might result in the interim period becoming permanent if Applicant's current position is subsequently approved by the Staff after its review. The County also pointed to the fact that the requirements in Reg. Guide 1.97 were developed in response to the TMI accident. The County suggested that implementation of the requirements is necessary because delay in implementation would allow Shoreham to achieve the same level of fuel exposure equivalent to 90 days full power operation as experienced at TMI without meeting the requirements of Reg. Guide 1.97. Tr. 9598 (Minor).

27:14 However, the implementation schedule for Reg. Guide 1.97 was developed over a period of years and took into consideration the continued operation of existing plants. Plants were considered to be allowed to continue to operate until the Reg. Guide could be implemented. Tr. 9533 (Rossi). The implementation schedule was also reviewed by the Committee to Review Generic Requirements and the Commission. The Commission ultimately approved the schedule in SECY 82-111. Tr. 9533 (Rossi).

27:15 Once the Reg. Guide has been implemented in accordance with the terms of SECY 82-111, the Staff will perform a formal audit review at which time licensees will have to comply with the Reg. Guide items or provide justification for deviations or alternatives. Tr. 9481 (Rossi).

27:16 The interim operation of plants, until implementation of Reg. Guide 1.97, Rev. 2, has been approved by the Commission in publishing the schedule for implementation in SECY 82-111, and the record supports a finding that the implementation schedule for the requirements contained in Reg. Guide 1.97 will not result in any undue risk to the public health and safety.

IV. CONCLUSIONS OF LAW

A. The Board has considered all of the evidence submitted by the parties on the contentions covered by this partial initial decision. Based on the findings of fact set forth in Part III above, which are supported by reliable, probative and substantial evidence as required by the Administrative Procedure Act and the Commission's Rules of Practice, the Board concludes that:

(1) The Applicant has met its burden of proof with respect to each of the following contentions -- SC 4; SC/SOC 7B, SOC 19(b); SC 10;

SC 11; SC 16; SOC 19(e); SC 21; SC/SOC 22, SC 28(a)(vi)/SOC 7A(6); and SC 27/SOC 3; and

(2) As regards those aspects of water hammer, safety classification and systems interaction, ECCS core spray, passive mechanical valve failure, ATWS, seismic design, Mark II containment, safety relief valves, and post accident monitoring that are in controversy in this proceeding, there is reasonable assurance that the Shoreham Nuclear Power Station, Unit 1 can be operated without endangering the health and safety of the public.

B. The Board has considered all of the evidence submitted by the parties on the generic unresolved safety issues identified in Part I above. Based on the findings of fact set forth in Part III above and in Part III of the NRC Staff's Proposed Opinion, Findings of Fact and Conclusion[s] of Law on Unresolved Safety Issues in the Form of a Partial Initial Decision (Nov. 2, 1982), this Board also concludes that:

(1) The Staff has adequately considered the impact on Shoreham of generic unresolved safety issues A-1, A-8/A-39, A-9, A-10, A-11, A-17, A-31, A-36, A-40, A-42, A-43, A-44, A-45, A-47 and A-48, and, notwithstanding the pendency of these issues, there is reasonable assurance that Shoreham can be operated without posing an undue risk to the health and safety of the public.

C. The Board concludes that as to the matters decided here, the Director of Nuclear Reactor Regulation is authorized, upon making requisite findings with respect to matters not resolved in this partial initial decision, to issue to the Applicant a license to operate the Shoreham Nuclear Power Station, Unit 1.

V. ORDER

WHEREFORE, IT IS ORDERED, in accordance with 10 C.F.R. §§ 2.760, 2.762, 2.785, and 2.786, that this partial initial decision shall become effective and constitute the final action of the Commission thirty (30) days after the date of its issuance, subject to any review pursuant to the above cited regulations.

Exceptions to this partial initial decision or designated portions of it must be filed within ten (10) days after service of the decision. A brief in support of the exceptions must be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC staff). Within thirty (30) days of the filing and service of the brief of the appellant (forty (40) days in the case of the NRC staff), any other party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

____
Lawrence Brenner, Chairman
ADMINISTRATIVE JUDGE

____
Dr. James H. Carpenter, Member
ADMINISTRATIVE JUDGE

____
Dr. Peter A. Morris, Member
ADMINISTRATIVE JUDGE

Bethesda, Maryland
____________________, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322
	)	(OL)
(Shoreham Nuclear Power Station,	)	
Unit 1)	)	

NRC STAFF PROPOSED OPINION, FINDINGS OF FACT,
AND CONCLUSIONS OF LAW IN THE FORM
OF A PARTIAL INITIAL DECISION

David A. Repka
Richard J. Rawson
Counsel for NRC Staff

February 11, 1983

VOLUME TWO OF TWO
(Contentions 7B and 19(b))

VOLUME TWO
TABLE OF CONTENTS

Page

Page

Page

Explanatory note: The Staff has used asterisks to call attention to specific portions of the proposed initial decision which the Staff preliminarily believes may be affected by the February 9, 1983 Affidavit of James H. Conran, Sr. regarding his testimony on these contentions. In the findings of fact, the Staff has placed an asterisk behind the particular record citation which may be affected. In the opinion, the Staff has marked with an asterisk any citation to a finding of fact which contains an asterisk.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 O.L.
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

NRC STAFF'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A PARTIAL INITIAL DECISION

I. OPINION

A. INTRODUCTION AND BACKGROUND

Intervenors Suffolk County ("SC" or "the County") and Shoreham Opponents Coalition ("SOC") proferred for litigation in this proceeding several contentions raising related issues concerning the safety classification and analysis of structures, systems and components at Shoreham Nuclear Power Station. (Finding 7B:1). SOC Contention 7B(1) and SC Contention 29 alleged that event tree and fault tree logic such as that used in NRC's Interim Reliability Evaluation Program ("IREP") must be applied at Shoreham in an analysis of the reliability of systems which prevent or mitigate accidents, in contrast to what has been done in the past in the licensing of nuclear power reactors. SOC Contention 7B(2) and SC Contention 7 contended that a Shoreham-specific systems interaction analysis was required to assure that adverse interactions had been identified. In SOC Contention 7B(4) and SC Contention 6, Intervenors asserted that, in the absence of a systematic event tree/fault tree accident sequence analysis for Shoreham, there could be no assurance that all items "important to safety" as that term is used in General Design Criteria 1 had been properly classified and appropriate design and quality assurance standards applied. SC Contention 6 also alleged that a proper classification analysis would include a review of the Shoreham emergency operating procedures to ensure proper classification of all equipment relied upon in the procedures.

Long Island Lighting Company ("LILCo") and the NRC Staff ("Staff") both argued against the admission of these contentions on the grounds that they were barred by the Commission's Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses, CLI-80-42, 12 NRC 654 (December 8, 1980), which provided guidance on the extent to which issues arising out of the reviews of the Three Mile Island, Unit 2 accident may be litigated in individual operating license proceedings. In a Memorandum and Order dated March 15, 1982, this Board confirmed rulings it had made at a prehearing conference of March 9 and 10, 1982 and overruled the objections of LILCo and the Staff to the admission of these contentions. The Board found the contentions as submitted to be too vague to put the parties and the Board on notice as to which plant systems were allegedly inadequate or improperly classified. However, the Board held that the contentions raised a litigable issue as to whether the historic methodology applied by LILCO and the Staff in the design and review, respectively, of Shoreham was adequate to assure adequate protection against accident sequences which should be considered. Accordingly, the Board reformulated contentions SOC 7B(1),(2) and (4), SC 29, SC 7 and SC 6 into the following contention which was admitted for litigation:

> "LILCo and the Staff have not applied an adequate methodology to Shoreham to analyze the reliability of systems, taking into account systems interactions and the classification and qualification of systems important to safety, to determine which sequences of accidents should be considered within the design basis of the plant, and if so, whether the design basis of the plant in fact adequately protects against every such sequence. In particular, proper systematic methodology such as the fault tree and event tree logic approach of the IREP program or a systematic failure modes and effect analysis has not been applied to Shoreham. Absent such a methodological approach to defining the importance to safety of each piece of equipment, it is not possible to identify the items to which General Design Criteria 1, 2, 3, 4, 10, 13, 21, 11, 12, 24. 29, 35, 37 apply, and thus it is not possible to demonstrate compliance with these criteria."

(Finding 7B:1).

The Board also took steps to place limitations on the scope of the litigation of the reformulated Contention 7B in recognition of its breadth. Intervenors were required to prefile their testimony first and to present their testimony at hearings before LILCo and the Staff were required to prefile their respective testimony. Further, the Board stated that Intervenors would be limited to a maximum of three examples of plant design which, in their view, would illustrate the inadequacy of the methodology applied in the plant design and review.

Several events subsequent to the admission of Contention 7B resulted in a substantial expansion of the scope of the litigation under Contention 7B. First, during Intervenors' discovery, LILCo was requested to produce a copy of a draft of the probabilistic risk assessment ("PRA") study which LILCo had voluntarily undertaken for Shoreham. LILCo declined to produce the document, Intervenors moved to compel production, and this Board granted Intervenors' motion. The Board warned, however, that it had no intention of sitting for lengthy testimony on the specific details of the draft PRA. The prefiled testimony of Intervenors and, to a much greater extent, LILCo did discuss the Shoreham PRA and its relation to the contention. The Board decided to permit the introduction of most of this testimony.

Second, Intervenors decided to combine their case on SOC Contention 19(b) with that on Contention 7B.[1/] Because of the close relation between these

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1/ SOC Contention 19(b) reads in full as follows:

> "SOC contends that the NRC Staff has not required LILCo to incorporate measures to assure that Shoreham conforms with the standards or goals of safety criteria contained in recent regulatory guides. As a result, the Staff has not required that Shoreham structures, systems and components be backfit as required by 10 C.F.R. § 50.55a, § 50.57, and § 50.109 with regard to:
>
> (b) Regulatory Guides 1.26 and 1.29. -- LILCo's general list of quality group and seismic design classifications listed in FSAR Table 3.2.1-1 is not in compliance with 10 C.F.R. Part 50, Appendix A, Criteria 1 and 2, 10 C.F.R. § 50.55a, and 10 C.F.R. Part 100, Appendix A in that:
>
> (1) the quality group classifications contained in FSAR Table 3.2.1-1 do not comply with the regulatory position of Revision 3 of Regulatory Guide 1.26 for safety-related components containing water, steam or radioactive materials;
>
> (2) the seismic design classifications contained in FSAR Table 3.2.1-1 do not comply with the regulatory position of Revision 3 of Regulatory Guide 1.29 with regard to control room habitability and radioactive waste systems;
>
> (3) LILCo has not revised the FSAR Table 3.2.1-1 to expand the list of safety-related equipment as reflected in NUREG-0737 and as a result of the NRC Staff review of the Q-list as set forth in Supplement 1 of the SER on page 17-1; and
>
> (4) LILCo's list of safety-related equipment contained in FSAR Table 3.2.1-1 does not include equipment upon which the plant operators will rely in response to accidents outlined in the Shoreham emergency operating procedures."

(Finding 7B:2)

contentions, the Board permitted this consolidation and LILCo and the Staff shaped their prefiled testimony accordingly. Third, Intervenors' prefiled testimony went beyond the three systems permitted by the Board's March 15, 1982 Memorandum and Order to raise issues concerning the adequacy of the classification of several additional systems, albeit briefly, and also questioned the adequacy of LILCo's summary classification table, Table 3.2.1-1 of the Final Safety Analysis Report ("FSAR"). LILCo and the Staff moved to strike these and certain other portions of Intervenor's prefiled testimony. After argument by the parties, the motions were denied. Tr. 1093-1103. The prefiled testimony of LILCo and the Staff addressed these additional systems.

Hearings on Contention 7B (and SOC Contention 19(b)) were held on May 4-7, June 15-18, June 22-25, July 6-9, July 13-16 and July 21-22. Intervenors, LILCo and the Staff each presented a panel of witnesses; a total of twenty witnesses was heard by the Board during those twenty-two hearing days. (Finding 7B:3).

B. SUMMARY OF AFFIRMATIVE CASES PRESENTED

Intervenor's case on Contention 7B consisted of the testimony of a panel of four witnesses: Gregory C. Minor, Richard B. Hubbard, Marc W. Goldsmith and Susan J. Harwood. Mr. Minor and Mr. Hubbard are vice-presidents of MHB Technical Associates, an engineering and consultant firm. Both Mr. Minor and Mr. Hubbard are engineers with experience in the nuclear industry at General Electric. Mr. Goldsmith and Ms. Harwood are president and a research engineer, respectively, of Energy Research Group, Inc., an energy consulting firm. Both Mr. Goldsmith and Ms. Harwood are nuclear engineers. (Finding 7B:4).

Intervenors' testimony attempted to demonstrate that deficiencies exist in the methodology utilized by LILCo in the classification of structures, systems and components. Three particular arguments were raised. First, Intervenors' experts examined Table 3.2.1-1 of the FSAR and pointed to certain alleged inconsistencies and shortcomings of the table. Second, the testimony compared the equipment relied upon by the Shoreham Emergency Operating Procedures with that relied upon in the FSAR Chapter 15 analysis of design basis events. Third, the testimony gave several examples of systems that allegedly failed to satisfy applicable classification criteria. These examples included the standby liquid control system, the turbine bypass, the reactor core isolation cooling system, the level 8 trip and the rod block monitor.

Intervenors' experts further testified that an incomplete methodology had been utilized by LILCO for detecting and analyzing systems interactions which could adversely affect plant safety. The water level indication system was discussed at length as an example of a system which could be adversely affected by interactions with other systems or equipment to the detriment of plant safety.

Intervenors faulted LILCo for its alleged failure to utilize what Intervenors' experts considered improved techniques for safety classification, such as PRA, failure modes and effects analyses, systems interaction analyses and dependency analyses. According to Intervenors, in the absence of the application of such methods, LILCo may not have properly recognized, classified and treated all structures, systems, and components which are important to safety. In Intervenors' view, compliance with the General Design Criteria cannot be demonstrated given these inadequacies.

LILCo presented a panel of nine witnesses on Contention 7B. Robert M. Kascsak is the Nuclear Systems Engineering Divison Manager at LILCO. Mr. Kascsaks' education and experience are in the areas of mechanical and nuclear engineering. George F. Dawe, George Garabedian and Paul W. Rigelhaupt are from Stone & Webster Engineering Corporation ("S&W"), the architect-engineer for Shoreham. Mr. Dawe, Supervisor of Project Licensing, has over 15 years experience in the nuclear power field and demonstrated extensive knowledge of and familiarity with the Shoreham plant. Mr. Garabedian, a Senior Power Engineer, also has been involved for several years with the Shoreham project. Mr. Rigelhaupt, an Assistant Engineering Manager at Stone & Webster, has lengthy experience in chemical and nuclear engineering. David J. Robare and Pio W. Ianni are employees of General Electric Company ("GE"), the nuclear steam supply system vendor for Shoreham. Mr. Robare, the Manager of BWR 4/5 Projects Licensing, has been responsible for the licensing of Shoreham for GE since 1975. Mr. Ianni, the Manager of Nuclear Systems Performance Engineering, has been employed by GE since 1951 and is presently responsible for directing overall BWR performance evaluations. Paul J. McGuire, a consultant to LILCO from United Energy Services Corporation, has been a certified senior reactor operator and Plant Manager at Pilgrim Station. Edward T. Burns, from Science Applications Inc. ("SAI"), is the lead analyst for the Shoreham PRA; Dr. Burns has extensive experience in engineering analysis and logic model construction for BWR PRA work. Finally, Vojin Joksimovich of NUS Corporation is a member of the peer review group for the Shoreham PRA; Dr. Joksimovich is a nuclear engineer with many years of experience in nuclear power risk assessment techniques. (Finding 7B:5).

LILCo's witnesses testified that LILCO and its contractors had applied a proper, well-established and accepted methodology to the design and classification of structures, systems and components at Shoreham. This methodology, which is the basis on which plants have consistently been licensed, involves compliance with the deterministic criteria contained in NRC regulations, industry standards, the Staff's Standard Review Plan and regulatory guidance documents. The design quality control and quality assurance standards of General Electric and Stone & Webster applicable to both safety-related and nonsafety-related items were described at length. The witnesses testified that a large body of knowledge and experience, reflected and documented in NRC regulations, regulatory guides and industry standards, was applied at Shoreham and that those sources of information and guidance are themselves developed through a systematic approach to nuclear plant design and classification. The application of these deterministic standards was said to provide assurance that plant equipment has been analyzed and classified properly.

LILCo's experts addressed Intervenors' evidence concerning Table 3.2.1-1, the emergency operating procedures and the specific systems cited by Intervenors' witnesses. The conclusion was drawn that no inadequacy in the methodology for classification of structures, systems and components had been identified, as shown by a detailed examination of several systems. LILCo further addressed the analysis of systems interactions at Shoreham and presented evidence that several types of systems interactions studies had been performed for Shoreham, some of which utilized the methodologies highlighted by Intervenors' testimony (i.e., PRA, failure modes and effects analyses, walkdowns). In particular, the cited interactions concerning water level indication were addressed by LILCo's testimony both in terms of the adequacy of the methodology used and in terms of the lack of any impact on public health and safety.

Finally, LILCO cited the PRA it had voluntarily undertaken for Shoreham in arguing that it had systematically utilized the methodologies cited by Intervenors and that systems interactions had been systematically analyzed. LILCo stressed that the PRA was not a regulatory requirement and that compliance with the Commission's regulations could be and had been demonstrated without reference to the Shoreham PRA. LILCO's testimony concluded that a systematic methodology had been utilized for the analysis and classification of structures, systems and components at Shoreham, and that compliance with all applicable regulatory requirements had been demonstrated.

The Staff's panel on Contention 7B originally consisted of six witnesses, and a seventh was later added. Themis P. Speis was, at the time of the testimony, Assistant Director for Reactor Safety in the Division of Systems Integration;[2/] much of the review for Shoreham was completed under the supervision of Dr. Speis. Walter P. Haass was, at

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2/ After completion of the testimony on Contentions 7B and 19(b), Dr. Speis was named Director of the Division of Safety Technology, Office of Nuclear Reactor Regulation.

the time of the testimony, Branch Chief of the Quality Assurance Branch,[3/] and has had oversight responsibilities for portions of the Shoreham review. Marvin W. Hodges is a Section Leader in the Reactor Systems Branch; Mr. Hodges conducted portions of the Shoreham review. C.E. Rossi is a Section Leader in the Instrumentation and Control Systems Branch; Dr. Rossi was responsible for portions of the Shoreham review. James H. Conran, Sr. is a Principal Systems Engineer in the Systems Interaction Section, Reliability and Risk Assessment Branch; Mr. Conran is knowledgeable on the subjects of safety classification terminology and the Staff's systems interaction program. Robert Kirkwood is a Principal Mechanical Engineer in the Mechanical Engineering Branch, and had responsibility for the review of the classification of the safety-related structures, systems and components at Shoreham except for electric and electronic equipment. Finally, Ashok C. Thadani was added to the panel after testimony had begun. Mr. Thadani, Branch Chief of the Reliability and Risk Assessment Branch, addressed questions which the Board had raised concerning PRA and systems interaction issues. (Finding 7B:6).

The Staff testified, as Applicant had, that a systematic methodology had been applied to the analysis and classification of structures, systems and components through the use of the Standard Review Plan and various regulatory guidance documents and the accumulated experience and

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3/ In a recent reorganization, the Quality Assurance Branch was moved from the Office of Nuclear Reactor Regulation to the Office of Inspection and Enforcement; Mr. Haass is now Deputy Branch Chief of the Quality Assurance Branch in the Division of Quality Assurance, Safeguards and Inspection Programs.

judgments they represent. This systematic methodology has been used for the licensing of all operating plants. The Staff explained this methodology and demonstrated its application to the several systems cited by Intervenors' witnesses.

The Staff testified that Shoreham could be licensed for operation despite the pendency of Unresolved Safety Issues A-17 and A-47 relating to systems interactions. Staff's witnesses discussed the status of generic programs relating to those issues and explained why Shoreham could be operated safely.

The Staff also testified that the alternative methodologies proposed by Intervenors were not required by the Commission's regulations or by Staff practice, and that the application of these methodologies for the analysis and classification of structures, systems and components was not necessary in order to ensure adequately that there is no undue risk to public health and safety in the operation of Shoreham.

In rebuttal testimony, the Staff focused on one significant area of disagreement with the Applicant. LILCo's witnesses acknowledged that they had not used the term "important to safety" in the classification of structures, system and components at Shoreham (Finding 7B:44) but argued that the results in term of plant design and construction were no different than would have been the case had the term been used (Finding 7B:131). The Staff's witnesses testified that there appeared to be close agreement between LILCo and the Staff on the substantive issues involved and that they were not aware of any area in which the difference over language had actually made a substantive difference at Shoreham (Finding 7B:131*); the Staff took the view, however, that LILCO's failure to have made certain commitments for the future at Shoreham in language meaning what the Staff understood it to mean would create the potential for divergence from full regulatory compliance in the operation of Shoreham. (Finding 7B:136). The Staff filed rebuttal testimony through Mr. Conran on this point (Finding 7B:6) and that testimony was explored at length in cross-examination by the parties and by questioning from the Board.

C. STATEMENT OF MATTERS IN CONTROVERSY

The Board described Contention 7B in its March 15, 1982 Memorandum and Order reformulating and admitting the contention as "a general inquiry into the methodology used by LILCo and the Staff to determine whether there is reasonable assurance that the Shoreham design adequately protects from credible accidents."[4/] This general inquiry has focused on several areas and has addressed many issues within those areas. The principal issues addressed under Contention 7B are:

a. What are the regulatory requirements concerning the classification of structures, systems and components?

b. What is the methodology utilized by Applicant and the Staff to analyze the adequacy of the design of the Shoreham Nuclear Power Station?

c. Is the methodology adequate to ensure that structures, systems and components are properly classified and that appropriate quality standards and quality assurance requirements are applied?

d. Is the methodology adequate to ensure that systems interactions will not adversely affect plant safety? and

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4/ Memorandum And Order Confirming Rulings Made At The Conference Of Parties' (Regarding Remaining Objections To Admissibility Of Contentions And Establishment Of Hearing Schedule), dated March 15, 1982, at 13.

e. Is it necessary to apply the alternative methodologies cited by Intervenors' witnesses to the classification of Shoreham's structures, systems and components in order to make a finding that there is reasonable assurance of no undue risk to public health and safety?

D. RESOLUTION OF MATTERS IN CONTROVERSY

1. Summary

We decide that, contrary to the position taken by Intervenors, a "proper systematic methodology" has been used to analyze the reliability of structures, systems, and components at Shoreham, taking into account both the classification and qualification of plant items and the possibility of adverse systems interactions. This methodology consists of the application and satisfaction of deterministic criteria which are embodied in the Staff's Standard Review Plan and other regulatory guidance documents and in appropriate industry standards and practices. It is an established methodology which has evolved and proven its worth over many years of application; the Commission has relied consistently upon this proven methodology in licensing nuclear power plants in the past.

Applicant and the Staff have applied these deterministic criteria in the design and review, respectively, of the Shoreham Nuclear Power Station. We find that the application of these deterministic criteria has resulted in a nuclear power plant which generally meets the applicable regulatory requirements. Applicants' failure to have given proper meaning to the term "important to safety" has not been shown to have affected its compliance with the regulations; nevertheless, the definition for which the Staff and Intervenors argued is correct and will be a binding and enforceable part of any operating license which may issue as a result of this proceeding.

Intervenors would have us find that various types of additional analytical technique including PRA, failure modes and effects analyses and walkdowns, must be applied at Shoreham before this plant may be licensed for operation. We cannot agree with such findings. While the Commission may at some future time impose requirements for these or other analytical techniques in the assessment of the reliability of the structures, systems and components of a nuclear power plant, compliance with existing regulatory requirements can be and has been demonstrated without recourse to the supplemental methodologies cited by Intervenors. This Board may require no more than a demonstration of compliance with existing regulatory requirements. Reasonable assurance, rather than absolute assurance, of no undue risk to public health and safety is the standard set by the Commission's regulations.

We have afforded Intervenors considerable latitude and ample opportunity to prove their case. The record established on these contentions is massive and the post-hearing submissions are lengthy. Having carefully considered the evidence of record and the arguments of the parties, we decide that Contentions 7B and 19(b) lack merit.

2. Design Requirements for Nuclear Power Reactors Generally

a. Defense-in-depth philosophy

A concept called "defense-in-depth" provides the foundation and guiding principle for the design of a nuclear power plant. "Defense-in-depth" involves the use of multiple, successive barriers to the escape of radioactivity and the assurance that these barriers are not compromised as a result of transients or accidents. Several levels of protection are involved. (Finding 7B:7).

The first level of protection is provided by designing a plant for safety in normal plant operation and with tolerance for system malfunctions. Design criteria for many structures, systems and components required for normal plant operation, such as the main feedwater system and effluent control system, are found in the regulations and regulatory guidance documents. These criteria generally emphasize quality, redundancy and inspectability. (Finding 7B:8).

