

UNITED STATES OF AMERICA  
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of )  
LONG ISLAND LIGHTING COMPANY )  
(Shoreham Nuclear Power Station, )  
Unit 1) )

Docket No. 50-322 O.L.

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

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

February 11, 1983

May 16, 1983 (Revised)

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

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Explanatory note: The Staff has adopted the conventions used by LILCo in its revised findings to identify for the Board those portions of the Staff's proposed findings which have been added, deleted or modified in the revised findings. If a finding or a paragraph in the opinion consists of material which is totally new, an asterisk appears at the beginning of that paragraph or finding. If a finding or a paragraph in the opinion is modified, two asterisks appear at the beginning of that paragraph or finding. In addition, any new material is underlined and a line is drawn through any material which has been deleted.

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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") proffered 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.<sup>1/</sup> 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, Intervenor's 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, 1982. Intervenor, 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).

\*On February 24, 1983, the Board ordered that the record be reopened after one of the Staff's witnesses sought to modify certain of the testimony he had given. (Finding 7B:6A, 6B). Additional hearing sessions were held April 5-8, 1983. A total of fourteen witnesses were heard during this reopened phase of the hearings on these contentions. (Finding 7B:6C).

\*Proposed findings of fact and conclusions of law have been submitted by the Applicant, the County and the Staff.

#### 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 Division Manager at LILCO. Mr. Kascsak's 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 Intervenor's 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 Intervenor's 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;<sup>2/</sup> much of the review for Shoreham was completed under the supervision of Dr. Speis. Walter P. Haass was, at

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<sup>2/</sup> 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,<sup>3/</sup> 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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<sup>3/</sup> 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<sup>9</sup>); 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 7E:6) and that testimony was explored at length in cross-examination by the parties and by questioning from the Board.

\*After the close of the record on Contention 7B, on January 25, 1983, Staff counsel informed the Board and parties by letter that one of the Staff's witnesses who had testified in the proceeding on Contention 7B, James H. Conran, sought to modify certain of his testimony since he could no longer support some aspects of the testimony previously given by him. Mr. Conran prepared a written statement of his present views which was provided to the Board and parties on February 8.<sup>4/</sup> The Board then directed that the parties file statements of their views on the Conran submittal, particularly as to the need for the reopening of the record for receipt of the Conran submittal and for additional testimony by any party. (Finding 7B:6A).

\*Both the Staff and the County favored reopening the record; LILCo opposed such a step. After considering the arguments of the parties, the Board decided on February 24 that the record on Contention 7B should be reopened to receive Mr. Conran's statement in evidence and also to hear such testimony as was necessary in light of Mr. Conran's new testimony. The Board established a schedule for the filing of additional prefiled testimony and for further hearings. Supplemental Staff testimony was

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<sup>4/</sup> The February 8, 1983 statement was provided in unexecuted affidavit form. On February 9, 1983, an executed version of the affidavit was distributed by the Staff.

filed on March 10 by a panel consisting of the following: Roger J. Mattson, Director of the Division of Systems Integration; Richard H. Vollmer, Director of the Division of Engineering; Charles E. Rossi, a Section Leader in the Instrumentation and Control Systems Branch and a previous witness on this contention; Ashok C. Thadani, Branch Chief of the Reliability and Risk Assessment Branch and also a prior witness on this contention; and Franklin D. Coffman, Jr., Section Leader in the Systems Interaction Section of the Reliability and Risk Assessment Branch. The County filed supplemental testimony on March 25 by a panel consisting of Messrs. Goldsmith, Minor and Hubbard, all of whom had testified previously. LILCo decided against offering additional testimony. (Finding 7B:6B).

\*Additional hearing sessions were held on April 5-8, 1983.

Mr. Conran's submittal and the prefiled supplemental testimony of the Staff and the County were received in evidence and cross-examination and Board questioning were conducted. On the afternoon of April 7, after having heard the oral testimony of Mr. Conran and the staff witnesses, the Board asked the Applicant to provide additional oral testimony on certain aspects of the controversy. On April 8