A second level of protection assumes that accidents will occur and requires the provision of systems to detect incipient failure and to shut down the plant when such incidents occur. (Finding 7B:9).

The third level of protection assumes the occurrence of damaging accidents; structures, systems and components are required to be provided to limit or control the consequences of postulated accidents. Analyses are conducted of specific "anticipated operational occurrences" and "accidents" to assure that plant trip or safety system equipment actuation occurs with sufficient capability and in sufficient time that the consequences of the occurrence or accident are within specified, acceptable limits. In addition, these "design basis analyses" are used to demonstrate that potential consequences are within acceptable limits when only safety-related equipment and systems are used to mitigate the consequences of the postulated events. The reactor fuel cladding, the reactor coolant system pressure boundary and the reactor containment building constitute the key parts of the third level of "defense-in-depth," though it includes many other systems as well. (Findings 7B:10, 12-16).

Another level of protection is provided by the trained plant operator and the emergency operating procedures developed for his use. In addition to the design basis events, analyses assuming various event sequences (including multiple failures) that could occur and fall outside the required design envelope have been utilized in the preparation of the emergency operating procedures. These emergency operating procedures are designed to permit operators to recognize and react to certain symptoms of events; in this way, the operator can gain control of the plant no matter what combination of failures caused the particular event. (Findings 7B:11, 18).

The various levels of protection which are involved in the "defense-in-depth" approach to nuclear power plant safety require the inclusion of many thousands of structures, systems and components in the design of a nuclear power reactor. Many are required simply for the reliable generation of power. Many others are designed into a plant to protect safety in the normal course of plant operation and in the prevention and mitigation of accidents and their consequences. (Findings 7B:8-16).

b. Regulatory requirements and terms

The Commission's regulations require that the principal design criteria for a nuclear power plant be identified and addressed in an application for an operating license. 10 CFR § 50.34(a)(3). Appendix A to 10 CFR Part 50 contains 64 criteria which are designated the General Design Criteria (or "GDC"). The Introduction to Appendix A explains that the principal design criteria for a proposed facility "establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems and components important to safety; that is, structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." The General Design Criteria, it goes on, "establish minimum requirements for the principal design criteria . . . ." (Finding 7B:40).

The General Design Criteria do not prescribe a particular methodology or methodologies to be used in the design and analysis of nuclear power plant systems, structures and components. Rather, criteria are established and the task is left to an applicant to demonstrate its compliance with these criteria. (Finding 7B:41). "General Design Criteria (GDC), as their name implies are 'intended to provide engineering goals rather than precise tests or methodologies by which reactor safety [can] be fully and satisfactorily gauged.'" Petition For Emergency and Remedial Action, CLI-78-6, 7 NRC 400, 406 (1978) (quoting Nader v. NRC, 513 F.2d 1045, 1052 (1975)). If an applicant demonstrates compliance with the GDC's, an adequate basis is provided for the licensing of the plant. A licensing board may not in the ordinary case require an applicant to satisfy requirements which go beyond those contained in the GDC's.[5/]

> "In the nuclear sphere, the Commission is the body which has been designated by Congress to make the hard decisions respecting what constitutes adequate protection to the public health and safety in the operation of a reactor -- and to give content to those decisions through the promulgation of appropriate standards and limitations with which the reactor must comply."

Maine Yankee, supra, at 1010.

The General Design Criteria establish various requirements "for structures, systems and components important to safety." The interpetation of this regulatory term is a significant area of disagreement among the parties in this proceeding. The term is used in several places in the regulations in addition to the General Design Criteria (see, e.g., 10 CFR § 50.34(a)(11), 50.34(b)(6)(vii), 50.49(b), 50.59(a)(2), 10 CFR Part 21). A second safety classification term -- "safety-related" -- also appears in the regulations (see, e.g., 10 CFR Part 50, Appendix B, Section I; 10 CFR § 50.55a(g)(1)). (Finding 7B:42).

The Commission, as we later detail, has recently reiterated the important distinction between the terms "important to safety" and "safety-related". This distinction was explained in a November 20, 1981 memorandum from Harold Denton, Director of the Office of Nuclear Reactor Regulation, to all NRR personnel (Suffolk County Attachment 1). "Important to safety" structures, systems and components are defined as those which provide

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5/ Maine Yankee Atomic Power Co. (Maine Yankee Atomic Power Station), ALAB-161, 6 AEC 1003, 1006-11 (1973), affirmed, CLI-74-2, 7 AEC 2, affirmed sub nom. Citizens for Safe Power v. NRC, 524 F.2d 1291, 1299-1300 (D.C. Cir. 1975); Public Service Co. of New Hampshire, et al. (Seabrook Station, Units 1 and 2), ALAB-422, 6 NRC 33, 42-43 (1977); see NRC Policy Statement, 45 Fed. Reg. 41738 (June 20, 1980).

reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. (Finding 7B:43). "Safety-related" is defined with reference to 10 CFR Part 100, Appendix A as describing those structures, systems and components which are necessary to assure: (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of Part 100. (Finding 7B:43). The Denton memorandum explains that safety-related is a subset of the class of important safety items. (Finding 7B:43).

Applicant took the position that these two terms are synonymous and that both refer to the plant items necessary to assure the three functions cited in 10 CFR Part 100, Appendix A. The application for Shoreham was prepared using the terms in this way. (Finding 7B:44).

In its proposed initial decision, Applicant characterizes its disagreement with the Staff and the Intervenors as "important and fundamental." Applicant reviews the "legislative history" of several sections of the regulations and compares the language used by the regulations in different places in an attempt to resolve a perceived ambiguity in the relationship of the terms "important to safety" and "safety-related." This ambiguity is summarized by LILCo on page 24 of Volume II its proposed initial decision, where LILCo states that the definition of "important to safety" which appears in the introduction of Appendix A to Part 50 "does not answer the question whether the class of important to safety is broader than that of safety-related; the safety-related set could easily be those needed to give reasonable assurance that the facility can be operated without undue risk to the public health and safety."

The entire regulatory exegesis presented by LILCo is grounded on the lack of a clear answer by the Commission to this question of whether the class of important to safety is broader than that of safety-related. Fortunately, a clear answer was very recently provided by the Commission: important to safety _is_ broader than safety-related in the Commission's view. On January 6, 1983, the Commission unanimously approved a revision to 10 CFR § 50.49 ("Environmental qualification of electric equipment important to safety for nuclear power plants"). In the statement of consideration accompanying the new rule, the Commission stated as follows:

> The scope of the final rule covers _that portion of equipment important to safety commonly referred to as 'safety-related' (which the Commission interprets as essentially 'Class 1E'_ equipment defined in IEEE 323-1974), and nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent the satisfactory accomplishment of required safety functions by safety-related equipment." (emphasis added)

(48 Fed. Reg. 2728, 2730 (1983)).

The language of the rule itself also makes clear that the class of important to safety is broader than that of safety-related. The new rule states in Section 50.49(b) that:

> "[e]lectric equipment important to safety covered by this section is (1) the safety-related equipment _and_ (2) the nonsafety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of specified safety functions _and_ (3) certain post-accident monitoring equipment" (emphasis added).

Significantly, in determining that the Regulatory Flexibility Act of 1980, U.S.C. 605(b) was not applicable, the Commission stated that "this rule codifies existing requirements." There is, accordingly, no reason to undertake the exegetical exercise suggested by the Applicant in an attempt to discern whether the class of important to safety is broader than that of safety-related. The Commission has clearly stated that it is and this Board is bound by that statement.[6/]

While the Commission has now made it clear that important to safety refers to a class of plant items which includes but is broader than the class of safety-related items, the Commission has not set out the specific bounds of the class of important to safety items. We adopt the definition of important to safety argued for by the Staff and Intervenors -- the so-called "Denton definition."[7/] Staff witness James H. Conran, Sr. presented the Staff's position on the Denton memorandum and its definitions. Mr. Conran was closely involved in the drafting of the Denton memorandum as a result of his appearance as a witness in the TMI-1 Restart proceeding. An issue in that hearing caused Mr. Conran to undertake an effort to find in the regulations the clear meaning of the terms "important to safety" and "safety grade." This involved an extensive review of those portions of the regulations in which safety classification terms are defined and safety classification

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6/ See, e.g., Northern States Power Company (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 51 (1978).

7/ At least one other licensing board has found that the safety classification definitions contained in the Denton memorandum most nearly reflect the system contemplated by the regulations. See Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1), LBP-81-59, 14 NRC 1211, 1342-46 (1981).

concepts established (i.e., 10 CFR Parts 20, 50 and 100). (Finding 7B:45). After testifying as a Staff witness at TMI-1 Restart, Mr. Conran was asked to prepare a statement of the definitions of these terms. He reviewed the many regulatory guidance documents (e.g., regulatory guides, Standard Review Plan, NUREG publications) in which those safety classification terms and concepts are further interpreted, developed and applied. Mr. Conran discussed these regulatory terms with Staff members whose background reflected a wide variety of experience including standards development, project management, technical review and management, and legal review. Mr. Conran also discussed the safety terms with the cognizant ACRS subcommittee. This effort covered more than a year, and it included review and concurrence in the definitions by all senior technical management officials in the Office of Nuclear Reactor Regulation prior to Mr. Denton's issuing these definitions in his November 20, 1981 memorandum. (Findings 7B:45, 46).

Mr. Conran also interacted with knowledgeable representatives of utility, vendor and architect-engineer organizations during the period in which the Denton memorandum was being prepared. Mr. Conran testified that he could not recall any industry representative giving any indication of fundamental disagreement with the "standard definitions" ultimately set forth in the Denton memorandum. (Finding 7B:47).

Mr. Conran emphasized that, as the Denton memorandum itself states, the Denton memorandum was not intended to impose new technical requirements on any licensee or applicant. Nor was it intended to clarify what any regulatory requirements are. It was intended, rather, to eliminate a terminological problem which had arisen because individual Staff members had in the past used the terms inconsistently.[8/] (Finding 7B:48).

We find the policy rationale supporting the Denton definition persuasive. Limiting the meaning of important to safety to safety-related would remove from the Commission's consideration a large number of systems, structures and components which the Staff considers necessary to assuring public health and safety. Certain items in the plant would no longer be subject to appropriate quality assurance requirements under GDC-1. Modifications could be made under 10 CFR § 50.59 (in systems that are not safety-related) that might degrade safety and yet be beyond effective Staff oversight. A licensee might overnarrowly construe its reporting obligations under 10 CFR Part 21. In sum, we agree with the Staff that LILCo's definition of important to safety would create a void in the regulations that provide assurance of public health and safety.[9/] (Finding 7B:50).

8/ LILCo suggests in its proposed initial decision (at 43), without record citation or any evidence whatsoever, the Mr. Conran "responded to [the Kemeny Commission's] criticism" of the NRC's safety classification scheme in his TMI testimony and the Denton memorandum. We reject this conclusion, and its implications, as totally unsupported by this record.

9/ By the logic for which LILCo argues, the Commission would be stripped of regulatory authority over a large number of plant structures, systems and components which even LILCo's witnesses agreed play a role in the safe operation of the plant. For example, effluent treatment systems are placed in a plant to ensure compliance with 10 CFR Part 20 requirements. (Finding 7B:27). These systems are also addressed in the GDC's. See GDC-60. Acceptance of LILCo's interpretation of "important to safety" in GDC-1 would mean that the Commission has no control over the quality standards and quality assurance program for systems which are clearly important in meeting the Commission's safety requirements (e.g., Part 20). This single example can be multiplied many times over.

c. Design and review of nuclear power reactors

No specific methodology is required by the regulations in deciding which plant items are "important to safety" and to what extent given criteria must be applied to them. Appendix B of 10 CFR Part 50 does require an applicant to "identify the structures, systems and components to be covered by the quality assurance program" mandated by Appendix B; no specification is given, however, as to the methodology to be used in that identification process.

The NRC Staff and applicants for operating licenses for nuclear power plants have developed deterministic criteria[10/] to ensure that the general requirements contained in the regulations are applied and satisfied in such a way as to provide reasonable assurance of no undue risk to the public health and safety. These deterministic criteria, based on many years of accumulated experience and technical judgments and analyses, are contained in the Staff's Standard Review Plan (NUREG-0800) and other regulatory guidance documents. (Findings 7B:21, 29, 32).

The Standard Review Plan embodies thinking, judgments and experience accumulated over many years of review and analysis of nuclear power reactors. (Finding 7B:21). It documents a systematic methodology for identifying structures, systems and components important to safety in the Staff's view. (Finding 7B:24). This methodology is understood and applied by applicants, including LILCo, in the preparation of an FSAR. (Finding 7B:24). By complying with the requirements of the Standard Review Plan, an applicant identifies and properly treats important to

10/ By "deterministic criteria," we mean established qualitative standards or requirements rather than numerical or probabilistic goals. (Finding 7B:206).

safety items because implicit in the criteria of the Standard Review Plan is an understanding of how important a system is and what quality standards it must meet. (Findings 7B:22).*

The Staff conducts an extensive audit-type review of the operating license application. This review effort focuses on safety-related structures, systems and components. However, an application prepared in accordance with the Standard Review Plan contains substantial information about items which are important to safety but not safety-related, and a substantial fraction of the Staff's review effort is concentrated on these plant items. (Findings 7B:33-35). Based upon its review of an applicant's adherence to these criteria, the Staff can conclude (and does here) that the requirements of the regulations have been satisfied. (Finding 7B:22,* 23, 37, 39).11/

This is the general methodology which has been utilized in the design and review of the Shoreham plant. Intervenors' Contention 7B and the testimony filed in support thereof question the adequacy of the methodology which has evolved as it relates to the classification of structures, systems and components and the analysis of systems interactions. Intervenors do not allege that Applicant and the Staff have failed to use any methodology in the analysis and classification of

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11/ Intervenors suggest that Staff's failure to realize until the submission of testimony in this hearing that LILCo had equated "important to safety" and "safety-related" calls into question the Staff's review methodology. First, it is LILCo's compliance with the regulations which is at issue here. Second, the failure to recognize this fact earlier despite the submission of a lengthy FSAR was made possible, in part, by the very systematic and detailed guidance the Standard Review Plan provides in terms of quality standards and design requirements for important to safety items.

plant structures, systems and components. Rather, Intervenors suggest that there are deficiencies in the methodology used and in the way the methodology was applied at Shoreham. They suggest several alternative methodologies which would, in their view, rectify these perceived deficiencies by supplementing the existing methodology.

We turn now to a closer examination of the way in which this general methodology has been brought to bear on the classification of Shoreham's structures, systems and components and the analysis of systems interactions. We examine the adequacy of Applicant's classification and treatment of specific Shoreham structures, systems and components selected by Intervenors and the adequacy of Applicant's evaluation of systems interactions at Shoreham particularly in relation to a specific system selected by Intervenors. Finally, we address the alleged need for the alternative methodologies discussed by Intervenors' witnesses.

3. Classification of Structures, Systems, and Components at Shoreham

a. Applicant's classification of safety-related structures, systems and components

The regulations require that an applicant identify the structures, systems and components to be covered by its Part 50, Appendix B quality assurance program, which applies to safety-related items.[12/] 10 CFR

12/ Although there is evidence that it was the original intent of the drafters of Appendix B of 10 CFR Part 50 to apply that appendix to all of the plant items to which Appendix A of that part applies, the application of Appendix B has consistently been only to safety-related structures, systems and components. The Staff is working on a proposed rule to expand the list of structures, systems and components subject to Appendix B and to provide regulatory guidance for appropriate quality assurance criteria for important to safety items. Research projects are ongoing in support of that Staff effort. (Finding 7B:79).

Part 50, Appendix B. In Table 3.2.1-1 of the FSAR, LILCo identifies these safety-related items. LILCo has drawn on information from several sources in identifying these safety-related items. The design basis analyses of Chapter 15 of the FSAR were examined to identify the structures, systems and components which are necessary to perform the critical safety functions of 10 CFR Part 100, Appendix A, at Shoreham. (Finding 7B:52). The Applicant has also taken into account accumulated industry experience and published guidance (ANS-22) for the classification of safety-related structures, systems and components at Shoreham. (Findings 7B:51, 53). In addition, the regulations themselves and regulatory guidance documents issued by the Staff (e.g., Regulatory Guides 1.26 and 1.29) have been utilized by the Applicant in classifying Shoreham plant items. (Findings 7B:51, 54-58). The Staff has reviewed Applicant's Table 3.2.1-1 and is satisfied that Applicant has used an adequate methodology and that a sufficient set of safety-related items has been identified. (Finding 7B:62).

LILCo's Table 3.2.1-1 was attacked by Intervenors as inadequate on two principal grounds: (1) alleged inconsistencies in the classification of particular components; and (2) alleged inadequacies in the scrutability of Table 3.2.1-1 and the level of detail presented therein. We find that LILCo's testimony has explained satisfactorily the reasons for the seemingly inconsistent classifications cited by Intervenors. (Findings 7B:64-70). We further find that Table 3.2.1-1 is understandable and adequate for the summary purposes for which it is presented. (Findings 7B:71, 72).

b. Applicant's classification and qualification of important to safety but not safety-related structures, systems and components

No list equivalent to Table 3.2.1-1 is provided for structures, systems and components which are important to safety but not safety-related; neither the regulations nor Staff guidance require the compilation of such a list, although structures, systems and components within this class are discussed throughout the FSAR. (Finding 7B:74). Intervenors claim, however, that absent the systematic identification of structures, systems and components important to safety but not safety-related, assurance cannot be had that applicant has complied with regulatory requirements for these items, particularly the quality standards and quality assurance requirements of GDC-1. For the reasons discussed below, we find that Applicant's FSAR, which was prepared in accordance with the Staff's Standard Review Plan and other regulatory guidance documents such as Regulatory Guide 1.70, does provide a systematic and sufficient identification of the Shoreham structures, systems and components which are important to safety and of the standards applied to those items. (Findings 7B:20-39).* Further, we are satisfied that Applicant and its contractors have generally applied appropriate quality standards and quality assurance requirements to the structures, systems and components of Shoreham. (Findings 7B:75-81).

1) Application of quality standards and quality assurance requirements generally

GDC-1 was the litmus selected by Intervenors for assessing whether Applicant had treated structures, systems and components important to safety but not safety-related consistently with regulatory requirements.

GDC-1 requires that important to safety structures, systems and components be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed and that a quality assurance program be established and implemented to provide adequate assurance that these plant items will satisfactorily perform their safety functions. 10 CFR Part 50, Appendix A. Considerable testimony was adduced by Applicant to demonstrate that, despite the question of the proper scope of GDC-1, all of Shoreham's structures, systems and components received appropriate quality standards and quality assurance treatment.

All of the Shoreham plant systems, including nonsafety-related systems, have been examined and evaluated for their significance to total plant function. (Finding 7B:75). Both General Electric and Stone & Webster evaluate nonsafety-related items to determine what standards are to be applied based on an assessment of the particular component's function and the expected service conditions. (Findings 7B:75, 79, 81). Although compliance with Appendix B of 10 CFR Part 50 is not required for nonsafety-related items, the principles of a comprehensive quality assurance program which the Appendix B criteria represent are applied to nonsafety-related items commensurate with the specific function performed. (Finding 7B:79).

2) Assessment of Specific Systems

This general description of the treatment of nonsafety-related structures, systems and components by Applicant and its contractors was tested by an examination of the treatment of certain specific Shoreham systems selected by Intervenors in their testimony. These included the standby liquid control system, the turbine bypass system, the reactor core isolation cooling system, the rod block monitor and the level 8 trip. In their proposed initial decision, Intervenors have abandoned their position that equipment may be misclassified with respect to three of these five systems (standby liquid control, reactor and isolation cooling and level 8 trip) by failing even to propose findings on them. Nevertheless, we address each of the five systems below.

(a) standby liquid control system (SLC)

The SLC system is designed to inject a neutron absorber solution (sodium pentaborate) into the reactor to shut the reactor down from rated power operation in the event that not enough control rods will be inserted to shut down the reactor. It provides a diverse, back-up means of reactivity control. (Finding 7B:85).

An analysis of the quality standards applied to the SLC system and the function it performs demonstrates that the system has been properly classified. All of the equipment essential for the injection of the sodium pentaborate solution into the reactor is safety-related. SLC system equipment not essential to solution injection has been designed to high standards and several specific design features assure the reliability of the system. (Findings 7B:87-90). The Board finds that the SLC system is properly classified and that it meets the requirements of GDC-1. (Finding 7B:91).

(b) turbine bypass system

The turbine bypass system is used during normal start-up and shutdown to pass partial steam flow to the condenser. The turbine bypass valves also operate automatically following a turbine trip or load rejection to stop the steam flow to the turbine. The accumulation of steam pressure may cause the turbine bypass valves to open in order to reduce the pressurization rate by directing some steam to the condenser. Careful design, procurement, installation and testing requirements have been applied to the turbine bypass system. The system is addressed by the Staff in the SER and complies with the Staff's Branch Technical Position incorporated in the Standard Review Plan. The Staff has also required a technical specification ordering periodic surveillance to confirm the operability of the turbine bypass sytem. (Findings 7B:93-102).

The Board finds that the turbine bypass system is properly classified. The Board is satisfied that the turbine bypass system need not be classified in its entirety as safety-related and that it has been properly designed with quality standards and quality assurance requirements commensurate with the importance of the safety function it performs. (Finding 7B:103).

(c) reactor core isolation cooling (RCIC)

The RCIC system is a high pressure system which provides core cooling during reactor shutdown by pumping makeup water into the reactor vessel in case of loss of flow from the main feedwater system; it can also be used to supplement the high pressure core injection system at high pressure conditions. Although the RCIC system is not a part of the emergency core cooling system network, the RCIC system initiates on low vessel water level during a loss of coolant accident and delivers a rated flow to the vessel through a connection in the feedwater system. (Finding 7B:105).

Almost all of the RCIC system is classified as safety-related; all of the equipment necessary for the RCIC system to perform its intended safety function of automatically injecting water is safety-related. (Finding 7B:106). The Board finds that the RCIC system, which is very nearly completely safety-related, is properly classified. The Board is satisfied that not all of the RCIC system needs to be classified as safety-related and that the system has been properly designed to quality standards and quality assurance requirements commensurate with the importance of its safety function. (Finding 7B:110).

(d) <u>rod block monitor (RBM)</u>

The rod block monitor, together with two other systems, performs the rod block function, which is designed to prevent erroneous withdrawal of a control rod or rods during normal operation, possibly resulting in local fuel damage. The principal objective of the rod block monitor is to extend fuel life by restricting rod movement to minimize local flux peaking. The RBM does not mitigate the control rod drop or any other accident and is not required to perform the critical safety functions of 10 CFR Part 100, Appendix A. (Findings 7B:112, 113).

The RBM is not a safety-related system. Nevertheless, special design features and other considerations have been applied to the RBM to assure its reliability; the RBM system meets most design principles of safety-related systems. Technical specification surveillance requirements are to be imposed further to assure rod block function operability.

(Findings 7B:114-117). The Board finds that the RBM system is properly classified and has been properly designed to quality standards and quality assurance requirements commensurate with its limited safety function. (Findings 7B:118).

(e) level 8 trip

The level 8 trip signal automatically trips the turbine and shuts down the feedwater pumps in the event that an excess of feedwater reaches the high water level trip setpoint. It is one line of defense against a feedwater controller failure transient, in which feedwater controller function is lost and a maximum feedwater flow is erroneously initiated; back-ups exist in the event of failure of the level 8 trip. (Finding 7B:120).

The level 8 trip is not safety-related, although it is a high quality designed and manufactured system. Technical specifications will limit the time during which portions of the level 8 trip system may be inoperable. (Findings 7B:121, 122). The Board finds that the level 8 trip need not be classified as safety-related. The Board further finds that the design and teatement of the level 8 trip is in compliance with the requirements of GDC-1. (Finding 7B:123).

c. Resolution of "important to safety" definitional controversy

We have discussed in an earlier section the controversy surrounding the terms "important to safety and "safety-related" and the recent Commission action consistent with the Staff's intepretation of these terms. Having reviewed Applicant's classification methodology and the application of that methodology to several specific systems, we are prepared to draw conclusions as to the significance of this definitional controversy in this proceeding.

The relevant question is whether Applicant's failure to have used the separate category of "important to safety" as that term is used by the Staff calls into question Applicant's compliance with certain regulatory requirements, <u>i.e.</u>, those which relate to items important to safety but not safety-related. The findings we have summarized in this section concerning Applicant's treatment of nonsafety-related items and of the several specific systems cited by Intervenors are consistent with the conclusions drawn by witnesses for Applicant and the Staff at the hearing: there is no evidence that the Applicant's incorrect definition of "important to safety" has had a substantive impact on the design and construction of the Shoreham plant. (Finding 7B:131).* Applicant has utilized the Standard Review Plan in preparing its FSAR and accordingly addresses the Staff's requirements for important to safety structures, systems and components. (Findings 7B:133-134).* Intervenors' testimony has not established a single case in which Applicant's failure to have used the term "important to safety" correctly has actually resulted in a substantive defect in the treatment of a structure, system or component at Shoreham. (See generally Findings 7B:84-123). The Staff's witnesses testified that they were aware of no specific example of a substantive difference in the plant caused by the definitional issue. (Finding 7B:131).* Applicant's witnesses testified on several occasions that no such substantive differences exist. (Finding 7B:131).*

Three "unacceptable implications" relating to the definitional controversy were described by the Staff witnesses at the hearings. The first concern was that the audit review procedure relied upon by the Staff might not have identified all areas in which Applicant's incorrect use of the term "important to safety" could result in less than complete compliance with regulatory requirements. (Finding 7B:136). The record which has been compiled in this proceeding, together with our finding that the Denton definition of "important to safety" is correct, is sufficient to allay that concern. Applicant's testimony that no substantive differences exist (Finding 7B:131)* stands uncontradicted. By virtue of this decision and 10 CFR § 50.54(h), Applicant will be bound to the Denton definition of "important to safety" and the Staff may take whatever steps become necessary if an area of substantive noncompliance should come to light in the future. No more is required at this time.

A concern was also raised by the Staff at the hearing that Applicant's incorrect use of the term "important to safety" could have an effect during the operational phase of Shoreham. Specifically with respect to this contention, the Staff testified that the commitment contained in FSAR § 3.1.2.1 relating to compliance with GDC-1 does not extend to the future operation of the plant because of Applicant's narrower construction of the term "important to safety." (Findings 7B:136, 138-139). While the Staff was satisfied with the quality assurance program described by Applicant for important to safety items (Finding 7B:140), the Staff lacked the commitment for the future it thought it had. (Findings 7B:138-141).* The Staff also expressed concern about Applicant's future compliance with Part 21. (Finding 7B:136). Again, our ruling that Applicant's construction of the regulatory term "important to safety" is incorrect and our endorsement of the Staff's interpretation will eliminate these potential problems. Since 10 CFR § 50.54(h) makes a license subject to "all rules, regulations, and orders of the Commission," our ruling as to the meaning of "important to safety" is binding on the Applicant and fully enforceable as part of any license which issues for Shoreham. For this reason, we find it unnecessary to impose a specific license condition on these points.13/

4. Analysis of Systems Interactions at Shoreham

a. Applicant's evaluation of systems interaction at Shoreham

One of the important concerns raised by Intervenors' testimony is that no adequate evaluation has been done of potential adverse systems interactions at Shoreham. Intervenors cited the water level indication system as an example of a system subject to adverse interactions. Extensive testimony was presented by the parties on the analysis of systems interactions at Shoreham and on the potential for interactions affecting the water level indication system specifically.

13/ A second factor now present is certain extra-record correspondence between Applicant and the Staff. During hearings on Contention 7B, the Board expressed surprise that discussions had not proceeded between Applicant and the Staff to resolve the definitional controversy outside of the evidentiary hearings. On December 16, 1982, LILCo's Vice President - Nuclear, M.S. Pollock, wrote to the Staff a letter which committed LILCo to continue to apply for the operational phase of Shoreham the quality standards and quality assurance requirements about which testimony was given. In a letter dated January 10, 1983, Thomas M. Novak, Assistant Director for Licensing, stated for the Staff that, in light of LILCo's commitment to implement an operational quality assurance program as required by GDC-1 of Appendix A for all features important to safety under the Staff's definition of that term, the scope of the quality assurance programs for Shoreham is acceptable. Accordingly, the Staff seeks no license condition on this subject and, as stated above, we see no need to impose one.

For the purposes of this opinion, we accept the Staff's definition of systems interaction: "the possibility of one reactor plant system acting on one or more systems in a way not consciously intended by design so as to adversely affect the safety of the plant." (Finding 7B:142).

Systems interactions are addressed throughout the design process by General Electric and Stone & Webster. (Findings 7B:143-150). Design practices and procedures at both General Electric and Stone & Webster incorporate measures to ensure appropriate dissemination and control of information, review and verification, and utilization of design and operating experience. Through these practices and procedures, potential interactions are identified and evaluated. (Findings 7B:143-147).

Beyond the basic practices and procedures used by General Electric and Stone & Webster in the design, manufacture and installation of structures, systems and components at Shoreham, a number of specific system interaction studies and programs have been conducted which relate specifically to Shoreham. Eighteen examples of such studies were discussed in Applicant's testimony. These included studies of missiles, cable separation, electrical bus failures, protection systems and scram reliability and many others. (Findings 7B:148-149). In addition, LILCo has established an organization (ISEG) to evaluate operational data, including information concerning systems interactions. (Finding 7B:151).

The Board finds that extensive evaluation has been conducted of potential adverse systems interactions at Shoreham. This evaluation has included both deterministic and probabilistic methodologies. Major parts of this evaluation are documented in the FSAR; other parts, such as the Shoreham draft PRA, have been conducted independent of any regulatory requirement. (Finding 7B:152). We turn now to a consideration of whether the adequacy of this process of evaluation of systems interactions is called into question by interactions relating to the water level indication system.

b. <u>Water level indication system interactions</u>

Intervenors selected the water level indication system (WLI) as an example of a plant system which is subject to interactions in a way that allegedly demonstrates the inadequacy of Applicant's methodology for analyzing the adequacy of plant design. Intervenors' witnesses testified that water level measurement is an important factor which can be adversely affected by a combination of high drywell temperature and low reactor vessel pressure to the point that emergency core cooling could be delayed. In Intervenors' view, the existing analysis and review techniques as documented in the FSAR and SER failed to discover this problem. (Finding 7B:153).

Reactor vessel water level is measured by differential pressure transmitters which measure the difference in status head between two columns of water. One column is a "cold" (ambient temperature) reference leg outside the reactor vessel; the other is the reactor water inside the reactor vessel and the variable leg. The measured differential pressure is a function of reactor water level. (Finding 7B:154).

All parties agree that high drywell temperature can cause boil-off or flashing of the water in the reactor water level sensing lines if the reactor is depressurized while these high temperatures exist. (Finding

7B:157). Such high drywell temperatures can be caused in several ways. Two scenarios were the focus of the testimony: (1) an incident at Pilgrim Nuclear Station in which loss of containment coolers caused flashing in the WLI reference leg; and (2) steam line breaks which discharge hot steam into the drywell causing boil-off in the WLI reference leg. (Findings 7B:157-158).

The interaction between the drywell coolers and WLI system was considered in the original design of Shoreham. (Finding 7B:172). Analyses have been conducted by General Electric and the Staff has reviewed these WLI system interactions specifically for Shoreham. (Findings 7B:158, 166-168). The design of Shoreham is adequate to ensure safety against both types of WLI system interaction cited. (Finding 7B:172). Cooling equipment is provided, temperatures are monitored and shutdown procedures are contemplated for these situations. (Finding 7B:172). Most importantly, the maximum water level measurement error is of little or no direct safety significance at Shoreham. (Finding 7B:160). Adequate cooling water will remain even in a worst case scenario and these errors in water level measurement indication are unlikely to delay emergency core cooling system actuation. (Findings 7B:160, 161). Specific emergency operating procedures address this contingency. (Findings 7B:162-164).

We find that the potential for such interactions adversely affecting the WLI system has been identified and reviewed through the methodology used by the Applicant and the Staff; there is ample evidence in the record that the loss of water in the water level sensing lines and resultant erroneous water level indication does not create undue risk to public health and safety at Shoreham. (Findings 7B:159-172).

c. Unresolved safety issues concerning systems interactions

Intervenors' testimony on Contention 7B questions the adequacy of the Staff's explanation, required by North Anna,[14/] as to why operation of Shoreham may be permitted despite the pendency of Unresolved Safety Issues A-17 and A-47 on the subject of system interactions.[15/] Under North Anna, the Staff is obliged to describe those generic problems under continuing study which have relevance to a given facility and which involve potentially significant public safety implications. This description is normally provided in the Safety Evaluation Report. In addition to a description of the issue and of the Staff's plan for resolving it, there must be some explanation why operation may proceed even though an overall solution has not been found. The most common explanations are that a situation has been implemented for the particular facility, that a restriction in the level or nature of operation has been imposed, or that the safety issue arises only in later years of operation. These are not the only acceptable explanations, however. For example, the explanation for operation pending resolution of the generic issue may be that current regulatory standards are adequate but confirmatory work is desirable or improved criteria are being sought. See Pacific Gas and Electric Co. (Diablo Canyon Nuclear Plant, Units 1 and 2), LBP-81-21, 14 NRC 107, 118 (1981).

14/ Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978).

15/ Both Applicant and the Staff moved to strike Intervenors' testimony regarding Unresolved Safety Issues A-17 and A-47 on the grounds that this testimony was beyond the scope of Contetion 7B. These motions were denied. See Tr. 1093-1103.

1) A-17 ("Systems Interactions")

The general concern involved in the systems interaction issue is the possibility of one reactor plant system acting on one or more other systems in a way not consciously intended by design so as to adversely affect the safety of the plant. The specific objective of a systems interaction analysis is to provide further assurance that the independent functioning of safety systems is not jeopardized by preconditions within the plant design (particularly dependencies hidden in supporting and interfacing systems). Events have occurred, the frequency and possibile implications of which have prompted the Staff to consider whether additional system interaction analysis requirements should be developed and imposed. (Findings 7B:174-175).*

The purpose of the A-17 task is to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems. (Finding 7B:176).

A program for studying the systems interaction issue was initiated in 1978. A candidate methodology for systems interaction analysis was developed and tested through application at Watts Bar, but this initial effort was deemed unsuccessful. (Findings 7B:177-178). In the aftermath of the TMI-2 accident, the TMI-2 Action Plan (NUREG-0660) incorporated the A-17 effort. The expanded systems interaction program under Action Plan Item II.C.3 has included surveys conducted by the national laboratories, seismic-initiator systems interaction reviews at Diablo Canyon and San Onofre, and a systems interaction study at Indian Point Unit 3.

It has been the Staff's intention to apply the systems interaction analytical methodologies on a trial basis, either as part of a "Pilot Program" or as part of the ongoing Systematic Evaluation Program or National Reliability Evaluation Program efforts. (Findings 7B:179-183).*

It is the Staff's intention that this generic program will provide the basis for making an orderly decision as to the possible need for additional systems interaction requirements. (Finding 7B:188)*. The program is confirmatory in nature, however, and the Shoreham SER concludes that reasonable assurance of public health and safety is provided by compliance with current requirements and procedures. (Findings 7B:176, 188). This conclusion is consistent with the position taken by the Staff before the Advisory Committee on Reactor Safeguards (ACRS) last year. In a February 12, 1982 letter from William J. Dircks, Executive Director for Operations, to Paul Shewmon, Chairman of ACRS, Mr. Dircks wrote that: "NRR continues in the confidence that current regulatory requirements and procedures provide an adequate degree of public health and safety." (Finding 7B:189). The Board agrees with the Staff and finds that the Staff has satisfied its obligations under North Anna to explain why operation of Shoreham may be permitted despite the pendency of unresolved safety issue A-17. (Finding 7B:202).

2) A-47 ("Safety Implications of Control Systems")

Unresolved safety issue A-47 concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. The purpose of the A-47 task is to examine the criteria and philosophy used by the Staff in the review of control systems to determine if they are sufficient and whether new criteria are appropriate. (Findings 7B:192-193). Should the resolution of A-47 indicate that additional criteria for control system designs are necessary or that specific problems require resolution, appropriate action will be taken by the Staff for plants in the licensing process as well as for plants now in operation. (Finding 7B:198). At this time, however, the Staff knows of no specific control system failures or actions on Shoreham or any other plant which would lead to undue risk to the health and safety of the public. (Finding 7B:198).

As part of the Staff's review effort relating to control systems, questions are asked of applicants relating to the effect of power supply and sensor and siren impulse line failures on several control systems simultaneously and to a plant-specific evaluation of the effect of high-energy line breaks on control systems. These are open items in the Staff's review of Shoreham at this time. (Findings 7B:199-201).

The Staff and LILCo have taken the position that the record on Contention 7B may be closed despite the pendency of these open items. We agree. We do not believe that the results of the Staff's review of further responses on these items by LILCo can reasonably be expected to add any new or different perspective to the extensive exploration of methodology which has been conducted throughout the evidentiary record on these contentions. As relates specifically to Unresolved Safety Issue A-47 and the Staff's North Anna obligations, we are satisfied that the Staff has provided the explanation required by North Anna (Finding 7B:202) and that the issue raised by the open requests of the Staff's A-47 review may be left for post-hearing resolution by the Staff. See Consolidated

Edison Co. of New York (Indian Point Station, Unit No. 2), CLI-74-23, 7 AEC 947, 951-52 (1974). In the absence of special circumstances and adequate justification by LILCo, the Staff will require resolution of the open item prior to fuel load.

5. Alternative Methodologies Proposed By Intervenors

At the heart of Intervenors' contention is the assertion that LILCo is unable to demonstrate that it has complied with the regulations in the absence of certain alternative methodologies, including PRA, for the analysis of systems interactions and the classification of structures, systems and components at Shoreham. Substantial evidence was presented by the parties on these alternative methodologies and their proper role in the regulatory process. As discussed below, we decide that the alternative methodologies cited by Intervenors are not required by the regulations and that it would be unnecessary and imprudent for us to rely on the Shoreham draft PRA and related testimony for the identification of intersystem dependencies or the classification of plant structures, systems and components.

a. Regulatory status of the alternative methodologies cited

Intervenors argue that such analytical methodologies as PRA, dependency analyses of various types, and a review of operating procedures must be applied to the analysis and classification of plant items in order to demonstrate compliance with the regulations. PRA is an analytical technique which permits the quantification of the probabilities and consequences associated with accidents and malfunctions by applying probabilistic and statistical techniques to an evaluation of plant reliability and safety. By using PRA, a safety assessor attempts to set into better perspective the contributors to various accident sequences and risk in order that appropriate remedial action may be taken. (Finding 7B:204).

The NRC's use of PRA in the regulatory process is in a state of development. No specific regulation requires a plant-specific PRA for Shoreham and the Staff has not requested that one be done. Both the Staff and LILCo argue that LILCo has gone beyond current regulatory requirements in undertaking a plant-specific PRA. (Findings 7B:205-206).

Intervenors have pointed to no specific regulation requiring a plant-specific PRA for Shoreham. Rather, they imply the need for a PRA from several regulations, including 10 CFR § 50.57 and 10 CFR Part 50, Appendix A. The need to imply such a requirement, however, arises even under Intervenors' argument only if Applicant is unable to demonstrate compliance with the regulations in the absence of a PRA. We are satisfied that Applicant has demonstrated its compliance with the regulations as they relate to the analysis and classification of Shoreham's structures, systems, and components. See Findings 7B:124, 152, 171, 203, 209. Accordingly, this Board need not and does not reach the question here whether the regulations may be read to imply a PRA requirement under appropriate circumstances.

The same conclusion necessarily applies with respect to systems interaction analyses other than PRA and operating procedures analyses. Certain provisions in the regulations do require systems interaction analyses of various types to be performed for particular plant systems.

Applicant's testimony demonstrates its compliance with these regulatory requirements. No regulatory requirement exists at this time, however, for the application on a plant-wide basis of any of these analytical methods. (Findings 7B:207, 211).

An additional reason exists for not requiring the use of PRA in the classification of plant structures, systems and components and the ranking of items by their importance to safety. There is not at present a systematic methodology for using PRA for the purpose of classification or the ranking of plant items by safety importance. The absence of standards for the use of PRA for classification or the ranking of plant by safety importance would render such analyses valueless. (Finding 7B:213). Further, there is no basis in the record before us for concluding that it is likely that a PRA would require a change in the classification of any structure, system, or component. (Finding 7B:214).

b. Reliance on the Shoreham draft PRA

Applicant devoted sixty-two pages of prefiled testimony to the subject of the Shoreham draft PRA and its relation in particular to the systems interaction issue raised by Contention 7B. The testimony of Dr. Vojin Joksimovich, a member of the peer review group for the Shoreham PRA, emphasized his opinion as to the effectiveness of the event tree/fault tree methodology utilized in the Shoreham PRA as tool for the analysis of systems interactions. Indeed, Dr. Joksimovich expressed his opinion that "the Shoreham PRA approach provides a meaningful and efficient, if not the only, framework for examining "the systems interaction issue." He went on to describe the Shoreham PRA as the "best means for addressing the issue." (Finding 7B:215). Dr. Edward T. Burns, SAI's principal analyst for the Shoreham PRA, described the methodology utilized and its application in the Shoreham PRA. Dr. Burns agreed with Dr. Joksmovich on the efficacy of PRA for systems interaction analysis:

> "SAI judges that fault tree/event tree methodlogy is the best available technique for augmenting the existing deterministic evaluations and NRC regulations to ensure that systems interactions are exposed and potential areas of concern are identified."

(Finding 7B:215).

LILCo's PRA witnesses have, thus, taken the position that the Shoreham PRA applies precisely the type of alternative methodological approach described as necessary by Intervenors. LILCo's witnesses also expressed their conclusion that the Shoreham PRA confirms the adequacy of the treatment of systems interactions at Shoreham. (Finding 7B:216). While this Board struck several such conclusions in the prefiled testimony at Intervenor's motion on the grounds that the conclusions (as opposed to the methodology) of the Shoreham PRA were beyond the scope of this contention, similar conclusions were elicited upon the record by Intervenors' own cross-examination. (Finding 7B:216).

Prior to the filing of Staff's direct testimony, this Board expressed its interest in the Staff's plans with respect to its review of the Shoreham PRA and the schedule for any such review. That interest was heightened by the extent to which LILCo's direct testimony relied on the Shoreham PRA. When it became clear to the Staff that the Board intended to inquire more deeply into this matter than the Staff's panel of witnesses were prepared to respond, the Staff moved and was permitted to add as a witness Ashok C. Thandani, Branch Chief of the Reliability and

Risk Assessment Branch of NRR. Mr. Thadani was most helpful to the Board in explaining the bases for the Staff's position with respect to the Shoreham draft PRA and in answering questions which arose on the subject of PRA generally.

The Staff emphasized repeatedly that it had not required the performance and submission of a PRA for Shoreham as part of the regulatory review process for issuing an operating license to LILCo and that LILCo had gone beyond regulatory requirements in conducting such a study. (Finding 7B:217). The Staff also testified that it lacked "specific criteria for evaluating such an assessment for Shoreham." Until the Commission promulgates specific criteria against which to compare PRA's, the Staff's approach is to learn from these studies whether there are areas which the Staff should be pursuing further. Judgments that are made depend on considerations other than just the numerical estimates. (Findings 7B:218-219). Despite these problems, the Staff will require submittal of the final Shoreham PRA and will review it to gain added insight into potential safety improvements. (Finding 7B:220).

With respect to the schedule for the Staff's review of the Shoreham PRA, the Staff testified to its expectation that the review effort would take approximately one year from the time the final Shoreham PRA is submitted. Mr. Thadani testified that the Staff cannot afford to to expand its limited resources on the review of draft PRA's because they generally change "radically" as time goes on. Mr. Thadani expected the Shoreham draft PRA to undergo substantial changes as a result of mistakes, omissions or new understandings before it became final. (Finding 7B:221).

In light of the schedule of this proceeding, the Board asked whether it would be possible to examine the Shoreham draft PRA on a short term basis specifically to evaluate its treatment of dependencies. Mr. Thadani considered the question overnight and responded that even such a quick review for treatment of dependencies would take 3 to 6 months in order to develop supportable views, assuming the availability of resources which the Staff does not believe are presently available. (Finding 7B:222).

While the Staff was unable to provide testimony specific to the Shoreham PRA for these reasons, Staff's witnesses did address the subject of PRA generally in response to questions from the Board. Among other subjects, that testimony addressed the question to what extent PRA can be used in a comprehensive way to identify intersystem dependencies.

The Staff does not at present have a position on the preferability of event tree/fault tree methodology as against other methodologies for the identification of intersystem dependencies. The Staff believes that it is premature at this time to draw any conclusion in this regard; the Staff is pursuing a program to identify the best, most effective technique. Under the Staff's program, another year or two of development and testing of techniques should permit identification of the most effective methods and the depth of analysis required to ensure that important dependencies have not been missed. (Finding 7B:224).

Many methods, including PRA, can be used to search for systems interactions. The difficulty is not in the use of event tree/fault tree methodlogy, but in how far these methods are carried: are the fault trees simplified or are they detailed down to the component level? An enormous amount of effort is required to do detailed fault trees on a large number of systems. (Finding 7B:226).

PRA has certain limitations at present. Limitations exist in the data base for probabilistic estimates. Quantification of factors such as sabotage may be impossible. Design errors may go unidentified. Potential dependencies may exist by design, by oversight or by operational considerations. Large areas of uncertainty must also be recognized. For example, probabilistic treatment of external events such as earthquake, flood, external fires and high wind displays large uncertainties. (Findings 7B:227-228).

Mr. Thadani described for the Board an "ideal approach" to the identification of important dependencies. The critical point, however, is that the Staff cannot say today how much analysis is enough to ensure adequate identification of dependencies. Dependencies are the hardest parts of a probabilistic analysis to identify and quantify. No single PRA to date has used all of the approaches which Mr. Thadani described as the ideal situation. (Findings 7B:230-231).

The Board finds that it is not prepared to rely on the Shoreham draft PRA for firm conclusions as to the identification of intersystem dependencies. First, it is a draft document still undergoing peer review. Changes may be made which would invalidate particular conclusions this Board might draw at present. Second, the Board does not have the benefit of the Staff's review of the document. Third, the Shoreham draft PRA excludes external events, for which large uncertainties exist.

Finally, the cautions raised by the Staff in its explanation of its position on whether PRA is, as LILCo argues, the "best method" of identifying dependencies cause us to hesitate to embrace LILCo's position at the present time.16/ (Finding 7B:232).

Nothing we have said should be taken as implying any belief that PRA is not a useful analytical technique. LILCo has gone beyond regulatory requirements in contracting for a PRA for Shoreham and it is to be commended for that undertaking. We simply hold that we are not prepared to place reliance on the Shoreham draft PRA on the basis of the present record to draw conclusions about its efficacy in identifying intersystem dependencies. Since we do not need to rely on such conclusions in view of our findings concerning the deterministic licensing criteria used by LILCo and the Staff, our unwillingness to rely on the Shoreham draft PRA has no effect on the licensing of Shoreham.

16/ After the close of the record on Contentions 7B and 19(b) but before the filing of findings, LILCo sought to have received in evidence excerpts of the deposition of Dr. Robert Jay Budnitz, a consultant for Intervenors on issues unrelated to these contentions. The pertinent portions of Dr. Budnitz's deposition made the points that: 1) the Shoreham draft PRA is a "state of the art" effort; and 2) the Shoreham draft PRA addresses systems interactions. We decline to reopen the record to receive the opinions and do not consider them in reaching our decision. The reasons for this decision are several. LILCo's offer of this evidence was untimely and good cause for the late offer was not shown. More importantly, the evidence does not have a material bearing on the outcome of our decision on the merits of these contentions since we decline to base our decision in any way on the Shoreham draft PRA. Further, this evidence is so conclusory as to be entitled to little weight.

6. Conclusion

a. Contention 7B

We conclude as follows with respect to Contention 7B:

1) Applicant has utilized a systematic methodology in the design of Shoreham Nuclear Power Station. That methodology is embodied in the regulations, the Standard Review Plan and other regulatory guidance, and industry standards and practices.

2) This systematic methodology has been applied at Shoreham in a way that ensures that Shoreham's structures, systems and components are properly classified and qualified, that appropriate quality standards and quality assurance requirements are applied, and that systems interactions will not adversely affect plant safety.

3) Intervenors were to select specific systems to demonstrate the alleged inadequacy of Applicant's methodology as it related to the classification of structures, systems and components and the analysis of systems interactions. The examples selected failed to demonstrate any inadequacy in the methodology utilized.

4) The Staff's interpretation of the regulatory term "important to safety" is correct and will be a binding and enforceable part of any operating license issued for Shoreham. Applicant's failure to have used a separate category of "important to safety" has made no substantive difference in the design of Shoreham.

5) The Staff has satisfied its North Anna obligations with respect to Unresolved Safety Issues A-17 and A-47; pending open items in the Staff's review under A-47 may be left for post-hearing resolution by the Staff.

6) The alternative methodologies cited by Intervenors are not necessary to demonstrate Shoreham's compliance with the regulations and we decline to rely on the Shoreham draft PRA.

b. Contention 19(b)

We conclude as follows with respect to Contention 19(b):

1) Applicant's classification of Shoreham's structures, systems and components meets the guidance provided in Regulatory Guide 1.26 and 1.29.

2) Applicant's seismic design classification of control room and radioactive waste systems are consistent with Regulatory Guide 1.143 and other applicable guidance and satisfies regulatory requirements.

3) Applicant has suitably documented its commitment to meet the requirement of NUREG-0737 relating to the classification of additional safety-related equipment.

4) Applicant's Table 3.2.1-1 need not include all equipment upon which plant operators may rely under the Shoreham emergency operating procedures.

II. FINDINGS OF FACT

A. INTRODUCTION AND BACKGROUND

7B:1. Intervenor Suffolk County ("SC" or "the County") and Shoreham Opponents Coalition ("SOC") proferred for litigation in this proceeding several contentions raising related issues concerning the safety classification and analysis of structures, systems and components at the Shoreham Nuclear Power Station. Long Island Lighting Company ("LILCo") and the NRC Staff ("Staff") both argued against the admission of these contentions. In a Memorandum and Order dated March 15, 1982, this Board confirmed rulings it had made at a prehearing conference of March 9 and 10, 1982 and overruled the objections of LILCo and the Staff to the admission of these contentions. The Board reformulated contentions SOC 7B(1),(2) and (4), SC 29, SC 7 and SC 6 into the following contention which was admitted for litigation:

> "LILCo and the Staff have not applied an adequate methodology to Shoreham to analyze the reliability of systems, taking into account systems interactions and the classification and qualification of systems important to safety, to determine which sequences of accidents should be considered within the design basis of the plant, and if so, whether the design basis of the plant in fact adequately protects against every such sequence. In particular, proper systematic methodology such as the fault tree and event tree logic approach of the IREP program or a systematic failure modes and effect analysis has not been applied to Shoreham. Absent such a methodological approach to defining the importance to safety of each piece of equipment, it is not possible to identify the items to which General Design Criteria 1, 2, 3, 4, 10, 13, 21, 11, 12, 24, 29, 35, 37 apply, and thus it is not possible to demonstrate compliance with these criteria."

7B:2. Intervenors decided and were permitted to combine their case on SOC Contention 19(b) with that on Contention 7B. SOC Contention 19(b) reads in full as follows:

"SOC contends that the NRC Staff has not required LILCo to incorporate measures to assure that Shoreham conforms with the standards or goals of safety criteria contained in recent regulatory guides. As a result, the Staff has not required that Shoreham structures, systems and components be backfit as required by 10 C.F.R. § 50.55a, § 50.57, and § 50.109 with regard to:

(b) Regulatory Guides 1.26 and 1.29. -- LILCo's general list of quality group and seismic design classifications listed in FSAR Table 3.2.1-1 is not in compliance with 10 C.F.R. Part 50, Appendix A, Criteria 1 and 2, 10 C.F.R. § 50.55a, and 10 C.F.R. Part 100, Appendix A in that:

(1) the quality group classifications contained in FSAR Table 3.2.1-1 do not comply with the regulatory position of Revision 3 of Regulatory Guide 1.26 for safety-related components containing water, steam or radioactive materials;

(2) the seismic design classifications contained in FSAR Table 3.2.1-1 do not comply with the regulatory position of Revision 3 of Regulatory Guide 1.29 with regard to control room habitability and radioactive waste systems;

(3) LILCo has not revised the FSAR Table 3.2.1-1 to expand the list of safety-related equipment as reflected in NUREG-0737 and as a result of the NRC Staff review of the Q-list as set forth in Supplement 1 of the SER on page 17-1; and

(4) LILCo's list of safety related equipment contained in FSAR Table 3.2.1-1 does not include equipment upon which the plant operators will rely in response to accidents outlined in the Shoreham emergency operating procedures."

7B:3. Hearings on Contention 7B (and SOC Contention 19(b)) were held on May 4-7, June 15-18, June 22-25, July 6-9, July 13-16 and July 21-22. Intervenors, LILCo and the Staff each presented a panel of witnesses; a total of twenty witnesses were heard by the Board during those twenty-two hearing days.

7B:4. Intervenor's case on Contention 7B consisted of the testimony of a panel of four witnesses: Gregory C. Minor, Richard B. Hubbard, Marc W. Goldsmith and Susan J. Harwood. Mr. Minor and Mr. Hubbard are vice-presidents of MHB Technical Associates, an engineering and consultant firm. Both Mr. Minor and Mr. Hubbard are engineers with experience in the nuclear industry at General Electric. Mr. Goldsmith and Ms. Harwood are president and a research engineer, respectively, of Energy Research Group, Inc., an energy consulting firm. Both Mr. Goldsmith and Ms. Harwood are nuclear engineers. Minor et al., ff. Tr. 1113.

7B:5. LILCo presented a panel of nine witnesses on Contention 7B. Robert M. Kascsak is the Nuclear Systems Engineering Divison Manager at LILCo. Mr. Kascsaks' education and experience are in the areas of mechanical and nuclear engineering. George F. Dawe, George Garabedian and Paul W. Rigelhaupt are from Stone & Webster Engineering Corporation, the architect-engineer for Shoreham. Mr. Dawe, Supervisor of Project Licensing, has over 15 years experience in the nuclear power field and demonstrated extensive knowledge of and familiarity with the Shoreham plant. Mr. Garabedian, a Senior Power Engineer, also has been involved for several years with the Shoreham project. Mr. Rigelhaupt, an Assistant Engineering Manager at Stone & Webster, has lengthy experience in chemical and nuclear engineering. David J. Robare and Pio W. Ianni are employees of General Electric Company, the nuclear steam supply system vendor for Shoreham. Mr. Robare, the Manager of BWR 4/5 Projects Licensing, has been responsible for the licensing of Shoreham for GE since 1975. Mr. Ianni, the Manager of Nuclear Systems Performance Engineering, has been employed by GE since 1951 and is presently responsible for directing overall BWR performance evaluations. Paul J. McGuire, a consultant to LILCO from United Energy Services Corporation, has been a certified senior reactor operator and Plant Manager at Pilgrim Station. Edward T. Burns, from Science Applications Inc. ("SAI"), is the lead analyst for the Shoreham probabilistic risk assessment ("PRA") study. Dr. Burns has extensive experience in engineering analysis and logic model construction for BWR PRA work. Finally, Vojin Joksimovich of NUS Corporation is a member of the peer review group for the Shoreham PRA; Dr. Joksimovich is a nuclear engineer with many years of experience in nuclear power risk assessment techniques. Burns _et al_., ff. Tr. 4346. Mr. William J. Roths of General Electric also appeared on behalf of Applicant. _See_ Tr. 4563 (Professional Qualifications of William J. Roths).

7B:6. The Staff's panel on Contention 7B originally consisted of six witnesses, and a seventh was later added. Themis P. Speis was, at the time of testimony, Assistant Director for Reactor Safety in the Divison of Systems Integration;[1/] much of the review for Shoreham was completed under the supervision of Dr. Speis. Walter P. Haass was, at the time of the testimony, Branch Chief of the Quality Assurance Branch,[2/] and has had oversight responsibilities for portions of the Shoreham review. Marvin W. Hodges is a Section Leader in the Reactor Systems Branch; Mr. Hodges conducted portions of the Shoreham review. C.E. Rossi is a Section Leader in the Instrumentation and Control Systems Branch;

1/ Dr. Speis has since been named Director of the Division of Safety Technology, Office of Nuclear Reactor Regulation.

2/ In a recent reorganization, the Quality Assurance Branch was moved from the Office of Nuclear Reactor Regulation to the Office of Inspection and Enforcement; Mr. Haass is now Deputy Branch Chief of the Quality Assurance Branch in the Division of Quality Assurance, Safeguards and Inspection Programs.

Dr. Rossi was also responsible for supervising portions of the Shoreham review. James H. Conran, Sr. is a Principal Systems Engineer in the Systems Interaction Section, Reliability and Risk Assessment Branch; Mr. Conran is knowledgeable on the subjects of safety classification terminology and the Staff's system interaction program. Robert Kirkwood is a Principal Mechanical Engineer in the Mechanical Engineering Branch, and had responsibility for the review of the classification of the safety-related structures, systems and components at Shoreham except for electrical and electronic items. Speis _et al_., ff. Tr. 6357. The Staff filed rebuttal testimony through Mr. Conran on an issue relating to safety classification terminology. Conran, ff. Tr. 6368. Ashok C. Thadani was added to the panel after testimony had begun. Mr. Thadani, Branch Chief of the Reliability and Risk Assessment Branch, addressed questions which the Board had raised concerning PRA and systems interaction issues. _See_ Tr. 6453 (Professional Qualifications of Ashok C. Thadani).

B. DESIGN REQUIREMENTS FOR NUCLEAR POWER REACTORS GENERALLY

1. Defense in depth philosophy

7B:7. Current licensing requirements are founded on the principle of "defense-in-depth." Staff Ex-2A, at B-9. In nuclear power plant design, defense-in-depth has several elements. These can be stated as follows: (1) provide a well-engineered plant that operates reliably; (2) provide protection against operational transients (or "anticipated operational events") due to equipment failures or malfunctions; and (3) provide multiple back-ups such that critical safety functions will be performed in the event of accidents. Burns *et al*., ff. Tr. 4346, at 27.

7B:8. The first level of protection is provided by designing the plant for safe and reliable normal operation and with tolerance for system malfunctions. It emphasizes quality, redundancy and inspectability. Criteria and requirements applied to the structures, systems and components needed for normal operation (*e.g.*, primary pressure boundary, main feedwater system, main steam system, turbine, radiation monitoring system, effluent control system, the control room and control room systems) are found in the General Design Criteria and in regulatory guidance documents such as the Standard Review Plan and Regulatory Guides. Speis *et al*., ff. Tr. 6357, at 18-19.

7B:9. The second level of protection assumes that incidents will occur in spite of care in design, construction and operation. It requires the provisions of systems to detect incipient failure and to shut down the plant so as to prevent or minimize damage when such incidents occur. Speis *et al*., ff. Tr. 6357, at 19.

7B:10. A third level of protection is provided by "safety-related" systems, structure and components, which limit or control the consequences of accidents. Speis et al., ff. Tr. 6357, at 19. Safety-related structures, systems and components are those necessary to assure the required safety functions, i.e., (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. Speis et al., ff. Tr. 6357, at 6.

7B:11. Another level of protection is provided by the trained operator and the emergency operating procedures. The operator, utilizing these procedures, is trained to take actions to maintain the plant in a safe condition independent of the type or number of equipment or system failures which occur. In performing the key functions, the operator may use, by procedure, systems which are not safety-related; however, safety-related systems provide adequate protection should the unsafety-related systems fail. Speis et al., ff. Tr. 6357, at 20.

7B:12. A basic premise in the licensing of nuclear power plants is that the "safety-related" items can be singled out from the many thousands of structures, systems and components in a plant and given more stringent design criteria and quality assurance standards and more extensive NRC review than other plant items receive. Speis et al., ff. Tr. 6357, at 6. In some cases, safety-related structures, systems and components are used during normal plant operation (e.g., reactor coolant system). In other cases, safety-related items are provided for the sole purpose of accomplishing safety functions (e.g., reactor trip and decay heat removal). Speis et al., ff. Tr. 6357, at 6.

7B:13. Having a specific, well-defined group of safety-related structures, systems and components allows both an applicant and the Staff to concentrate their efforts on the items most important in achieving critical safety functions in case of an accident or emergency situation. Speis et al., ff. Tr. 6357, at 7.

7B:14. To ensure that the proper systems, structures and components are classified as safety-related, an applicant conducts analyses of specific "anticipated operational occurrences" and "accidents" in Chapter 15 of its Final Safety Analysis Report (FSAR). Staff review procedures for these "design basis" analyses are delineated in Chapter 15 of the Standard Review Plan. Speis et al., ff. Tr. 6357, at 15-16.

7B:15. The design basis analyses are utilized to demonstrate that plan trip and/or safety system equipment actuation occurs with sufficient capability and on a time frame such that the consequences are within specified, acceptable limits. Conservative initial plant conditions, core physics parameters, equipment availability and instrumentation setpoints are assumed. Conservative core parameters (such as heat fluces, temperatures, pressures and flows) are also assumed. Among the specific set of "anticipated operational occurrences" and "accidents" analyzed are the limiting events resulting from both mechanistic and non-mechanistic equipment and system failures. The conservative bounding analyses performed are used to demonstrate that the potential consequences to the health and safety of the public are within acceptable limits for a wide range of postulated events even though specific actual events might not follow the same assumptions made in the analyses. Speis et al., ff. Tr. 6357, at 16.

7B:16. The analyses performed are used to demonstrate that the potential consequences to the health and safety of the public are within acceptable limits (i.e., offsite exposures are less than the guideline exposures of 10 CFR Part 100) when only safety-related equipment and systems are used to mitigate the consequences of the postulated events. Sufficient safety-related equipment is provided to assure that essential safety functions will be performed even with the most limiting single failure. Speis et al., ff. Tr. 6357, at 16-17.

7B:17. The Chapter 15 design basis analyses do not include all possible accident sequences. It is not possible to analyze or even define all possible accident sequences for any nuclear power plant. However, the transients and accidents analyzed are representative of classes of events that have been judged to be of significant severity and sufficient likelihood to require consideration. The methods of analysis and the acceptance criteria are conservative, acting as bounding representations of actual or expected conditions. Speis et al., ff. Tr. 6357, at 17-18. The analyses include some multiple failure sequences, including some independent multiple failures. Tr. 1720-22 (Minor).

7B:18. In addition to the design basis events, analyses assuming various event sequences (including multiple failures) that could occur and fall outside the required design envelope have been utilized in the preparation of the emergency operating procedures. Speis et al., ff. Tr. 6357, at 20; see Tr. 1722-23 (Minor, Goldsmith). The objective of this approach, which was a result of the lessons learned from the TMI-2 accident, is to further assure that the operator is able to respond to the complete spectrum of possible events. Operators are trained to recognize symptoms of events and to respond to those symptoms rather than to any specific event. In this way, the operator can gain control of the plant no matter what combination of failures caused the particular event. Speis et al., ff. Tr. 6357, at 20-21.

7B:19. The design basis approach and defense in depth philosophy have been applied at Shoreham. Burns et al., ff. Tr. 4345, at 27-30.

2. Design and review of nuclear power reactors

7B:20. Design criteria and quality standards for structures, systems and components important to safety are required to be addressed in the FSAR. Speis et al., ff. Tr. 6357, at 9; Tr. 7079 (Speis).

7B:21. The FSAR is reviewed by the Staff against the specific criteria provided by the Standard Review Plan (NUREG-0800). The Standard Review Plan embodies thinking, judgments, and experience accumulated over many years of review and analysis of a number of nuclear power plants. Tr. 6583 (Conran); Tr. 6574 (Rossi).

7B:22. By complying with the requirements of the Standard Review Plan, an applicant identifies and properly treats important to safety items because implicit in the criteria of the plan is an understanding of how important a system is and what quality standards that system must meet. Tr. 6583 (Conran).* Compliance with Standard Review Plan requirements is used to demonstrate compliance with the regulations. Tr. 6584 (Conran).

7B:23. The Staff's use of the Standard Review Plan ensures that an applicant has properly addressed the plant items the Staff considers important to safety. Tr. 7093-98 (Rossi, Conran).

7B:24. The Standard Review Plan documents a systematic methodology for identifying structures, systems and components under Staff practice. Tr. 6577, 6581 (Rossi). This methodology is understood and applied by applicants in the preparation of FSAR's. Tr. 6580 (Rossi).

7B:25. The Standard Review Plan includes the basis for reviewing nonsafety-related as well as safety-related items. Tr. 7474 (Speis). For example, the turbine bypass is an example of a nonsafety-related system covered in the SRP. Tr. 7474 (Speis). The relevant Standard Review Plan section, 3.2.2-12, refers to a specific General Electric publication for appropriate quality control procedures. Tr. 7435 (Kirkwood).

7B:26: Important to safety items are addressed throughout the Standard Review Plan and discussed throughout the FSAR. Dr. Rossi gave examples of design bases for nonsafety-re ed items from the FSAR which included portions of the rod block monitor system, the traversing in-core probe subsystem, the reactor manual control system and the feed-water control system. Tr. 7093-95 (Rossi). Dr. Speis cited the analysis in Chapter 10 of the FSAR relating to the steam and power conversion system. Tr. 7101 (Speis). Mr. Conran added the example of the Standard Review Plan process for review of high energy line breaks, including many nonsafety-related systems, and described the methodology required for that analysis as "very extensive [and] very sophisticated." Tr. 7098 (Conran).

7B:27. The Shoreham FSAR describes Applicant's treatment of many important to safety structures, systems and components. For example, Chapter 11 of the FSAR discusses radioactive waste management systems. Burns et al., ff Tr. 4346, at 41. These are systems which are in the plant to meet 10 CFR Part 20 requirements. Tr. 5430 (Dawe).

7B:28. Everything discussed in the FSAR is important to safety, "that is why it is there." By putting an FSAR together and addressing the systems that the Staff requires to be addressed through the regulations and regulatory guidance, an applicant identifies items important to safety. Tr. 6974 (Conran). Design criteria and quality standards for all structures, systems and components important to safety are required to be addressed, some in considerably more detail than others, in the applicant's Safety Evaluation Report. Speis et al., ff. Tr. 6357, at 9. Compliance with the criteria and requirements of approved regulatory guidance documents assures that the important to safety items are properly classified and addressed. Id., at 10.

7B:29. A well-developed, systematic process for classification of plant structures, systems and components is embodied in the Standard Review Plan and regulatory guides. Tr. 6563-65 (Rossi, Conran).

7B:30. Compliance with the Standard Review Plan constitutes a systematic methodology for the classification of structures, systems and components. Tr. 6582-84 (Conran); 7098 (Rossi). Speis et al., ff. Tr. 6357, at 9-10.

7B:31. The Shoreham application has been reviewed extensively by the Staff. The Staff's review of the Shoreham application has been ongoing for about 6 years. Tr. 7464 (Speis). The Staff estimated that about 26 staff years of review effort have been devoted to Shoreham by approximately two dozen technical branches of the Office of Nuclear Reactor Regulation. A staff year is 1800 productive hours. Tr. 7466-67, 7472 (Speis, Rossi).

7B:32. Shoreham plant systems design was reviewed against the criteria and requirements of approved regulatory guidance such as applicable Regulatory Guides and Standard Review Plan sections. Speis et al., ff. Tr. 6357, at 23.

7B:33. Staff witness Rossi described this review, which is characterized by the Staff as an "audit review." A reviewer in a technical branch of the Office of Nuclear Reactor Regulation reads the appropriate section of the FSAR. Questions are then developed both to seek additional information and to obtain specific commitments from an applicant as to particular design features in the plant. The actual review is concentrated in areas where NRC Staff members think it would be most difficult for the applicant and the architect-engineer to meet the design criteria. Special attention is also given to issues recently highlighted within the agency and to areas that are new in a particular plant design. The audit is selective in nature rather than random. Tr. 6947-48 (Rossi). Dr. Speis described the audit review as a selective "picking and choosing process." Tr. 7977 (Speis)

7B:34. The Staff concentrates its review effort on structures, systems and components which are most important in achieving the critical safety functions of 10 CFR Part 100, Appendix A (i.e., the safety-related items). A substantial fraction of the Staff's review effort, however, is applied to items whose proper operation can help prevent accidents or emergency conditions and, in fact, whose operation is important in assuring public health and safety even if there is never an accident (i.e., the important to safety but not safety-related items). Speis et al., ff. Tr. 6357, at 7; Tr. 7815 (Speis).

7B:35. Dr. Speis estimated that approximately 25% of the Staff's review effort is directed to the important to safety but not safety-related class of structures, systems and components. Tr. 7808 (Speis). It is and has been consistent Staff practice to review particular structures, systems and components important to safety but not safety-related as part of its licensing review. Tr. 7815 (Speis).

7B:36. The Staff has drawn judgments as to the degree to which a given GDC's are applicable to particular items in the plant. Those judgments are reflected in various sections of the Standard Review Plan. Tr. 7086-87 (Kirkwood).

7B:37. The classification of safety-related structures, systems and components is reviewed specifically by the Staff. While the Staff does not review specifically the classification of those items which are important to safety but not safety-related, the Staff's review of an applicant's compliance with the criteria and requirements of approved regulatory guidance documents assures that such structures, systems and components are properly classified and addressed in an applicant's submittal. Speis et al., ff. Tr. 6357, at 8-10.*

7B:38. The Standard Review Plan suggests exact words that should be included in the Staff's Safety Evaluation Report upon a determination that appropriate standards and criteria have been satisfied. Tr. 7096 (Conran).

7B:39. On the basis of its review utilizing the Standard Review Plan, the Staff has concluded that those structures, systems and components that provide reasonable assurance that Shoreham can be operated without undue risk to the health and safety of the public have been adequately addressed by the Applicant and the Staff in terms of their safety classification and reliability through the design and review process. Speis et al., ff. Tr. 6357, at 46.* Thus, the Staff, on the basis of its systematic review process, has concluded that Shoreham meets the General Design Criteria. Tr. 7850 (Speis).

3. Regulatory requirements and terms

7B:40. Appendix A to 10 CFR Part 50 contains 64 criteria which are designated the General Design Criteria (or "GDC"). The Introduction to Appendix A explains that the principal design criteria for a proposed facility "establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems and components important to safety; that is, structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." The General Design Criteria, it goes on, "establish minimum requirements for the principal design criteria . . . . " 10 CFR Part 50, Appendix A.

7B:41. The general Design Criteria do not prescribe a particular methodology or methodologies to be used in the design and analysis of nuclear power plant systems, structures and components. Rather, criteria are established and the task is left to an applicant to demonstrate its compliance with these criteria. Tr. 1792-93 (Hubbard).

7B:42. The term "important to safety" is used in several places in the regulations in addition to the General Design Criteria (see e.g., 10 CFR § 50.34(a)(11), 50.34(b)(6)(vii), 50.49(b), 50.59(a)(2), 10 CFR Part 21). A second safety classification term -- "safety-related" also appears in the regulations (see e.g., 10 CFR Part 50, Appendix B, Section I; 10 CFR § 50.55a(g)(1)).

7B:43. "Important to safety" structures, systems and components are those which provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. "Safety-related" is defined with reference to 10 CFR Part 100, Appendix A as describing those structures, systems and components which are necessary to assure: (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of Part 100. These definitions are set out in a November 20, 1981 memorandum from Harold Denton, Director of the Office of Nuclear Reactor Regulation, to all NRR personnel (Suffolk County Attachment 1) for the definition of these terms. The Denton memorandum explains that safety-related is a subset of the class of important to safety items. The definitions embodied in the Denton memorandum constitute the Staff's position on what the regulations mean. Tr. 6957-58 (Conran). Intervenors concur in these definitions.

7B:44. Applicant took the position that these two terms are synonymous and both refer to the plant items necessary to assure the three functions cited in 10 CFR Part 100, Appendix A. Tr. 4790 (Robare);

Tr. 7057 (Haass). The application for Shoreham was prepared using the terms in this way. Tr. 4470, 4485 (Dawe). No separate category of "important to safety" was recognized by LILCo. Minor et al., ff. Tr. 1113, at 19; Tr. 6527 (Kirkwood); Tr. 6961-62 (Conran).

7B:45. Staff witness James H. Conran, Sr. presented the Staff's position on this issue at the hearing. Mr. Conran was closely involved in the drafting of the Denton memorandum as a result of his appearance as a witness in the TMI-1 Restart proceeding. An issue in that hearing caused Mr.Conran to undertake an effort to find in the regulations the clear meaning of the terms "important to safety" and "safety grade." This involved an extensive review of those portions of the regulations in which safety classification terms are defined and safety classification concepts established (i.e., 10 CFR Parts 20, 50 and 100). He reviewed the many regulatory guidance documents (e.g., regulatory guides, Standard Review Plan, NUREG publications) in which those safety classification terms and concepts are further interpreted, developed and applied. Conran, ff. Tr. 6368, at 3-4.

7B:46. After testifying as a Staff witness at TMI-1 Restart, Mr. Conran was asked to prepare a statement of the definitions of these terms. Mr. Conran discussed these regulatory terms with Staff members whose background reflected a wide variety of experience including standards development, project management, technical review and management, and legal review. Mr. Conran also discussed the safety terms with the cognizant ACRS subcommittee. This effort covered more than a year, and it included review and concurrent in the definitions by all senior technical management officials in the Office of Nuclear Reactor Regulation prior to Mr. Denton's issuing these definitions in his November 20, 1981 memorandum. Conran, . Tr. 6368, at 4-5.

7B:47. Mr. Conran also interacted with knowledgeable representatives of utility, vendor and architect-engineer organizations during the period in which the Denton memorandum was being prepared. Mr. Conran testified that he could not recall any industry representative giving any indication of fundamental disagreement with the "standard definitions" ultimately set forth in the Denton memorandum. Conran, ff. Tr. 6368, at 5; Tr. 7762 (Conran).

7B:48. The purpose of the Denton memorandum was to eliminate a terminological problem which had arisen because individual Staff members had in the past used the terms incorrectly and inconsistently. It was not intended to impose new technical requirements on any licensee or applicant or to clarify regulatory requirements. Conran, ff. Tr. 6368, at 5; Tr. 7734, 7839-40 (Conran).

7B:49. Contrary to Applicant's proposed finding B-169, the Instrumentation and Control Systems Branch does use the term and applies the concept "important to safety" as defined in the Denton memorandum. Tr. 6574; 6577 (Rossi). A major portion of that branch's work, however, relates to reviewing safety-related systems. Tr. 6505-07 (Rossi).

7B:50. The Board concurs in the safety classification definitions contained in the Denton memorandum and finds tht the three-stage classification scheme described by the Staff and the Intervenors most nearly reflects that contemplated by the regulations. At least one other licensing board has so found. See Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit No. 1), LBP-81-59, 14 NRC 1211, 1342-56

(1981). Limiting the meaning of important to safety-related (as all parties agree on the definition of that latter term) would remove from the Commission's consideration a large number of systems, structures and components which the Staff considers necessary to assure public health and safety. The NRC's concern for public health and safety goes beyond the accident-related releases of Appendix I to Part 100. It also includes the lower releases of limits of Appendix I to Part 50 and of Part 20; it includes normal operation as well as accidents. Tr. 6535-36 (Conran). Under LILCo's narrow interpretation of important to safety, certain items in the plant would no longer be subject to appropriate quality standard and quality assurance requirements under GDC-1. Tr. 7817 (Haass). Modifications could be made under 10 CFR § 50.59 in systems that are not safety-related that might degrade safety and yet be beyond effective Staff oversight. Tr. 7819 (Rossi). A licensee might overnarrowly construe its reporting obligations under 10 CFR Part 21. Speis et al., ff. Tr. 6357, at 7. In sum, there would be a void in the regulations that provide assurance of public health and safety. Tr. 7817 (Rossi, Haass, Conran).

C. CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS AT SHOREHAM

1. Applicant's Classification and Qualification of Safety-Related Structures, Systems and Components

a. Methodology and application

7B:51. The methodology used for classification of systems, structures and components at Shoreham involved the application of design basis evaluations, industry standards, regulations, regulatory guides and design and operating experience. Burns *et al*., ff. Tr. 4346, at 27.

7B:52. The design basis analyses contained in Chapter 15 of the FSAR enable an applicant to determine those features of the plant that will be necessary to provide mitigation of accidents as required by 10 CFR Part 100. Those structures, systems and components which are relied upon to perform the three critical safety functions of 10 CFR Part 100[3/] are classified as safety-related. Burns *et al*., ff. Tr. 4346, at 27-30.

7B:53. Industry experience in the design and classification of numerous boiling water reactors prior to Shoreham led to the compilation by the industry of guidance for classification in ANS-22 (now issued as

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3/ These critical safety functions are assuring:

(1) the integrity of the reactor coolant pressure boundary;

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

Speis *et al*., ff. Tr. 6357, at 6.

ANSI/ANS-52.1). Burns et al., ff. Tr. 4346, at 29; see Tr. 1322 (Goldsmith). It is the purpose of this industry standard to set out functional safety requirements for design, to be responsive to NRC regulatory requirements and industry technical requirements, and to provide a uniform basis for design safety requirements to be reflected in licensing documents. ANS-22 was used in establishing the classification of structures, systems and components for Shoreham. The equipment classification table provided in the Shoreham FSAR (Table 3.2.1-1) was structured to provide a description of these classifications with content and format similar to that provided in ANS-22. Burns et al., ff. Tr. 4346, at 30-31. The development of ANS-22 itself included a comprehensive examination of the safety aspects of boiling water reactors. Attachments 2 and 3 to LILCo's prefiled testimony provide detailed background of the development of ANS-22 and the types of analyses which underlay it. Burns et al., ff. Tr. 4346, at 31-34.

7B:54. The NRC Staff has published guidance for the classification of nuclear power reactor structures, systems and components in the form of regulatory guides. Regulatory Guide 1.26 provides quality group classifications for fluid system components (i.e., water, steam and radioactive waste containing components). Regulatory Guide 1.29 identifies those structures, systems and components that should be designed to withstand the effects of the Safe Shutdown Earthquake and remain functional. As stated in FSAR Sections 3.2.1, 3.2.2 and Appendix

3B, the structures, systems and components of Shoreham were classified in accordance with these two regulatory guides. Burns _et al._, ff. Tr. 4346, at 35; Speis _et al._, ff. 6357, at 10-13.

7B:55. Revision 1 of Regulatory Guide 1.26 was used by LILCo since this was the revision in effect at the time the FSAR was docketed. The current revision of Regulatory Guide 1.26 is Revision 3 which is not substantially different from Revision 1. As there are no changes in Revision 3 which would cause a change in the system quality group classifications of the water, steam and radioactive waste containment components at Shoreham, the use of Revision 1 is acceptable. Speis _et al._, ff. Tr. 6357, at 12.

7B:56. Revision 1 of Regulatory Guide 1.29 was used by LILCo since this was the revision in effect at the time the FSAR was docketed. The current revision of Regulatory Guide 1.29 is Revision 3, which is not substantially different from Revision 1. As there are no changes in Revision 3 that would cause a change in the seismic classification of the structures, systems and components at Shoreham, the use of Revision 1 is acceptable. Speis _et al._, ff. Tr. 6357, at 11.

7B:57. Shoreham's radioactive waste management systems are classified in accordance with Regulatory Guide 1.143. Speis _et al._, ff. Tr. 6357, at 12. The control room air conditioning system is seismic Category I, subject to Appendix B quality assurance, and is in conformance with current applicable regulatory requirements. Id. at 14-15.

7B:58. Compliance with Appendix A of 10 CFR Part 100, 10 CFR Part 50 Appendices A and B and 10 CFR 50.55a specifically constituted a part of the methodology for the classification of structures, systems and components at Shoreham. Burns et al., ff. Tr. 4346, at 38-39.

7B:59. General Electric boiling water reactors have compiled over 400 reactor-years of operating experience. All of this operating experience has been brought to bear on the classification of Shoreham structures, systems and components. Burns et al., ff. Tr. 4346, at 40.

7B:60. Part of the methodology for the classification of structures, systems and components at Shoreham was a General Electric review effort called the nuclear safety operational analysis. This effort was undertaken to provide an organized approach to identification of situations in which safety related systems would be called upon. The analyses assume various transient and accident initiations and identify the mitigating or back-up equipment needed to terminate the events. Burns et al., ff. Tr. 4346, at 32-34; Tr. 5414 (Robare); Tr. 5497 (Ianni).

7B:61. There was a complete reanalysis by GE of the Shoreham equipment classification in 1979. Tr. 4609 (Ianni). This reanalysis included both safety-related and nonsafety-related equipment within GE's scope of supply. Tr. 4628 (Robare, Ianni). The review was conducted by the lead system engineers and the component engineers in conjunction with licensing engineers. It consisted of reviewing the engineering documents, piping and instrumentation diagrams, component documents, equipment specification and a review of systems. Tr. 4611 (Ianni). Only one change resulted from the 1979 General Electric classification review. Tr. 4631 (Ianni).

b. Assessment of FSAR Table 3.2.1-1

7B:62. Table 3.2.1-1 provides a listing of the safety-related structures, systems and components. This table is reviewed by the various technical branches within the Office of Nuclear Reactor Regulation to determine the correctness and completeness in the area of review responsibility for each branch. Speis et al., ff. Tr. 6357, at 3. The Staff is satisfied that LILCo has used an adequate methodology and that a sufficient set of safety-related items has been identified. Tr. 7603 (Speis).

7B:63. Beyond certain criticisms of Table 3.2.1-1 which are addressed below, Intervenors did not question the adequacy of Applicant's treatment of safety-related plant items within the context of this contention.

7B:64. LILCo's classification table for Shoreham, FSAR Table 3.2.1-1, was attacked by Intervenors as inadequate on several bases. Minor et al., ff. Tr. 1113, at 22-31. Fundamentally, Intervenors alleged that a non-systematic approach to safety classification has been applied to items included in Table 3.2.1-1. Minor et al., ff. Tr. 1113, at 35. Table 3.2.1-1 was included in Intervenor's Attachment 2; hand-marked revisions to Table 3.2.1-1 which were discussed during the oral testimony are included in Intervenor's Attachment 3.

7B:65. The construction of Table 3.2.1-1 was based on Regulatory Guides 1.26 and 1.29. Minor et al., ff. Tr. 1113, at 17. Intervenors alleged that items falling within Quality Group D of Regulatory Guide 1.26 must be designated safety-related. They cited 52 cases where LILCo and General Electric entries allegedly do not match the Regulatory Guide standard. Minor et al., ff. Tr. 1113, at 24. LILCo defended its classification table, saying that its classification of Quality Group D components as LILCo Quality Assurance Category II, Seismic Category NA is consistent with Regulatory Guide 1.26. Burns et al., ff. Tr. 4346, at 161-164.

7B:66. The description of Quality Group D in Regulatory Guide 1.26 does include the term "safety-related". However, the Staff's interpretation and application of its own regulatory guidance does not require that Quality Group D items be classified safety-related. Speis et al., ff. Tr. 6357, at 14. This is made clear by the fact that Standard Review Plan Section 3.2.2 permits use of "the corresponding ANS classification system of safety classes"; ANS-22 establishes classes including a class (corresponding to Category D) which is not a safety-related classification. Burns et al., ff. Tr. 4346, at 162-63.

Intervenors' witness Goldsmith agreed that as much as eight to ten years ago, it was Staff and industry practice that Category D is not considered safety-related. Tr. 1486 (Goldsmith).

7B:67. Forty-nine of the alleged inconsistencies cited by Intervenors in their Table 4-1 disappear because of the fact that Category D of Regulatory Guide 1.26 is not safety-related. Tr. 1498-1500 (Minor).

7B:68. The other three inconsistencies cited by Intervenor are explained by Intervenors as an improper inclusion of nonsafety-related notations within a system (reactor water clean-up) that is classified as a safety-related Category C by Regulatory Guide 1.26. Minor et al., ff. Tr. 1113, at 25.

7B:69. This classification is not improper. The components cited are beyond the reactor coolant pressure boundary and need not be safety-related. This classification is consistent with ANS-22. Burns et al., ff. Tr. 4346, at 165; Speis et al., ff. Tr. 6357, at 14.

7B:70. Intervenors' testimony also argues that there are instances where quality assurance categories are inconsistent with seismic categories. Minor et al., ff. Tr. 1113, at 27. Twenty-four of these are instances involving cable, firestops and waterproof doors, classified as safety-related by quality assurance category but nonsafety-related by seismic category. LILCo's testimony satisfactorily explains the reasons for these seemingly inconsistent classifications. Burns et al., ff. Tr. 4346, at 166-68. For example, Intervenors conceded that if the cable referred to in 22 instances cited is in seismically qualified raceways, the inconsistencies would be largely resolved. Tr. 1502-09 (Minor). Similarly, the seven instances cited by Intervenors as non-safety-related by quality assurance category but safety-related by seismic category are either cited incorrectly by Intervenors or are classified in accordance with specific Staff requirements. Burns et al., ff. Tr. 4346, at 168-69.

7B:71. The remainder of Intervenors' charges may be described as problems with the completeness and the scrutability of the FSAR Table 3.2.1-1. See Minor et al., ff. Tr. 1113, at 27-30. LILCo correctly notes, however, that this table is not a controlling design document and is not required or intended to be a detailed compilation of every structure, system and component at Shoreham. Rather, it is a summary of the classification of principal structures, systems and components, included in the FSAR for the NRC's information. Tr. 4616 (Robare).

7B:72. The table is consistent with the level of detail recommended in ANS-22. Burns et al., ff. Tr. 4346, at 172. Its actual use is necessarily in conjunction with the appropriate Piping and Instrumentation Diagram or other basic design documents. Speis et al., ff. Tr. 6357, at 11, 13; Burns et al., ff. Tr. 4346, at 171. Where the Staff has requested additional information or detail in Table 3.2.1-1, that information has been provided to the Staff's satisfaction. Burns et al., ff. Tr. 4346, at 171-72; Staff Ex. 2A, Supp. No. 1 at 17-1. This includes appropriate expansion of the list to include safety-related items reflected in NUREG-0737, and Applicant has documented its commitment to apply the pertinent requirements of Appendix B to equipment listed in NUREG-0737. Staff Ex. 2A, Supp. No. 1 at 17-1; Speis et al., ff. Tr. 6357, at 15.

75:73. The content and format of Table 3.2.1-1 for Shoreham is consistent with other licensing applications and is at least as detailed as that provided for currently licensed plants. Speis et al., ff. Tr. 6357, at 13.

2. Applicant's Classification and Qualification of Important to Safety but not Safety Related Structures, Systems and Components

a. Application of quality standards and quality assurance requirements generally

7B:74. No list equivalent to Table 3.2.1-1 is provided for structures, systems and components which are important to safety but not safety-related, nor is a listing of these items required by regulation or by the Staff's review process. Speis et al., ff. Tr. 6357, at 9. Such items are, however, addressed throughout the FSAR. See, e.g., FSAR Chapters 3 (plant structures), 7 (instrumentation and controls), 8 (electrical power systems), 9 (auxiliary systems), 10 (steam and power conversion systems), and 11 (radioactive waste management systems).

7B:75. LILCo's witnesses testified that all of the Shoreham plant systems, including nonsafety-related systems, have been examined and evaluated for their significance to total plant function. Both GE and S&W evaluate nonsafety-related items to determine what standards are to be applied based on an evaluation of the component's function and the expected service conditions. Tr. 4441 (Robare, Dawe). The expected service condition for nonsafety-related items includes operation during a transient. Tr. 4440 (Dawe). Nonsafety-related systems are considered to have a very important role in reliable power operation and they are designed, fabricated, erected and tested to quality standards and receive quality assurance commensurate with the goal of a reliable and safe power plant. Burns _et al._, ff. Tr. 4346, at 41.

7B:76. General Electric requires an appropriate degree of engineering design and quality assurance for all structures, systems and components independent of safety classification. The quality assurance requirements for procurement or manufacture of non-safety-related items are specified by the design and quality control engineers based on their evaluation of the function, complexity and importance to reliable power generation as well as to safety where the item has safety relevance. Burns _et al._, ff. Tr. 4346, at 42; Tr. 4435, 4962 (Robare); _see_ Tr. 1319, 1321 (Hubbard). General Electric's operating experience and safety record give it confidence that Shoreham's structures, systems and components are properly classified. Tr. 4933 (Robare).

7B:77. In many instances, General Electric goes beyond regulatory requirements. Engineering judgment is exercised based upon the function of an item in deciding how best to design it and maintain it without restriction to the minimum requirements of the GDC. Tr. 4933-34 (Ianni).

7B:78. The degree of quality assurance typically applied to nonsafety-related equipment within its scope of supply is very close to that applied to the safety-related item under Appendix B. Tr. 4443 (Robare). The specifications applied are based on experience with these nonsafety-related items. Tr. 4444 (Ianni).

7B:79. Stone & Webster also evaluates each structure, system and component within its scope of supply and applies quality assurance commensurate with the item's intended function. Burns et al., ff. Tr. 4346, at 44; Tr. 4395 (Garabedian). Two quality assurance categories are utilized for nonsafety-related items. Id. at 45. Applicable specifications clearly identify the assigned quality assurance category, which is selected based on the function involved. Id. at 45-46. Company organization and procedures are designed to ensure that each specification is complete and correct. Id. at 47. All nonsafety-related items are intended to be designed, procured, constructed and tested in accordance with applicable codes and standards and good design and construction practice. Id. at 47. Although compliance with Appendix B of 10 CFR Part 50 is not required for non-safety-related items,[4/] the principles of a comprehensive quality assurance program which the Appendix B criteria represent are applied to non-safety-related items commensurate with the specific activities performed. Burns et al., ff. Tr. 4346, at 47.

7B:80. Mr. Dawe of Stone & Webster testified that his company applies the same quality assurance program regardless of whether the class or item involved is safety-related or non-safety-related. He considered the constructs of "safety-related" and "important to safety" are "somewhat artificial" for these purposes. What is applied in terms of quality standards and quality assurance is the sophisticated engineering approach that engineers use. One does not stop when Appendix B criteria are met; engineering judgment continues to be applied in deciding what margins to provide or what the level of reliability should be in a design. Tr. 4928-29 (Dawe).

7B:81. LILCo, too, has in place quality programs and requirements for construction activities relating to fabrication and installation of

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4/ Although there is evidence that it was the original intent of the drafters of Appendix B of 10 CFR Part 50 to apply that appendix to all of the plant items to which Appendix A of that part applies (see 46 Fed. Reg. 53618 (1981)), the application of Appendix B has consistently been only to safety-related structures, systems and components. Speis et al., ff. Tr. 6357, at 5; Tr. 5240 (Robare); Tr. 1781 (Hubbard); Tr. 7830 (Speis); Tr. 6967 (Haass). The NRC is working on a proposed rule to expand the list of structures, systems and components subject to Appendix B (see NUREG-0660, Item I.F.1) and to provide regulatory guidance for appropriate quality assurance criteria for important to safety items and has research projects ongoing in support of that effort. Minor et al., ff. Tr. 1113, at 70; Tr. 6980 (Haass); Tr. 7070-71 (Haass); Tr. 7858-59 (Conran, Haass).

nonsafety-related items. Burns et al., ff. Tr. 4346, at 48. LILCo applies quality standards and quality assurance to all structures, systems and components of Shoreham commensurate with their importance to the safe and reliable operation of the plant. Burns et al., ff. Tr. 4346, at 50. Examples were provided in the areas of piping systems, welding procedures, and electrical equipment of the application of industry codes, construction inspections and qualification requirements. Id. at 48-50.

7B:82. The Staff does not review the quality assurance program for items important to safety but not safety-related, nor does it inspect for compliance with such a program. Speis et al., ff. Tr. 6357, at 8-9; Tr. 7063, 7480 (Haass); Tr. 16961, 17288-91 (Higgins).

b. Assessment of specific systems

7B:83. These general descriptions of the treatment of non-safety-related systems by General Electric, Stone & Webster and LILCo were tested by an examination of the treatment of certain specific Shoreham systems. These systems were selected by Intervenors to show that equipment had been misclassified in the design of Shoreham and was not adequate to perform safety-related or important to safety functions, respectively. The systems selected by Intervenors to prove their premise were the standby liquid control system, the turbine bypass system, the reactor core isolation cooling system, the rod block monitor and the level 8 trip.

1) Standby liquid control system (SLC)

7B:84. Intervenor's testimony cited the standby liquid control system as an example of classification deficiencies at Shoreham. In the opinion of Intervenors' witnesses, "the FSAR and SER do not demonstrate that the SLC is properly designed, classified, and qualified." Minor *et al*., ff. Tr. 1113, at 51. Specifically, Intervenors' testimony maintained that the SLC system is or should be a safety-related system but that not all of the vital components of the system are shown by the FSAR to be safety-related. Minor *et al*., ff. Tr. 1113, at 49-50.

7B:85. The SLC system is designed to inject a neutron absorber solution (sodium pentaborate) into the reactor to shut the reactor down from rated power operation to a cold condition in the event that not enough control rods could be inserted to shut down the reactor. Minor *et al*., ff. Tr. 1113, at 48; Burns *et al*., ff. Tr. 4346, at 159; Tr. 1681 (Goldsmith). It provides a diverse, backup means of reactivity control. Burns *et al*., ff. Tr. 4346, at 159; Tr. 4887 (Robare); Tr. 7133 (Hodges).

7B:86. The SLC system was referred to in the FSAR and by LILCo's witnesses as a "special safety system." FSAR Section 4.2.3.4.3; Burns *et al*., ff. Tr. 4346 at 159. Although the SLC is not fully safety-related, LILCo maintains that the SLC meets high quality standards and is properly classified. Tr. 4380-81 (Ianni); Tr. 4880 (Robare).

7B:87. An analysis of ... quality standards applied to the SLC system and the function it performs demonstrates that the system has been properly classified and qualified. First, all of the equipment essential for the injection of the boron solution into the reactor is safety-related equipment. Burns et al., ff. Tr. 4346 at 160; FSAR Section 4.2.3.4.3; Speis et al., ff. Tr. 6357, at 24; Tr. 4888 (Robare). Redundant loops are provided of active equipment necessary for boron injection. These redundant loops are powered by separate power sources capable of being connected to the standby AC power for operation during a station power failure. Burns et al., ff. Tr. 4346 at 160. Operation of the SLC system is manually initiated from the control room. Burns et al., ff. Tr. 4346 at 159; Tr. 4888 (Robare). The switch used to initiate the system is safety-related and the portion of the control board upon which the switch is mounted is designed to survive a seismic occurrence. Speis et al., ff. Tr. 6357, at 24.

7B:88. Non-essential equipment, such as test loop, drain and flush lines and SLC tank heater system, is not safety-related. Nevertheless, these are designed to high standards. The test loop, drain and flush lines are isolated from the main loops by safety grade isolation valves to assure integrity of the main loops. The tank heater system consists of redundant heaters, one automatically controlled by the tank temperature monitoring system and the other a larger manual heater. Burns et al., ff. Tr. 4346, at 159-60.

7B:89. Intervenors criticized the non-safety-related classification of the tank heaters because of the possibility that cooling of the solution could cause precipitation of the sodium pentaborate thereby defeating the successful function of the system. Several design features assure the reliability of the system. Constant temperature indication is given to the operator. Tr. 4897-98 (Robare). There is an alarm on one of the temperature sensors which is set 11 degrees above the temperatures at which the sodium pentaborate would precipitate out of the solution. Tr. 1682 (Minor); Tr. 4899 (Dawe); Burns <u>et al</u>., ff. Tr. 4346, at 160-61. There is a back-up heater. Tr. 4897-98 (Robare). The heaters are not the only thing that maintains the temperature of the solution. Tr. 1680-81 (Goldsmith). The ambient temperature is normally high enough (generally at least 70 degrees F.) in the vicinity of the tank that precipitation in the solution would be prevented even without operation of the tank heaters. Burns <u>et al</u>., ff. Tr. 4346, at 160-61; Tr. 4899 (Dawe); Tr. 4897-98 (Robare). Finally, tank solution contents, concentration and temperature are to be monitored at least once every 24 hours under proposed Shoreham Technical Specification 4.1.5. Burns <u>et al</u>., ff. Tr. 4346, at 160-61; Tr. 4897-98 (Robare). Even if the tank heaters were to fail, the solution would remain at a high enough temperature to prevent precipitation of the sodium pentaborate for at least 24 hours, during which time the tanks would be checked. Tr. 4899 (Robare).

7B:90. Again, the function of the SLC system is to provide a back-up, diverse means of shutting the reactor down during normal operation. The SLC system is not required to be redundant because it is only a back-up system. Tr. 7133 (Hodges); Speis _et al._, ff. 6357, at 25. The reactor protection system itself is redundant. Tr. 7135 (Hodges). The SLC system is not required for safe shutdown in terms of Appendix A to 10 CFR Part 100. Tr. 4879-81 (Robare). It is not used to mitigate any design basis accident. Tr. 4882-83 (Dawe).

7B:91. The Board finds that the SLC system is properly classified. The Board is satisfied that the SLC system need not be classified in its entirety as safety-related and that it has been properly designed and qualified to standards commensurate with the importance of its backup safety function as required by GDC-1. The design and operational requirements established for this system demonstrate that an adequate methodology has been applied with respect to it.

2) Turbine bypass system

7B:92. Intervenors point to the turbine bypass system as a system the function of which is sufficiently important that it should be classified as safety-related. The fact that it is not classified as safety-related is said to be "another example of the inadequate classification methodology utilized by LILCo for Shoreham." Minor _et al._, ff. Tr. 1113, at 40.

7B:93. The turbine bypass system is used during normal startup and shutdown to pass partial steam flow to the condenser. The turbine bypass valves also operate automatically following a turbine trip or load rejection. Following a turbine trip or a generator load rejection, the turbine stop valves or the turbine control valves will close immediately to stop the steam flow to the turbine. The accumulation of steam in the vessel pressurizes the reactor. The turbine bypass valves are designed to open automatically under such conditions in order to reduce the pressurization rate by directing some steam (25% of full power) to the condenser. Burns et al., ff. Tr. 4346, at 146.

7B:94. The turbine bypass system is described in Section 10.4.4 of the FSAR. As discussed there, it consists of two steam lines from the main steam header to the bypass valve chest, four bypass valves, and four steam lines to the condenser, each including a pressure reducer at the condenser connection. The bypass valves are controlled by the turbine generator electrohydraulic control (EHC) system. The power supply to the control system is from 120 VAC uninterruptable instrument and control power for high reliability and plant availability. This power source, although not safety-related, is available following loss of offsite power. In addition, an alternate power source is provided from a shaft driven permanent magnet generator supplied with the main turbine. Burns et al., ff. Tr. 4346, at 147; Tr. 4758-59 (Dawe, McGuire).

7B:95. The steam lines up to, but not including, the turbine bypass valves are Quality Group B, QA Category I, Seismic Category I (Table 3.2.1-1, item XXXI.3). The turbine bypass valves are Quality Group D, QA Category II, Seismic Category NA (Table 3.2.1-1, Item XXXI.5). The turbine bypass valves are, however, subject to the extensive quality assurance program of the supplier, General Electric, Large Steam Turbine Generator, (GE-LSTG). This program, which the Staff considers to be at a level equally equivalent to Quality Group B, is documented in GE-LSTG publication GES-4982A, "General Electric Large Steam Turbine Generator Quality Assurance Program." The EHC system is also subject to GEZ-4982A. The bypass system piping downstream of the bypass valves is not safety-related. It is designed, inspected and tested in accordance with ANSI B31.1. Burns et al., ff. Tr. 4346, at 147-48.

7B:96. This design is in compliance with Regulatory Guide 1.26, Revision 1. It also complies with Regulatory Guide 1.26, Revision 3, including footnote 5. The NRC Staff, in Appendix A to Standard Review Plan Sec. 3.2.2 (Attachment 7), has presented its position with respect to main steam components for BWR plants such as Shoreham. The Shoreham turbine bypass system, as described above, complies with the Branch Technical Position incorporated in the Standard Review Plan. Burns et al., ff. Tr. 4346, at 148.

7B:97. Should the bypass valves fail to open, reactor vessel pressure would be somewhat higher and the transient impact on the fuel would be increased. Analysis at full power conditions shows, however, that bypass failure would increase the change in Critical Power Ratio (CPR), an index relating to the reactor fuel heat transfer capability, by less than 0.08. The overall effect is a slight reduction of the fuel heat transfer capability. However, the majority of the fuel is still maintained well above the CPR limit criteria. The resulting dose effect (if any) does not approach a small fraction of the 10 CFR Part 100 criteria. Burns et al., ff. Tr. 4346, at 146-47.

7B:98. General Electric utilizes special standards and procedures for the design, manufacture, procurement and testing of the turbine generator system as opposed to existing codes and standards for products intended for more general service. These include such measures as detailed design procedures, material certification, subvendor inspection, in process quality control, audits, and record keeping. The program also includes nonconformance documentation and engineering disposition. Burns et al., ff. Tr. 4346, at 148.

7B:99. The turbine bypass system was field-erected under the supervision of GE-LSTG, received quality control under the Shoreham Construction Site Inspection Program, and is subjected to a preoperational test program as opposed to acceptance tests. Burns et al., ff. Tr. 4346, at 148.

7B:100. The use of preoperational testing rather than acceptance testing is indicative of the additional treatment given the turbine bypass system in recognition of its function even though it is not safety-related. The bypass system is also subjected to the start-up test program. The testing philosophy and procedure for Shoreham as well as specific tests involving the turbine bypass system, are summarized in Chapter 14 of the FSAR. Burns et al., ff. Tr. 4346, at 148.

7B:101. In addition to careful design, procurement, installation, and testing of the turbine bypass system, plant operation is subject to operability of the turbine bypass system by Technical Specification 3.7.10. Burns _et al._, ff. Tr. 4346, at 148-49.

7B:102. For its part, the Staff gives special consideration to the turbine bypass system through the requirement of a technical specification ordering periodic surveillance to confirm the operability of the turbine bypass system. Speis _et al._, ff. Tr. 6357, at 27; Staff Ex. 2A, ¶ 7.6.11.

7B:103. The Board finds that the turbine bypass system is not improperly classified. The Board is satisfied that the turbine bypass system need not be classified in its entirety as safety-related and that it has been properly designed and qualified to standards commensurate with the importance of its safety function. The design and operational requirements established for this system demonstrate that an adequate methodology has been applied with respect to it.

3) Reactor core isolation cooling (RCIC)

7B:104. Intervenors maintain that the RCIC, as a back-up for the High Pressure Coolant Injection (HPCI) system, should be classified as safety-related in its entirety. Failure so to classify the RCIC is cited as further evidence of the alleged inadequacy of LILCo's classification methodology. Minor _et al._, ff. Tr. 1113, at 40.

7B:105. The RCIC system is a high pressure system which provides core cooling during reactor shutdown by pumping makeup water into the reactor vessel in case of a loss of flow from the main feedwater system. It can also supplement the HPCI system by providing coolant makeup at high pressure conditions. Burns et al., ff. Tr. 4346, at 143; Speis et al., ff. Tr. 6357, at 25; Tr. 4806, 4807, 4813 (Robare). During a loss-of-coolant accident (LOCA), the RCIC initiates on low vessel water level and delivers rated flow to the vessel through a connection in the feedwater system. RCIC is not a part of the Emergency Core Cooling System (ECCS) network. It is similar to the auxiliary feedwater systems in PWRs. During limiting conditions of operation (LCO) (i.e., when HPCI is inoperable), power operation is allowed to continue for a period of time provided RCIC is operable. Moreover, credit is taken for RCIC when HPCI is inoperable in part of the Shoreham accident analysis (e.g., control rod drop accident.) Speis et al., ff. Tr. 6357, at 25; Staff Ex. 2A, § 7.4.1.

7B:106. Almost all of the RCIC system is classified as safety-related; all of the equipment necessary for the RCIC system to perform its intended safety function of automatically injecting water is safety-related. Burns et al., ff. Tr. 4346, at 144; FSAR Table 3.2.1-1; Speis et al., ff. Tr. 6357, at 25; Tr. 7486-87 (Hodges). In the opinion of Mr. Robare, GE could change the classification of the RCIC to safety-related notwithstanding that certain portions of the system are not safety-related because those portions are not pertinent to the safety function. Tr. 4815 (Robare).

7B:102. The only significant area in which the system is not safety-related is in its control and instrumentation. Even there, many aspects are safety-related. The system components which provide the safety functions of detecting low level and injecting water into the vessel are qualified for safety-related operations. The safety functions of the control and instrumentation are also designed in accordance with safety system criteria. Moreover, the RCIC system is separated in a completely different electrical division from the HPCI system. Burns et al., ff. Tr. 4346, at 144.

7B:108. The unqualified components of the RCIC include the barometric condensor whose failure would not preclude systems operation and four control room indicators whose failure would not impact the automatic operation of RCIC. The only other aspect of the RCIC design which does not meet full safety-related criteria is the single channel high level trip which prevents overfill of the reactor vessel. This does not affect the operation of the safety function of automatically injecting water. Burns et al., ff. Tr. 4346, at 144.

7B:109. Although the RCIC system is less reliable than the emergency core cooling systems (ECCS), no credit is taken for the RCIC in arriving at the ECCS criteria in the loss of coolant analysis. Tr. 7130-31 (Speis, Hodges).

7B:110. The Board finds that the RCIC system, which is very nearly completely safety-related, is not improperly classified. The Board is satisfied that not all of the RCIC needs to be classified as safety-related and that the system has been properly designed and qualified to standards commensurate with the importance of its safety function. The design and operational requirements established for this system demonstrate that an adequate methodology has been applied with respect to it.

4) Rod block monitor (RBM)

7B:111. Intervenors' witnesses testified that the rod block monitor should be, but is not, classified as safety-related because of the importance of its function. LILCo's methodology for classification is criticized because of this alleged failure properly to classify the rod block monitor. Minor et al, ff. Tr. 1113, at 40.

7B:112. The rod block monitor, together with two other systems, performs the rod block function, which is designed to prevent erroneous withdrawal of a control rod during normal operation possibly resulting in local fuel damage. The rod block monitor initiates a signal to the rod drive control system to stop drive motion. The principal objective of the rod block monitor is to increase fuel life by restricting rod movement to minimize local flux peaking. The rod block monitor does not mitigate the control rod drop or any other accident; local fuel damage caused by failure of the rod block function would pose no significant threat of radioactive release. Burns et al., ff. Tr. 4346, at 141, 143; Tr. 4798-99 (Robare); Tr. 4994-95 (Dawe); Tr. 4795 (McGuire).

7B:113. The rod block monitor is not required to perform the critical safety functions of 10 CFR Part 100, Appendix A. Tr. 4787-88, 4791 (Robare).

7B:114. The rod block monitor is not a safety-related system. Speis _et al_., ff. Tr. 6357, at 27; Burns _et al_., ff. Tr. 4346, at 142. Nevertheless, special design features and other considerations have been applied to the rod block monitor to assure its reliability. Speis _et al_., ff. Tr. 6357, at 27.

7B:115. The system meets most design principles of safety-related systems. It is redundant in that two channels of information must agree before rod motion is permitted (only one of the RBM channels is required to trip to prevent rod motion). The system has self-monitoring features with provisions to check the self-monitoring. Loss of power to the RBM will cause a rod block. Burns _et al_., ff. Tr. 4346, at 142.

7B:116. The following features are included in the RBM design:

a. Redundant, separate, and isolated RBM channels.

b. Redundant, separate, isolated rod selection information, including isolated contacts for each rod selection push button, are provided directly to each RBM channel.

c. Separate, isolated LPRM amplifier signal information is provided to each RBM channel.

d. Separate and electrically isolated Average Power Range Monitor reference signals are provided each RBM channel.

e. Independent, separate, isolated Average Power Range Monitor reference signals are provided each RBM channel.

f. Independent, isolated RBM level readouts and status displays are provided from the RBM channels.

g. There is a mechanical barrier between channel A and channel B of the manual bypass switch.

h. Independent, separate, isolated rod block signals are provided from the RBM channels to the manual control system circuitry. Burns et al., ff. Tr. 4346, at 142; Tr. 4803 (Robare).

7B:117. In addition to the high quality of the rod block monitor design, technical specification surveillance requirements are to be imposed further to assure rod block function operability. Burns et al., ff. Tr. 4346, at 143. The system has a self-testing feature, the operability of which must be demonstrated periodically. Speis et al., ff. Tr. 6357, at 27; Burns et al., ff. Tr. 4346, at 143; Staff Ex. 2A, ¶¶ 7.6.4, 7.6.11. In addition, a technical specification will require that the rod block monitor be operable at above 30 percent of rated power. Tr. 4798-99 (Robare).

7B:118. The Board finds that the rod block monitor is properly classified. The Board is satisfied that the rod block monitor need not be classified as safety-related and that it has been properly designed and qualified to standards commensurate with the importance of its limited safety function. The design and operational requirements established for this system demonstrate that an adequate methodology has been applied with respect to it.

5) Level 8 trip

7B:119. Intervenors' witnesses described the function of the level 8 trip as "to warn the operators of possible overfilling of the vessel . . ." They asserted that the system should be classified as safety-related and that the failure so to classify it is "another example of the inadequate classification methodology utilized by LILCo for Shoreham." Minor et al., ff. 1113, at 40.

7B:120. The level 8 trip signal automatically trips the turbine and shuts down the feedwater pumps in the event that an excess of feedwater reaches the high water level (level 8) trip setpoint. Burns et al., ff. 4346, at 145. It is one line of defense against a feedwater controller failure transient, in which feedwater controller function is lost and a maximum feedwater flow is erroneously initiated. If the level 8 trip should fail, turbine trip would be delayed until manual operator action is taken or until an increase in wet steam causes increased vibration which induces turbine trip. Burns et al., ff. Tr. 4346, at 145. The consequences of failure of the level 8 trip on transient severity are not significant. Burns et al., ff. Tr. 4346, at 145, 146.

7B:121. The level 8 trip is not safety-related. Speis et al., ff. Tr. 6357, at 27. It is, however, a high quality designed and manufactured system having significant tolerance to single failures. There are 3 trip channels with independent power supplies, two on battery busses and one on a 120 VAC instrument bus, so that any single electrical failure is tolerated without any effect on system functions. The vessel water level differential pressure transmitters and other instrumentation and control components associated with the level 8 feedwater pump trip, though not classified safety-related, are identical in design and manufacture to the fully safety-related components associated with the ECCS and RPS low vessel water level trips. Burns et al., ff. Tr. 4346, at 145-146; Tr. 4819 (Robare).

7B:122. The technical specifications will limit the time during which portions of the level 8 trip system may be inoperable. Speis et al., ff. Tr. 6357, at 27. Periodic surveillance requirements of the operability of the level 8 trip will be included in the technical specifications. Staff Ex. 2A, ¶ 7.6.11. It is on the basis of this high reliability and the technical specification requirements, together with the fact that the consequences of failure do not result in undue risk to public health and safety, that use of the level 8 trip is permitted in mitigation of the feedwater controller failure transient even though the system is not safety-related. Speis et al., ff. Tr. 6357, at 23-24; Staff Ex. 2A, ¶ 7.6.11.

7B:123. The Board finds that the level 8 trip is properly classified. The Board is satisfied that the level 8 trip need not be classified as safety-related and that it has been properly designed and qualified to standards commensurate with the importance of its limited safety function. The design and operational requirements established for this system demonstrate that an adequate methodology has been applied with respect to it.

7B:124. Intervenors have failed to prove that misclassification exists in the systems they selected, and thus have not proved that the design and review methodology applied at Shoreham was inadequate as alleged.

c. Assessment of emergency operating procedures review

7B:125. Intervenors' witnesses conducted a review of certain emergency operating procedures to identify equipment called upon therein. Minor et al., ff. Tr. 1113 at 31-38.5/ On the basis of this review, they concluded that "several key systems and/or components are separately called upon to assist in the mitigation of accidents, although such equipment has not been required to meet either the 'safety-related' quality standards as described in Table 3.2.1-1, or some other standards consistent with the GDC and the safety functions to be performed." The purpose of this testimony was to test the adequacy of LILCo's methodology in support of Contention 7B. It also relates to SOC Contention 19(b)(4), which states more unambiguously that LILCo has failed to include in Table 3.2.1-1 "equipment upon which the plant operators will rely in response to accidents."

7B:126. Emergency operating procedures in many instances direct an operator to call upon equipment which is not safety-related. The

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5/ Intervenors' witness Harwood, who was principally responsible for this review, has never been involved in the analyses or critique of emergency operating procedures for a specific nuclear power plant. Tr. 1275 (Harwood).

inclusion of the non-safety-related systems in these procedures is based on the principle that operators should be directed to use all available systems including the use of the normal, non-safety-related systems. Burns *et al.*, ff. Tr. 4346, at 139-40. It is expected that an operator will use the non-safety-related equipment which remains operable to the maximum extent possible in controlling the course of any accident. Speis *et al.*, ff. Tr. 6357, at 22. However, where a non-safety-related system is called upon in the emergency procedures, there is a safety-related system capable of preventing core damage in the event the non-safety-related system fails. Burns *et al.*, ff. Tr. 4346, at 139; Speis *et al.*, ff. Tr. 6357, at 26.

7B:127. Any equipment cited in an emergency procedure which is *necessary* to assure the critical safety functions of 10 CFR Part 100, Appendix A is classified safety-related. Speis *et al.*, ff. Tr. 6357, at 22.

7B:128. An example of a non-safety-related system being called upon by an emergency procedure is the plant feedwater system. The operator is very familiar with this particular system and would use it during a loss of coolant accident if it is available. It is not, however, necessary that the system be safety-related even though it might be used during an accident because other items which are safety-related are available to protect public health and safety. Speis *et al.*, ff. Tr. 6357, at 26.

7B:129. Emergency operating procedures have received special attention and review since the TMI-2 accident. The BWR Owners' Group Systems Subgroup, for example, undertook an assessment of emergency procedures and the capability of BWR systems to handle abnormal events, including multiple failures. As a result of this review, the Subgroup recommended development of simple, complete procedures so that operators can use the full capabilities of the plant, safety-related as well as non-safety-related, in dealing with problems that arise. Emergency procedure guidelines have been developed as a result of the Subgroup's recommendations. As the emergency procedure guidelines are an operator's logical approach to dealing with the symptoms presented by an abnormal occurrence, they typically start with normally used non-safety-related systems. If failures progress in non-safety-related equipment, the safety-related equipment comes into play. The current Shoreham emergency operating procedures are consistent with the recommendations of the Subgroup. Burns et al., ff. Tr. 4346, at 130-32. All emergency operating procedure accident or transient scenarios, however, are bounded ultimately by a safety-related system. Id. at 133.

7B:130. The Board has been pointed to no regulatory requirement that all equipment specified for use in emergency operating procedures be classified as safety-related and finds that there is no such requirement. Speis et al., ff. Tr. 6357, at 21. Further, given the purpose of calling upon non-safety-related equipment in emergency operating procedures, the Board finds that the use of such equipment for the mitigation of abnormal occurrences is not itself a reason for requiring that such equipment be classified as safety-related.

3. Resolution of "important to safety" definitional controversy

7B:131. There is no evidence that Applicant's improper use of the term "important to safety" has had a substantive impact on the design and construction of the Shoreham plant. Staff's witnesses testified specifically that they were aware of no specific example of a substantive difference in the plant caused by the definitional issue. Applicant's witnesses testified specifically that no such substantive differences exist. Tr. 4422-23, 4472-73 (Dawe); Tr. 7815 (Speis et al.); Tr. 6958-61 (Conran*).

7B:132. There appears to be close agreement between most important aspects of the respective positions and conclusions of the Staff and Applicant regarding adequacy of safety classification of Shoreham plant features, particularly as to the substantive technical safety classification considerations at issue. Conran, ff. Tr. 6368, at 2.*

7B:133. Even though Applicant did not use the term "important to safety" properly, by putting together an FSAR and addressing the criteria for structures, systems and components called for in the Standard Review Plan, Applicant has satisfied the Staff's requirements for items important to safety. Tr. 7495-96 (Conran);* see Tr. 6537 (Conran)*; Speis et al., ff. Tr. 6357, at 10.*

7B:134. The Staff's review process verifies that plant items important to safety meet the Staff's requirements as outlined in the Standard Review Plan. Tr. 6974-75 (Haass).

7B:135. Because the Standard Review Plan ensures that important to safety items have been addressed, the Staff does not perceive a need to re-review the FSAR despite the difference in Applicant's use of the language of the regulations. Tr. 7121-23. (Rossi, Hodges, Haass, Kirkwood). The Staff's review was conducted according to the Standard Review Plan by examining the function of particular systems and the requirements for that function. Tr. 7122-23 (Hodges).

7B:136. The Staff identified certain "unacceptable implications" of Applicant's incorrect use of "important to safety":

1. Because the Staff conducts an audit review, reliance must be placed on commitments by Applicants that all portions of the regulations are complied with (see, e.g., FSAR § 3.1.2.1). It is critical that these commitments mean what the Staff understands them to mean if the Staff's determination of "reasonable assurance" (which finding must be made in accordance with 10 C.F.R. § 50.35(c) in order to license a facility) is to be meaningful in the sense intended in the regulation.

2. It is clear under the Staff's understanding of "important to safety" (but not under Applicant's) that there exists in the regulations a requirement under GDC 1 for a QA program for certain non-safety-related structures, systems and components (i.e., those important to safety).

3. Under Applicant's constructure of "important to safety," the obligations imposed by 10 C.F.R. Part 21 might be more narrowly construed than would be the case under the Staff's broader definition of that term. Conran, ff. Tr. 6368, at 6-7.

7B:137. The Board agrees with the Staff that it is critical to the licensing and regulation of a nuclear power reactor that regulatory terms have a common meaning to the parties involved. See Tr. 7728 (Rossi).

7B:138. Section 3.1 of the FSAR contains a commitment by LILCo to comply with GDC-1 as follows:

> The detailed QA program developed by Long Island Lighting contractors satisfies the requirements of Criterion 1.

Because LILCo has equated the terms "important to safety" and "safety-related" in its FSAR commitments, this specific commitment was intended to relate only to safety-related plant items. See Tr. 4470, 4485 (Dawe).

7B:139. The Staff considered this a commitment which included important to safety plant items; in the Staff's view, GDC-1 applies to the entire broader class. Tr. 7080 (Rossi); Tr. 16960 (Higgins).

7B:140. Applicant's prefiled and oral testimony has described the specific quality assurance program applied to structures, systems and components to the Staff's satisfaction. Tr. 6974-75 (Haass).

7B:141. The Staff testified that it considers it necessary to obtain reconfirmation of Applicant's commitment using the correct definition of important to safety. Tr. 7122-23 (Conran,* Haass).

D. ANALYSIS OF SYSTEMS INTERACTIONS AT SHOREHAM

7B:142. The Staff witnesses defined adverse systems interactions as "the possibility of one reactor plant system acting on one or more systems in a way not consciously intended by design so as to adversely affect the safety of the plant." Speis et al., ff. Tr. 6357, at 34. We accept this definition.

1. Assessment of Applicant's Analysis of Systems Interactions

7B:143. Extensive discussion was provided in the Applicant's prefiled testimony concerning the organization and operation of the nuclear steam supply system vendor, General Electric, and the architect engineer, Stone & Webster, and the way in which systems interactions are addressed throughout the design process. Burns et al., ff. Tr. 4346, at 8-27.

7B:144. General Electric has a philosophy it calls "design discipline" to assure the safe and reliable operation of its products and services. Documented practices and procedures incorporate measures to assure that design activities, instructions and procedures, document control, purchasing, material control, process control and inspection activities are carried out in a planned, controlled and orderly manner. Burns et al., ff. Tr. 4346, at 9. Design documents are distributed to affected design organizations for information, review and coordination in order to assure interface compatibility and minimize opportunities for adverse interactions between and among systems. Burns et al., ff. Tr. 4346, at 10.

7B:145. Designs are subject to independent design verification within the various engineering organizations of GE. Burns et al., ff. Tr. 4346, at 11-12. In this way, all design aspects affecting a given system, including interface with other systems, are considered. Id. Teams of persons other than those directly responsible and accountable for the design conduct a formal evaluation of a design. Burns et al., ff. Tr. 4346, at 13. Control procedures require that design changes be documented, verified, approved and reviewed appropriately. Burns et al., ff. Tr. 4346, at 14. Complex design changes affecting multiple design groups are reviewed by a standing Change Control Board to assure that interfaces are properly addressed. Extensive assessments of systems interactions are made throughout this process by virtue of the knowledge and experience of the engineers involved. Burns et al., ff. Tr. 4346, at 15.

7B:146. General Electric has design and operating experience in the nuclear industry since 1946 involving 41 nuclear power plants in operation today and another 30 in design and construction. In General Electric's view, all of this design and operating experience has been brought to bear on Shoreham and it provides confidence that undetected adverse systems interactions of safety significance are unlikely to exist at Shoreham. Burns et al., ff. Tr. 4346, at 15-20.

7B:147. Stone & Webster's organization, procedures and experience have been brought to bear on Shoreham to anticipate and avoid, through appropriate plant design, those systems interactions that could interfere with the safe operation of the plant. Burns et al., ff. Tr. 4346, at 20-27. Stone & Webster has been involved in nuclear power plant design and construction for over 20 years; it believes that the practices and procedures that it has evolved during that time contribute to its ability to anticipate, properly consider, and account for potential systems interactions in the design process. Burns et al., ff. Tr. 4346, at 23-24. The design of systems and the evaluation of a system's function includes an evaluation of interactions associated with that system. Tr. 5142 (Dawe). Systems are looked at not only for their own functions, but also for their relationship in the plant to other things around it. Tr. 4463 (Dawe).

7B:148. Beyond the basic process used by General Electric and Stone & Webster for the design, manufacture and installation of systems, structures and components at Shoreham, a number of specific systems interaction studies and programs have been conducted which relate specifically to Shoreham. The specific examples of systems interaction studies cited by Applicant's witnesses included the following:

(1) pipe failure and internal flooding (Burns et al., ff. Tr. 4346, at 56; Tr. 5043-44, 5052-53, 5059-10, 5065 (Dawe));

(2) missiles (Burns et al., ff. Tr. 4346, at 57; Tr. 5073-74 (Dawe, Robare), Tr. 5070, 5077-79 (Dawe));

(3) fire hazard analysis (Burns et al., ff. Tr. 4346, at 57; Tr. 5087-5104 (Dawe));

(4) cable separation (Burns et al., ff. Tr. 4346, at 57; Tr. 5104-5110, 5567-70 (Dawe));

(5) failure modes and effects analyses (Burns et al., ff. Tr. 4346, at 58; Tr. 5113-17 (Dawe));

(6) electrical bus failures (Burns et al., ff. Tr. 4346, at 58; Tr. 5121, 5123, 5126 (Dawe));

(7) control system failures (Burns et al., ff. Tr. 4346, at 59; Tr. 5129-30 (Dawe));

(8) high energy line break (Burns et al., ff. Tr. 4346, at 59-60; Tr. 5144-47 (Dawe, Robare));

(9) PRA relating to plants other than Shoreham (Burns et al., ff. Tr. 4346, at 60; Tr. 5147-53 (Robare); Tr. 5164-65 (Ianni));

(10) heavy loads (Burns et al., ff. Tr. 4346, at 60; Tr. 5171-72 (Dawe));

(11) protection systems (Burns et al., ff. Tr. 4346, at 63; Tr. 5227-32 (Robare));

(12) scram reliability (Burns et al., ff. Tr. 4346, at 63; Tr. 5248-318 (Robare, McGuire));

(13) common mode failures in protection and control instrumentation (Burns et al., ff. Tr. 4346, at 64; Tr. 5321-29 (Robare));

(14) water level instrumentation (Burns et al., ff. Tr. 4346, at 64; Tr. 5336 (Robare));

(15) TMI-2 implications (Burns et al., ff. Tr. 4346, at 64; Tr. 5384-86, 5400 (Robare));

7B:149. Walkdown techniques were also utilized to attempt to identify spatial dependencies among systems as a part of the Shoreham probabilistic risk assessment. Burns et al., ff. Tr. 4346, at 102-103.

7B:150. The studies cited in the testimony are a sampling of the major studies that were formally conducted as part of the design process. More systems interaction studies have been done than are cited there. Tr. 5243 (Robare). In Applicant's view, the results of these various systems interaction studies demonstrate that potential interactions are adequately considered in the design and construction process because no significant or fatal flaws (as opposed to design enhancements) were found. Tr. 5084 (Dawe). In the final analysis, however, it is the comprehensive design process, rather than specific types of individual studies, that assures good design. Tr. 5292-94 (Rigelhaupt).

7B:151. LILCo has established a group, known as the Independent Safety Engineering Group ("ISEG"), to be responsible for the continuing review and application of data from licensee event reports, significant event reports and significant operating experience reports. Incidents involving systems interactions will be identified and evaluated. Burns et al., ff. Tr. 4346, at 61; Tr. 5524 (Kascsak).

7B:152. The Board finds that extensive evaluation has been conducted by General Electric and Stone & Webster of potential adverse systems interactions at Shoreham. This evaluation has included both deterministic and probabilistic methodologies. Major parts of this evaluation are documented in the FSAR; other parts, such as the probabilistic risk assessment, have been conducted independent of any regulatory requirement. See Findings 205, 210, 212, 217.

2. Water Level Indication System Interactions

7B:153. Intervenors' witnesses selected the water level indication system (WLI) to show that Shoreham is subject to systems interactions in a way that allegedly demonstrates the inadequacy of LILCo's methodology for analyzing the adequacy of plant design. They testified that water level measurement, the reliability of which is said to be critical, can be adversely affected by a combination of high drywell temperature and low reactor vessel pressure to the point that emergency core cooling could be delayed. Minor et al., ff. Tr. 1113, at 42-43. In Intervenors' view, "the existing analysis and review techniques as documented in the FSAR and SER failed to discover this problem . . . ." Minor et al., ff. Tr. 1113, at 47. As detailed below, Intervenors have failed to demonstrate through the system they chose that there is any inadequacy in the methodology utilized in terms of analyzing systems interactions.

7B:154. Figure 1 of LILCo Attachment 9 illustrates one of the two sets of cold reference leg reactor water level measurement instrumentation provided at Shoreham. Reactor vessel water level is measured by differential pressure transmitters which measure the difference in static head between two columns of water. One column is a "cold" (ambient temperature) reference leg outside the reactor vessel; the other is the reactor water inside the reactor vessel and the variable leg. The measured differential pressure is a function of reactor water level. Burns et al., ff. Tr. 4346, at 150. The WLI is largely a safety-related system. All portions of the system that are used in tripping the reactor are safety-related. Tr. 6836 (Rossi). In general, the portions of the WLI system used for protection are safety-related; those portions used for control are nonsafety-related. Tr. 6837 (Rossi). The WLI reference leg is classified safety-related. Tr. 1822 (Goldsmith).

7B:155. The cold reference leg is filled and maintained full of condensate water by a condensing chamber at its top which continuously condenses reactor steam and drains excess condensate back to the reactor vessel through the upper level tap connection to the condensing chamber. The upper vessel level tap connection is located in the steam zone above the normal water level inside the vessel. Thus, the reference leg presents a constant reference static head of water on the high pressure tap of the transmitter. The low-pressure tap of the transmitter is piped to a lower-level tap on the reactor vessel which is located in the water zone below the normal water level in the vessel. The low-pressure side of the transmitter thus senses the static head of water/steam inside the vessel above the lower vessel level tap. This head varies as a function of reactor water level above the tap and is the "variable leg" in the differential pressure measured by the transmitter. Lower taps for various instruments are located at various levels in the vessel water zone to accommodate both narrow and wide-range level measurements (see Figure 2 of LILCo Attachment 9). Burns et al., ff. Tr. 4346, at 150-51.

7B-156. Reactor level indicators and recorders are shown on Figure 3 of LILCo Attachment 9. This figure also shows the condensing chamber.

Shoreham level instrumentation, including elevations and setpoints, are shown in Figure 4 of LILCo Attachment 9. Burns et al., ff. Tr. 4346, at 151.

7B:157. All parties agree that high drywell temperature can cause boil-off or flashing of the water in the reactor water level sensing lines if the reactor is depressurized while these high temperatures exist. Burns et al., ff. Tr. 4346, at 154. Such high drywell temperatures can be caused in several ways. Intervenors cite small (e.g., 0.01 sq. ft.) and intermediate (e.g., 0.04 sq. ft.) break LOCA's which discharge hot steam into the drywell over an extended time period. Minor et al., ff. Tr. 1113, at 45. The Staff raises a similar situation resulting from a large break LOCA. Speis et al., ff. Tr. 6357, at 28.

7B:158. Even without a pipe break, loss of containment coolers can cause the containment to heat up and cause flashing as occurred at Pilgrim Nuclear Station. Minor et al., ff. Tr. 1113, at 45-46; Speis et al., ff. Tr. 6357, at 28. In each case, subsequent depressurization may cause some loss of water in the level sensing lines. In the Staff's words, "[t]here is the potential for flashing whenever the reactor coolant system (RCS) pressure drops below the saturation pressure corresponding to the temperature near the reference leg." Speis et al., ff. Tr. 6357, at 28. Loss of water in the level sensing lines, through flashing or otherwise, could result in a false high indication when core water level actually is low. Minor et al., ff. Tr. 1113, at 43. The potential for high drywell temperature to cause errors through flashing and boil-off was evaluated by the Staff in the Shoreham SER, Supp. 1 at § 7.3.8. Tr. 7806-07 (Hodges)

7B:159. There is ample evidence in the record that the loss of water in the water level sensing lines and resultant erroneous water level indication does not create undue risk to public health and safety at Shoreham. First, drywell temperature is maintained by cooling equipment and the performance of this air cooling system is monitored. Drywell air temperature is maintained during all normal plant operations by two unit coolers, each with four cooling coils and fans. The reactor building closed loop cooling water (RBCLCW) system is the cooling medium for the cooling coils. Although the drywell air cooling system is not safety-related, the fans, dampers and valves receive power from emergency power supplies to provide continued operation following a loss of offsite power with no accident signal present. The system is automatically shut down and isolated on an accident signal. Burns et al., ff. Tr. 4346, at 152. Drywell air cooling system performance is monitored in the main control room. Alarms are provided for a number of parameters, including various area and exhaust high temperatures, RBCLCW return high temperature, and unit cooler high supply air temperature. In addition, primary containment air temperature is monitored by temperature instruments located throughout the drywell. Shoreham proposed Technical Specification 3.6.1.7 (LILCo Attachment 8) requires initiation of plant shutdown if the containment average air temperature cannot be reduced to below 135° within 8 hours. The proposed Technical Specifications have been submitted to the NRC. Burns et al., ff. Tr. 4346, at 153.

7B:160. The maximum water level measurement error is of little or no direct safety significance at Shoreham. According to uncontroverted testimony of Applicant's witnesses, the maximum water level measurement error that could occur when the reactor is at rated pressure and temperature conditions would be less than six inches. Burns et al., ff. Tr. 4346, at 153-54. When the reactor is depressurized, the maximum water level measurement error increases. According to General Electric analyses for a worst case scenario, a maximum measurement error of 1.9 feet would result if the operators follow plant operating procedures. Burns et al., ff. Tr. 4346, at 154-56. Failure of the operators to follow plant operating procedures for refilling the reference legs by flooding the reactor and for initiating the drywell spray system could result in additional flashing and boil-off over a ten-hour period causing a maximum water level measurement error of approximately 9 feet. Burns et al., ff. Tr. 4346, at 156-157; Speis et al., ff. Tr. 6357, at 29-30. Intervenors' expert agreed that the 9 foot error is the maximum for the high drywell temperature depressurization situation they cited. Tr. 1666 (Goldsmith). The normal water level is approximately 16 feet above the top of the fuel. Speis et al., ff. Tr. 6357, at 30; Burns et al., ff. Tr. 4346, at 157; Tr. 1662 (Goldsmith). Therefore, even if the operator controls water level using the instrument with maximum error, the fuel would still be covered with water and would be adequately cooled. Speis et al., ff. Tr. 6357, at 30; Tr. 4856-57 (Robare).

7B:161. These errors in the water level measurement instrumentation are unlikely to delay ECCS actuation. Where flashing is the result of a small steamline break, there would be no delay in ECCS actuation. Where flashing is caused by a large break LOCA and subsequent depressurization by containment spray actuation, the ECCS would already have been actuated prior to containment spray; thus, there is no delay in ECCS actuation. Where flashing results from failure of drywell coolers, no ECCS actuation is necessary because there is no break. One can postulate the occurrence of a LOCA while the reactor is in the shutdown cooling mode of operation and while drywell temperature remains high. In such a situation there is a possibility for delayed ECCS actuation, but the staff's testimony that this is a very unlikely scenario was not controverted. Speis et al., ff. Tr. 6357, at 28-29.

7B:162. The reactor operator is trained to respond to a loss of water level indication and has specific emergency operating procedures at hand to respond to such a situation. Special consideration has been given in the EOP's to the importance of the water level in the reactor pressure vessel. Tr. 6911 (Rossi). The generic BWR emergency procedure guidelines include caution and action statements related to loss of level instrumentation. Suffolk County Attachment 5 (Attachment A at 9). Any time the operator cannot determine the water level, he is trained to depressurize and flood the vessel. Where loss of water level indication is due to flashing, of course, there will already have been some depressurization in the vessel. Tr. 7691-92 (Hodges). If the operator confronted with the conflicting indications perceived correctly that there was a malfunction in one leg of his instrument system, he would proceed to start RCIC and maintain water level with the reactor shut down. If he did not perceive the problem correctly, he would follow the emergency procedure. In either case, no fuel damage results. Tr. 6873 (Hodges).

7B:163. The steps to be taken by an operator to depressurize are set forth in the Shoreham emergency procedures. These steps involve more than one procedure, but the procedures are set up in a logical sequence. Also, the operator is quite familiar with these procedures through training. Tr. 6845 (Hodges). Flooding the vessel upon loss of water level indication involves several steps. Shoreham Procedure #29.023.01 states, at step 3-4, that if reactor pressure vessel water level cannot be maintained or determined, the operator should proceed to Procedure #29.023.04 on level restoration. The level control procedure is normally the first procedure the operator would enter following any abnormal situation. Tr. 6850 (Hodges). Step 3.3 of Procedure #29.023.04 gives a series of steps to be followed if water level cannot be determined. These steps involve starting up low pressure injection systems. Tr. 6851 (Hodges). It then refers the operator to Procedure #29.023.05 on rapid reactor pressure vessel depressurization. This procedure gives steps for vessel depressurization. Tr. 6851 (Hodges). The operator is then referred to Procedure #29.023.09 on reactor pressure vessel flooding. If water level in the pressure vessel cannot be determined, the operator is to commence injection into the pressure vessel with several systems until at least 3 safety relief valves open, thereby assuring that the vessel is full of water because water will be pouring out of the relief valves. Tr. 6851 (Hodges).

7B:164. These four procedures are grouped together physically and can be scanned quickly to find the instruction for the symptoms involved. Tr. 6852, 7805 (Hodges). According to Staff witness Hodges, identifying the procedures and taking appropriate steps could be accomplished in less than five minutes. Tr. 7806 (Hodges).

7B:165. After the submission of Intervenors' prefiled testimony on this contention, Intervenors obtained through a Freedom of Information Act request a copy of an internal NRC staff memorandum on the subject of a "Safety Concern Associated With Reactor Vessel Level Instrumentation In Boiling Water Reactors."[5/] The memorandum ("Michelson memorandum" or "Suffolk County Ex. 1") raises a concern that a break in the WLI reference leg would cause an interaction between plant control systems and protection systems which might adversely affect the ability of the protective system channels to perform their function. Suffolk County Ex. 1, at 1; Tr. 6855 (Hodges, Rossi).

7B:166. Applicant's testimony asserted that General Electric had studied this situation and concluded that the accident is bounded by design basis accidents already analyzed in Chapter 15 of the FSAR. Applicant also noted that the recipient of the Michelson memorandum, Harold Denton, had

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5/ This January 20, 1982 memorandum was from Carlyle Michelson, Director of the Office for Analysis and Evaluation of Operational Data, to Harold R. Denton, Director, Office of Nuclear Reactor Regulation. Enclosed with this memorandum is a study prepared by Mr. Michelson's office. The January 20, 1982 memorandum and enclosed study were received in evidence as Suffolk County Exhibit 1.

responded in a memorandum to Michelson that "the unaffected protective channels are sufficient to provide all protective functions" and that no immediate licensing action was required. Burns et al., ff. Tr. 4346, at 157-58, n. 39; LILCo Ex. 13, ff. Tr. 5496.

7B:167. The Staff's testimony states that, in the event of a break at Shoreham such as that hypothesized in the Michelson memorandum, there is sufficient redundancy in the WLI to prevent a sensing line malfunction and another random electrical failure from defeating actuation of emergency core cooling. Manual Action, however, would be required. Manual actuation within ten minutes following reactor trip would maintain the water level above the top of the active fuel. Speis et al., ff. Tr. 6357, at 31. This was the conclusion of a Shoreham-specific review conducted by the Staff after issuance of the Michelson memorandum. Tr. 6863 (Hodges).

7B:168. This Shoreham-specific analysis was not performed immediately upon publication of the Michelson memorandum because the problem was not considered unique to Shoreham. In addition, the Staff does not consider this type of event to be extremely significant from a safety standpoint since a reactor trip results and time is available for the operator to act. Tr. 6866 (Hodges, Rossi). Specifically, calculations by General Electric and by the Staff's consultants at Brookhaven show that it would take approximately 15 minutes to uncover the fuel in the case of an event of the type postulated in the Michelson memorandum. Roughly 30 more minutes would pass before temperatures above 2200° were reached.

Tr. 6916-17 (Hodges). General Electric has evaluated the Michelson scenario for Shoreham and determined that the protective systems are adequately designed to preclude this from being a safety concern. No fuel failure results. Tr. 4847-49 (Robare).

7B:169. Questioning by the Board focused on the issue whether the redundancy requirements of GDC 24 were met after an event of the type discussed in the Michelson memorandum. Tr. 6886-97. The Michelson memorandum itself questions whether selected BWR level instrumentation systems "meet the intent of the regulations for operation of protection and control systems single failure criterion as delineated in General Design Criterion 24." Suffolk County Ex. 1, at 19.

7B:170. Staff witness Rossi agreed that failure in a sensing line would eliminate the redundancy for some types of failures in the automatic actuation of the emergency core cooling system. However, without an additional single failure, automatic initiation of core cooling would still be operable. Tr. 6874-75 (Rossi). Staff practice has not been to preclude a failure in a sensing line from leaving a system which has no further redundancy; after the sensing line failure, the remaining portion of the protection system need not necessarily still meet the single failure criterion. Tr. 6889-90 (Rossi). LILCo meets GDC-24 based on the Staff's practice in interpreting it. Tr. 6895 (Rossi). The Staff considers GDC-24 to be satisfied because manual action can be taken quickly enough to actuate emergency core cooling. This is a judgment based on an examination of the relevant circumstances. Tr. 6890-91 (Rossi). The Board finds that this position is not arbitrary.

7B:171. The Board finds that the various interactions cited by Intevenors affecting the WLI do not demonstrate an inadequacy in the methodology applied by the Applicant in the evaluation of potential adverse systems interactions.

7B:172. The problem of flashing and boil-off in the WLI reference leg has been known for many years and the plant has been designed to protect against such an event through large drywell coolers, drywell temperature monitors and technical specification requirements (Burns et al., ff. Tr. 4346, at 153; Tr. 5558 (Robare) and emergency operating procedures in the event of loss of water level indication. Tr. 6911 (Rossi). The results of a break in WLI reference leg are within the Chapter 15 analyses and the Shoreham design provides adequate protection against such a failure. Burns et al., ff. Tr. 4346, at 157-58; Tr. 4847-49 (Robare). Applicant has shown that Shoreham's design provides reasonable assurance of no undue risk to the public health and safety from adverse systems interactions at Shoreham.

3. Unresolved Safety Issues Concerning Systems Interactions

7B:173. Unresolved safety issue A-17 is entitled simply "Systems Interactions in Nuclear Power Plants." Unresolved safety issue A-47, "Safety Implication of Control Systems," is considered by the Staff to be a specific subset of the systems interaction problem which deserves special consideration. Tr. 6485 (Conran).

a. A-17 ("Systems Interactions")

7B:174. The general concern involved in the systems interactions issue is the possibility of one reactor plant system acting on one or more other systems in a way not consciously intended by design so as to adversely affect the safety of the plant. In designing reactor plant systems, therefore, a primary objective has been to incorporate design features (e.g., redundancy and diversity in systems that perform required safety functions, and independence of safety systems from all other plant systems and from each other) such that, ideally, several independent system failures must occur to degrade unacceptably or to cause total failure of any necessary safety function. The specific objective of a systems interaction analysis is to provide assurance that the independent functioning of safety systems is not jeopardized by preconditions within the plant design (particularly dependencies hidden in supporting and interfacing systems) that cause faults to be dependent. Speis et al., ff. Tr. 6357, at 34-35.

7B:175. Some events have occurred in the past at operating plants that have adversely affected safety system redundancy, and the functioning of safety systems have actually been degraded in other events (e.g., the Browns Ferry partial failure-to-scram). The frequency and possible implications of such events has prompted the staff to consider whether additional system interaction analysis requirements should be developed and imposed in order to examine more fully than currently required the question of susceptibility of reactor plant systems to potential systems interactions. A program has been initiated to address these questions and has progressed significantly over the past few years. Speis et al., ff. Tr. 6357, at 36.*

7B:176. The purpose of Task A-17 is "to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems." Staff Ex. 2A at B-10.

7B:177. The Staff's program for studying the systems interaction issue as outlined above was initiated in May 1978 with the definition of USI A-17, "Systems Interaction in Nuclear Power Plants." The early phase of this program involved development of a candidate systems interaction methodology by Sandia Laboratory, and a limited-scope trial application of that methodology to the Watts Bar I facility. The objective of this effort was to attempt to evaluate both the methodology developed and (by comparison) the adequacy of existing Standard Review Plan procedures for uncovering potential systems interactions. Speis et al., ff. Tr. 6357, at 37; Staff Ex. 2A, at B-10.

7B:178. This Phase I analysis was performed by fault trees to identify component failure combinations (cut-sets). The total number of possible independent failure combinations that could have been analyzed was reduced by introducing six linking features into the analysis. This effort identified a few potentially adverse interactions within the limited scope of the study. The staff reviewed the interactions identified for safety significance and generic implications. The staff concluded that no corrective measures were needed immediately at Watts Bar I, except with regard to the potential for interaction between the power operated relief valve and its associated block valve. This interaction had been separately identified by analyses of the TMI-2 accident and corrective measures were already being implemented. This initial A-17 effort was deemed unsuccessful. Speis et al., ff. Tr. 6357, at 37-38; Staff Ex. 2A, at B-10.

7B:179. In May 1980, in the aftermath of the TMI-2 accident, the TMI-2 Action Plan (NUREG-0660) was approved by the Commission. Item II.C.3 of the Action Plan (Systems Interaction) incorporated the USI A-17 effort and broadened the systems interaction program. Special limited-scope (spatially coupled, seismic initiator) system interaction analyses were performed at Diablo Canyon Units 1 and 2 and at San Onofre Unit 2. The basic method used in both analyses was in situ visual examination of plant systems for potential failures of "sources" (i.e., non-seismic Category I piping/equipment) that could adversely affect the functioning of safety-related "targets". The Staff and ACRS accepted both analyses even though the results differed significantly in terms of the number of potentially adverse systems interactions discovered. The differences in results obtained were explainable in view of differences in design criteria applied at the two facilities. The San Onofre unit design criteria required both nonsafety-related and safety-related systems to be mounted with Seismic I qualified mountings. This design criteria had not been applied at Diablo Canyon. Speis _et al_., ff. Tr. 6357, at 38; Tr. 7150 (Conran).

7B:180. In January 1981, a staff assessment (based on surveys by three national laboratories under contract to the staff) of then available methodologies led to the conclusion that application of any single method could not identify all potentially important systems interactions. Therefore, the staff undertook a program to further develop available methods (or combinations of available methods) and to incorporate them into what has been termed "Interim Guidance" that could be used by licensees/applicants for a comprehensive, systematic systems interaction evaluation of specific facilities. The Interim Guidance was intended to describe an acceptable _general_ approach to a comprehensive systems interaction analysis effort, and to provide at least two distinct alternative detailed step-by-step illustrative procedures for accomplishing that objective. The documentation of one illustrative procedure (characterized as a Fault Tree/Interactive FMEA methodology) is essentially complete and ready for trial application at this point. Documentation of the second illustrative procedure (called the Matrix-based Digraph Method) was scheduled to be completed by August 1982. The Interim Guidance is based upon experience gained during the Watts Bar limited-scope analysis, the Diablo Canyon and San Onofre seismic-initiator systems interaction reviews, the surveys conducted by the national laboratories, and review of the Indian Point-3 program plan. Speis _et al_., ff. Tr. 6357, at 38-39.

7B:181. Another major element in the expanded systems interaction program included under Action Plan Item II.C.3 is the broad-scope systems interaction evaluation of the Indian Point-3 facility by the Power Authority of the State of New York (PASNY), employing a methodology developed by themselves. PASNY submitted a preliminary plan for this systems interaction study in March 1981, and staff review was completed six months later. PASNY's final program plan, incorporating staff's review comments, was received in January 1982 and has been approved/endorsed for performance at IP-3 by both the staff and the ACRS. Th actual study effort got underway in April 1982 and is progressing satisfactorily at this time. It is estimated that approximately one year will be required to complete this study. Speis et al., ff. Tr. 6357, at 39-40.

7B:182. One remaining major element in the staff's system interaction program plan under Action Plan Item II.C.3, which has not yet been approved or initiated by the NRC staff, is the so-called "Pilot Program" effort. As initially conceived, the Interim Guidance described in the proceeding was to be applied on a trial basis in several plants undergoing operating license review, and results from both the pilot plant analyses and the IP-3 study were to be evaluated in reaching a final decision regarding the need for additional requirements to perform expanded scope system interaction analyses on some or all LWRs. Speis et al., ff. Tr. 6357, at 40. *

7B:183. The staff has also given consideration to the option of requiring that systems interaction analyses be performed on the first group of

NREP/SEP Phase III[6/] plants using the PASNY methodology as the basis, after the staff has reviewed initial phases of the IP-3 study and identified any needed modifications. Subsequent NREP/SEP plants could perform systems interaction analyses using methodology that incorporates further improvements on the later pilot studies. Thus, the refinement of methodology, and the decision to proceed with each additional step, would depend on what had been learned to date. This option is consistent with the view that performance of systems interaction dependency analyses in combination with current PRAs will better assure that PRA results will provide adequate insight regarding the possible need for improvements in safety and reliability. Speis et al., ff. Tr. 6357, at 40-41; see Tr. 6463 (Thadani), Tr. 7139 (Conran).

7B:184. There has been slippage of more than a year from the schedule originally proposed by the Staff. Part of the reason for that slippage lies in the Indian Point PRA effort as well as operating problems there.

6/ The Systematic Evaluation Program (SEP) is an ongoing program involving a deterministic review of operating plants to assess the adequacy of the design and operation of existing reactors, to compare them with current safety criteria, and to provide the basis for integrated and balanced backfit decisions, if required. The program was initiated in 1977; Phase II of the program is now in progress. Phase III (SEP III) is scheduled to begin in FY 1983 for completion in FY 1989.

The National Reliability Evaluation Program (NREP) is a program proposed to assess design and operational deficiencies of all commerical operating power reactors employing probabilistic risk assessment (PRA) techniques. The staff is seeking Commission approval to coordinate NREP with SEP Phase III and require SEP III licensees to do PRA under NREP.

Another part of the reason is the difficulty in merging the NREP program and the systems interaction program. Tr. 7151 (Conran).

7B:185. Within the existing regulatory framework, the systems interaction concern is addressed by evaluating plant designs against well-established deterministic requirements and criteria embodied in existing regulatory guidance documents (e.g., Regulatory Guides and the Standard Review Plan). These current requirements are founded on the principle of "defense-in-depth," and they include provisions for design features such as physical separation and functional independence of redundant safety systems, as well as other measures that provide protection against hazards such as pipe ruptures, missiles, seismic events, fires, and flooding. Also, the quality assurance program that is applied during the design, construction, and operational phases for each plant provides additional assurance in this regard by helping to prevent inadvertent introduction of adverse systems interactions contrary to approved design. Speis et al., ff. Tr. 6357, at 35; Staff Ex. 2A, at B-9, B-10.

7B:186. The deterministic review process includes a certain review of potential dependencies in a plant. Tr. 6659 (Thadani). The Staff requires several types of studies of systems interaction through its standard review plan. These include, for example, fire protection and flooding requirements. Tr. 6779 (Thadani).

7B:187. Mr. Conran testified that there has been no indication from any sector that the requirements which existed prior to TMI, supplemented by post-TMI changes, are not adequate. Tr. 7153 (Conran).

7B:188. In the Staff's view, completion of the generic program may provide the basis for making an orderly decision as to the possible need for additional systems interaction requirements. In the interim, however, the Staff believes that adequate reasonable assurance of public health and safety is provided by compliance with current requirements and procedures. Speis _et al_., ff. Tr. 6357, at 36-37*; Staff Ex. 2A, at B-9 through B-11; Tr. 7141 (Conran)*; Tr. 7642 (Thadani). This conclusion is recorded in the SER for Shoreham in the following words:

> "[S]tudies to date indicate that current review procedures and criteria supplemented by the application of of post-TMI findings and risk studies provide reasonable assurance that the effects of potential systems interaction on plant safety will be within the effects on plant safety previously evaluated." Staff Ex. 2A, at B-11; Speis _et al_., ff. Tr. 6357, at 41-42.

7B:189. The same conclusion was expressed earlier this year by the Staff in response to a recommendation of the ACRS that some additional systems interaction requirements be imposed immediately on licensee/applicants. In a February 12, 1982 letter from William J. Dircks, Executive Director for Operations to Paul Shewmon, Chairman of ACRS, Mr. Dircks wrote as follows:

> "NRR continues in the confidence that _current_ regulatory requirements and procedures provide an _adequate_ degree of public health and safety."

Speis _et al_., ff. Tr. 6357, at 36, 37, 42. The Dircks memorandum reaffirms to the ACRS the Staff's position that compliance with existing requirements provides reasonable assurance that potential adverse systems interactions present no undue risk to public health and safety. Tr. 6374-75 (Conran)*; Tr. 6779 (Thadani); Staff Ex. 2A, at B-11.

7B:190. Contrary to Intervenors' proposed finding 7B:288, the Staff concluded in its testimony that current regulatory requirements and procedures do provide reasonable assurance of no undue risk to public health and safety against adverse systems interactions. Speis et al., ff. Tr. 6357, at 35-37, 41-42*; Tr. 7642, 7643-44 (Thadini).

7B:191. Both Mr. Thadani and Mr. Conran agreed with the statement in the Dircks memo that additional systems interaction analysis requirements should not be imposed until the Staff has drawn a conclusion as to the efficacy of such analyses. Tr. 7509 (Conran*, Thadani).

b. A-47 ("Safety Implications of Control Systems")

7B:192. Unresolved safety issue A-47 concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. Failures or malfunctions may occur independently or as a result of an accident or transient. One concern is the potential for a single failure such as a loss of a power supply, sensor impulse line failure, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those transients analyzed as "anticipated operational occurrences". A second concern is for a postulated accident to cause control system failures which would make the accident more severe than presently analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or physically damaging the control equipment. Speis et al., ff. Tr. 6357, at 42; Tr. 7470-71 (Rossi); Staff Ex. 2A, at B-15.

7B-193. The purpose of the A-47 task is to examine the criteria and philosophy used by the Staff in the review of control systems to determine if they are sufficient and whether new criteria are appropriate. Tr. 7436-37 (Rossi); Staff Ex. 2A, at B-15. The final Task Action Plan for A-47 has not been approved. Tr. 7439 (Rossi). Additional systematic studies will be done as a part of the determination of whether new criteria are required. Tr. 7437 (Rossi). Some specific plants are to be used as examples to evaluate present criteria. Tr. 7438 (Rossi).

7B:194. In general, until approximately one year ago systematic evaluation of control systems designs had not been performed to determine whether single event induced multiple control system actions could result in a transient such that limits established for "anticipated operational occurrences" are exceeded. Single failures or events which could induce multiple control system actions would presumably include events such as a loss of power supply or failure of sensor impulse line. If single failure-or event-induced multiple control system actions do indeed exist, experience with operating plants indicates that incidents resulting in transients more severe than currently analyzed as "anticipated operational occurrences" have a low probability. Speis et al., ff. Tr. 6357, at 43-44.

7B:195. Until approximately two and one half years ago systematic evaluations of control system designs had not been performed to determine whether postulated accidents could cause control system failures resulting in control actions which would make accident consequences more severe than presently analyzed. Licensees have, however, now reviewed the possibility of consequential control system failures which exacerbate the effects of some high energy line breaks and have taken action where needed, to assure that the postulated events would be adequately mitigated. Speis et al., ff. Tr. 6357, at 44.

7B:196. In accordance with Standard Review Plan Chapter 7, NRC staff reviews have been performed on currently licensed plants as well as on Shoreham with the goal of assuring that control system failures will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident". The approach has been either to provide independence between safety-related and nonsafety-related systems or to require isolating devices such as isolation amplifiers between safety-related and nonsafety-related systems such that failures of nonsafety-related equipment cannot propagate through the isolating devices to impair operation of safety-related equipment. Speis et al., ff. Tr. 6357, at 42-43.

7B:197. A specific set of "anticipated operational occurrences" and "accidents" has been conservatively analyzed to demonstrate that plant trip and/or safety system equipment actuation occurs with sufficient capability and on a time scale such that the consequences are within specified acceptable limits. The analyses are intended to be sufficiently conservative to verify that the potential consequences to the health and safety of the public are within acceptable limits for a wide range of postulated events even though specific actual events might not follow the same assumptions made in the analyses. Speis et al., ff. Tr. 6357, at 42-43.

7B:198. The resolution of Unresolved Safety Issue A-47 will systematically determine if current licensing practices with respect to control systems are adequate. Should the resolution of A-47 indicate that additional criteria for control system designs are necessary or that specific problems require resolution, appropriate action will be taken for plants in the licensing process as well as for plants now in operation. At this time, the staff knows of no specific control system failures or actions on Shoreham or any other plant which would lead to undue risk to the health and safety of the public. Speis et al., ff. Tr. 6357, at 44-45. Staff witness Rossi, one of the NRC reviewers involved in the program concerning A-47, could not recall a single instance in which applicable limits had been exceeded. Tr. 6504, 7455-56 (Rossi).

7B:199. There are two pending questions for Shoreham relating to A-47. The first relates to the effect of power supply, sensor and sensor impulse line failures on several control systems at the same time. The second deals with a Shoreham-specific evaluation of the effect of high-energy line breaks on control systems. Tr. 7440 (Rossi).

7B:200. The staff has requested that the applicant identify any power sources, sensors, or sensor impulse lines which provide power or signals to two or more control systems and demonstrate that failures of these power sources, sensors, or sensor impulse lines will not result in consequences more severe than those bounded by the analyses of "anticipated operational occurrences" in Chapter 15 of the FSAR. Speis et al., ff. Tr. 6357, at 45.

7B:201. The staff has also requested that the applicant perform a review to demonstrate that the harsh environments associated with high energy line breaks will not cause control system malfunctions resulting in consequences more severe than those of the Chapter 15 accident analyses. Upon completion of these efforts by the applicant to the satisfaction of the staff, the staff will be able to conclude, with reasonable assurance, that control system failures do not represent an undue risk to the health and safety of the public. The Applicant will, however, be required to address any additional staff guidance which may result from the resolution of Unresolved Safety Issues A-47 and A-17. Speis et al., ff. Tr. 6357, at 45; Tr. 7444 (Rossi).

7B:202. The Board finds that the Staff has satisfied its obligations under North Anna with respect to both Unresolved Safety Issues A-17 and A-47.

E. ALTERNATIVE METHODOLOGIES PROPOSED BY INTERVENORS

1. Regulatory Status of the Alternative Methodologies Cited by Intervenors

7B:203. Intervenors maintain that the methodology embodied in the design basis analysis is deficient with respect to the identification of potential systems interactions and the classification of plant structures, systems and components. Minor et al., ff. Tr. 1113, at 60. They allege that alternative methods exist which would supplement and improve the existing design basis analysis approach. Minor et al., ff. Tr. 1113, at 63. Specifically, Intervenors argue that probabilistic risk assessment, various types of dependency analysis, and a review of emergency operating procedures must be applied in order to demonstrate compliance with the regulations. We have previously discussed the adequacy of the present methodology in the classification of structures, systems and components and the analysis of systems interactions. We conclude for the reasons there given and for the reasons discussed below that these alternative methodologies need not be applied as a predicate for licensing Shoreham.

7B:204. PRA is an analytical technique which quantifies the probabilities and consequences associated with accidents and malfunctions by applying probabilistic and statistical techniques to an evaluation of plant reliability and safety. Burns et al., ff. Tr. 4346, at 66. By using PRA, a safety assessor attempts to set into better perspective the contributors to various accident sequences and risk and thereby identify the need for additional safety features, if any, improved equipment reliability and, where necessary, areas of research and testing. Burns et al., ff. Tr. 4346, at 67.

7B:205. The NRC's use of PRA in the regulatory process is in a state of development. In the case of certain construction permit applicants, a site-specific PRA is required by 10 CFR § 50.34(f)(1)(i). In addition, the Staff has requested site-specific PRA's for certain applications (e.g., Limerick) for operating licenses. No such request has been made by the Staff for Shoreham, and LILCo has gone beyond current regulatory requirements in contracting for a plant-specific PRA. Tr. 6621, 6464-65, 6778, 7667-68 (Thadani).

7B:206. The Staff believes, and the Board concurs, that the Staff's deterministic requirements[7/] provide an adequate licensing basis and a sufficient means of identifying dependencies and classifying plant structures, systems and components. For the present and near future, PRA's are considered an adjunct or useful supplement to those current deterministic requirements. Tr. 6594, 6460, 6464, 6774 (Thadani); Tr. 6764 (Conran). If Shoreham satisfies the deterministic criteria, there is an adequate degree of assurance of no undue risk to public health and safety. Tr. 6780 (Thadani).

7/ Dr. Speis defines "deterministic" as the use of a system based upon set criteria rather than probabilistic goals. Tr. 6496 (Speis).

7B:207. Methodologies such as PRA, failure modes and effects analysis, systems interaction analyses or dependency analyses are not required by regulations or staff practice in the safety classification of structures, systems and components. These techniques have been used in some cases to look for weak points in plant systems designs or to evaluate the risk of particular event sequences. They have been used to identify failure modes and the need for equipment changes, increased surveillance, additional testing, and improved procedures to reduce the risk of particular event sequences. Speis et al., ff. Tr. 6357, at 31-32.

7B:208. One important distinction between existing deterministic criteria and probabilistic analysis is that the Staff's deterministic review applies conservative, very restrictive assumptions to a model which is itself conservative, while probabilistic analysis attempts to utilize realistic assumptions without the addition of various ISMS conservatisms. Tr. 6497-99 (Thadani). The Staff's use of deterministic criteria is intentionally conservative rather than realistic. Tr. 6497 (Speis).

7B:209. The NRC's review of a PRA is totally separate from the hearing requirements or NRC regulations. Tr. 6725-26 (Thadani). The Staff's confidence in the safety of plants without PRA's derives from the amount of effort that goes into the design of a plant, the documentation of that design, the resources expended in review and the flow of information from applicant to the Staff. Tr. 6788-89 (Thadani). The bases for operating a plant are not grounded in PRA. Rather, they are embodied in the General Design Criteria and specified in the Standard Review Plan, regulatory guides, and other guidance documents. Tr. 6659 (Thadani).

7B:210. LILCo agreed that the PRA is not necessary to the licensing of Shoreham. In LILCo's view, more information about its plant is always better than less and the principal benefit of PRA is that it adds to one's understanding of the plant. PRA provides LILCo with a diverse method of reviewing the results of the deterministic process. Tr. 5981, 6149 (Burns). LILCo intends to use the Shoreham PRA, in part, as basic data for a utility risk management program. Burns, et al., ff. Tr. 4346, at 87; Tr. 5636 (Burns, Joksimovich); Tr. 5964-65 (Joksimovich).

7B:211. Dr. Burns was unable to state with certainty whether the Shoreham PRA looked at more systems interactions than the various deterministic standards had. Tr. 5983 (Burns).

7B:212. Intervenors have highlighted particular types of systems interaction analysis, such as failure modes and effects analysis, walkdowns, and dependency analysis, and have argued that such analyses must be applied on a plant-wide basis for the identification of system interaction and the classification of plant structures, systems and components. Minor et al., ff. Tr. 1113, at 63-68. No specific regulatory requirement exists, however, for a plant-wide application of any of these analytical methods. Tr. 1479 (Goldsmith). Neither is there any specific requirement in the regulations or in Staff practice to apply these methods or a review of emergency procedures in the safety classification of structures, systems and components. Speis et al., ff. Tr. 6357, at 31-32.

7B:213. Staff witnesses testified that there is not at the present time a systematic methodology for using PRA (or the other methodologies cited by Intervenors) for the purpose of classification or ranking of plant items. Speis et al., ff. Tr. 6357, at 32-34; Tr. 6570-73 (Rossi); Tr. 6684, 7616 (Thadani); Tr. 6700-02 (Rossi, Thadani).[8/] The absence of reasonably well understood methods and procedures would result in different results from different studies caused by the different assumptions utilized. Different lists of structures, systems and components would result. Tr. 6702-03 (Thadani).

7B:214. There is no basis for concluding that it is likely that a PRA would require a change in the classification of any system from important to safety to safety-related. Staff witness Thadani, who was familiar with several PRA's, could think of no example where PRA analysis

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8/ Since the 1970's, the IEEE has considered the need for additional safety classes of electrical equipment and methodologies which could be used to determine a "level of importance to safety" for nuclear power plant instrumentation and control systems. To date, the IEEE's efforts (including the development of a draft standard, IEEE P827) have not been successful in producing a methodology acceptable on a consensus bases to the IEEE. Speis et al., ff. Tr. 6357, at 32.

would have resulted in reclassification of a structure, system or component. Tr. 6643-44 (Thadani). This, together with the lack of a consistent methodology, is the reason the Staff is not recommending the use of PRA for classification of structures, systems and components. Tr. 6641-44, 7603-04 (Thadani).

2. Reliance on the Shoreham Draft PRA

a. Applicant's testimonial use of the Shoreham draft PRA

7B:215. Applicant's witness Dr. Joksimovich expressed his opinion that "the Shoreham PRA approach provides a meaningful and efficient, if not the only, framework for examining "the systems interaction issue". Burns *et al*., ff. Tr. 4346, at 81. He went on to describe the Shoreham PRA as the "best means for addressing the issue." *Id*. Dr. Edward T. Burns, SAI's principal analyst for the Shoreham PRA, described the methodology utilized and its application in the Shoreham PRA. Dr. Burns agreed with Dr. Joksimovich on the efficacy of PRA for systems interaction analysis:

> "SAI judges that fault tree/event tree methodology is the best available technique for augmenting the existing deterministic evaluations and NRC regulations to ensure that systems interactions are exposed and potential areas of concern are identified."

Burns *et al*., ff. Tr. 4346, at 97.

7B:216. LILCO's witnesses also expressed their conclusion that the Shoreham PRA confirms the adequacy of the treatment of systems interactions at Shoreham. Tr. 5897, 6159 (Kascsak); Tr. 5940 (Joksimovich, Burns); Tr. 5823 (Joksimovich). While this Board struck several such conclusions in the prefiled testimony at Intervenor's motion on the grounds that the conclusions (as opposed to the methodology) of the

Shoreham PRA were beyond the scope of this contention, similar conclusions were elicited upon the record by Intervenors' own cross-examination. See, e.g., Tr. 5897 (Kascsak).

b. Staff's plans with respect to the Shoreham PRA

7B:217. The Staff emphasized repeatedly that it had not required the performance and submission of a PRA for Shoreham as part of the regulatory review process for issuing an operating license to LILCo (Speis et al. ff. Tr. 6357, at 33) and that LILCo had gone beyond regulatory requirements in conducting such a study. Tr. 6778, 6464-65, 7667-68 (Thadani). There were no communications between the Staff and LILCo about doing a PRA for Shoreham. Tr. 6108 (Kascsak).

7B:218. The Staff has no "specific criteria for evaluating such an assessment for Shoreham." Speis et al., ff. Tr. 6357, at 33; Tr. 6457, 6649 (Thadani). Mr. Thadani explained that the Staff has not yet developed an audit guide for the review of PRA's, (Tr. 6693 (Thadani)), and that without such a model for evaluation there can be no confidence in the reproduceability of results obtained. Tr. 6591 (Thadani). A benchmark is needed against which the results of PRA's can be compared in terms of the acceptability of the numerical risk factors derived. Tr. 6692 (Thadani).

7B:219. The Staff is working toward developing an implementation plan for the Commission's proposed safety goals. Until the Commission promulgates specific criteria against which to compare PRA's, the Staff's approach is to learn from these studies whether there are areas which the Staff should be pursuing further. Tr. 6456 (Thadani). Judgments that are made depend on considerations other than just the numerical estimates. Tr. 6692 (Thadani).

7B:220. Despite these problems, the Staff will require submittal of the final Shoreham PRA and will review it to gain added insight into potential safety improvements. Speis et al., ff. Tr. 6357, at 33; Tr. 6456, 6458, 6644-53; 7647-53 (Thadani). If the NREP program goes forward, the Shoreham PRA will be reviewed within that program. Tr. 6455 (Thadani). However, the Staff will review the Shoreham PRA regardless of what happens with the NREP program. Tr. 6652-53 (Thadani).

7B:221. With respect to the schedule for the Staff's review of the Shoreham PRA, it is expected that the review effort would take approximately one year from the time the final Shoreham PRA is submitted. Speis et al., ff. Tr. 6357, at 33; Tr. 6458, Tr. 6645 (Thadani). The Staff cannot afford to expend its limited resources on the review of draft PRA's because they generally change "radically" as time goes on and it is expected that the Shoreham PRA to undergo substantial changes as a result of mistakes, omissions or new understandings before it becomes final. Tr. 6457, 6774, (Thadani). Staff review of the draft Shoreham PRA "would not be very helpful," (Tr. 6584 (Thadani)), because of the possibility that conclusions might be undercut by subsequent changes in the PRA results. Tr. 6458, 6595 (Thadani). Mr. Thadani also described the various tasks on which his branch was devoting its efforts and described these tasks as "more pressing." Tr. 6650 (Thadani). These activities are focused mostly on actions mandated by the Commission, including Indian Point, Limerick PRA, Zion PRA, Big Rock PRA, Clinch River, SEP, pressurized thermal shock, NREP and construction permit applications. In the Staff's view, the resources are not available to take on additional tasks. Tr. 6619-21 (Thadani).

7B:222. Even a quick review for treatment of dependencies would take 3 to 6 months in order to develop supportable views, assuming the availability of resources which the Staff does not believe are presently available. Tr. 6619, 6630, 6645 (Thadani). More specifically, this estimate was based on the availability of high quality documentation, of experienced reviewers, and of utility cooperation in the interaction that would be required. Tr. 6638-39 (Thadani). Interaction with the utility is an "extremely critical" and time consuming part of the review process. Tr. 6458-59 (Thadani).

7B:223. To properly examine PRA one must look at the methods, the treatment of initiators and their relation to mitigating systems, whether control systems are analyzed, what fault trees were done and to what depth, whether and how spatial and environmental effects were considered, the treatment of human coupling and the depth and extent of walkdowns. Tr. 6628-29 (Thadani). Eight to twelve man-months of effort would be required. Tr. 6639 (Thadani). Looking at a PRA to evaluate the appropriation of classification of items would take even a greater effort than it would to look at systems interactions. Tr. 6622 (Thadani).

c. PRA and the identification of dependencies

7B:224. The Staff does not at present have a position on the preferability of event tree/fault tree methodology as against other methodologies for the identification of intersystem dependencies. The Staff believes that it is premature at this time to draw any conclusion in this regard, as the Staff is pursuing a program to identify the best, most effective technique. Tr. 6747, 6749, 7536 (Conran, Thadani). Under the Staff's program, another year or two of development and testing of techniques should permit identification of the most effective methods and the depth of analysis required to ensure that important dependencies have not been missed. T [illegible] -28 (Thadani). The purpose of the pilot plant approach to systems interaction analysis requirements is to consider promising candidate methodology, to observe and compare results, and to see if the effort is worthwhile and if any one method is clearly preferable over others. The Staff is not in a position to draw those conclusions yet. Tr. 7508 (Conran).

7B:225. Attachment 1 to the Staff's prefiled testimony included a memorandum authored by Staff witness Conran discussing certain studies at Indian Point relating to systems interactions. That memorandum expressed Mr. Conran's view that systems interaction analysis "is a useful exercise and has inherent value completely aside and apart from PRA." Mr. Thadani explicitly agreed. Tr. 6763 (Conran); Tr. 6766 (Thadani). The memorandum also states that the use of PRA methodology for systems interaction analysis purposes has "not yet been satisfactorily demonstrated . . . in applications attempted to date." Speis et al., ff. Tr. 6357, Attachment 1.

7B:226. Many methods can be used to search for systems interactions. PRA can identify dependencies. The difficulty is not in the use of event tree/fault tree methodology, but in how far these methods are carried: are the fault trees simplified or are they detailed down to the component level? An enormous amount of effort is required to do detailed fault trees on a large number of systems. Tr. 6619, 6465-66 (Thadani); see also Tr. 5645 (Burns); Tr. 5964-65 (Joksimovich).

7B:227. PRA has certain limitations at present. Limitations exist in the data base for probabilistic estimates. Tr. 6460, 7638-41 (Thadani); see e.g., Tr. 5294-95 (Ianni) (weakness of data base cited in context of probabilistic assessment of Browns Ferry partial failure to scram event). Quantification of factors such as sabotage may be impossible. Tr. 5658 (Burns). Design errors may go unidentified. Potential dependencies may exist by design, by oversight or by operational considerations. Tr. 6461, 7537-38 (Thadani). Large areas of uncertainty must also be recognized. Tr. 6457 (Thadani). For example, probabilistic treatment of external events such as earthquake, flood, external fires and high wind displays large uncertainties. Tr. 7607 (Thadani); Tr. 6218 (Joksimovich).

7B:228. Exclusion of external events, such as seismic initiators, is a limitation of the Shoreham PRA which would severely limit its utility for classification purposes. Tr. 6622 (Thadani).

7B:229. These shortcomings were evidenced in a system interaction study of Watts Bar using fault tree methodology. The Watts Bar study was a limited application of the fault tree method to plant systems performing basic safety functions (i.e., achieving and maintaining safe shutdown, core heat removal, and maintaining the integrity of the reactor core coolant boundary). Tr. 7574-75 (Conran). Among the problems with the Watts Bar results were that certain events from operating experience would not have been identified and highlighted by the methodology and that the fault tree methodology was too unweildy to be applied to a scope of study much larger than was done. Tr. 7573, 7575 (Conran).

7B:230. Mr. Thadani described an "ideal approach" to the use of PRA to attempt to identify important dependencies. First, both functional and systematic event trees would be developed. Fault trees would be developed "to at least the component level." Environmental effects, such as dust, temperature, ice and steam would be included. Fault trees would be developed for non-safety-related as well as safety-related systems. Dependency tables and diagrams would be generated not just for front line systems but for front line support system connections as well. The degree and depth of walkdowns in considering spatial interactions is critical. The role of the operator, who forms an important coupling for some potential unforeseen interactions, would be examined carefully.

Initiators would be examined in terms of their causes as well as effects, and the possibility of the same cause also being responsible for other effects would be evaluated. Interactive failure modes and effects analysis would be a useful part of the analysis, as would digraph-based analytical techniques. Such an ideal approach might be prohibitive in terms of cost and resource allocation. Tr. 6625-27 (Thadani).

7B:231. The critical point is that the Staff cannot say today how much analysis is enough to ensure adequate identification of dependencies. Tr. 6627 (Thadani). Dependencies are the hardest parts of a probabilistic analysis to identify and quantify. Tr. 6624-25 (Thadani). No single PRA to date has used all of the approaches which Mr. Thadani described as the ideal situation. Tr. 6782 (Thadani).

d. <u>Conclusion on reliance on the Shoreham draft PRA</u>

7B:232. The Board finds that it cannot rely on the Shoreham draft PRA for firm conclusions as to the identification of intersystem dependencies. First, it is a draft document still undergoing peer review. Changes may be made which would invalidate particular conclusions this Board might draw at present. Second, the Board does not have the benefit of the Staff's review of the document. Third, the Shoreham draft PRA excludes external events, for which large uncertainties exist. Finally, the cautions raised by the Staff in its explanation of its position on whether PRA is, as LILCo argues, the "best method" of identifying dependencies cause us to hesitate to embrace LILCo's position at the present time.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322
	)	(OL)
(Shoreham Nuclear Power Station,	)	
Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of NRC STAFF'S PROPOSED OPINION, FINDINGS OF FACT, AND CONCLUSIONS OF LAW IN THE FORM OF A PARTIAL INITIAL DECISION and NRC STAFF'S COMMENTS ON APPLICANTS' "BACKGROUND OF PROCEEDINGS" STATEMENT in the above-captioned proceeding have been served on the following: one asterisk indicates by express mail on February 11, 1983; two asterisks indicates by hand on February 11, 1983; three asterisks indicates by first class United States mail on February 14, 1983; the rest were either deposited in first class United States mail or the Nuclear Regulatory Commission's internal mail system, this 16th day of February, 1983:

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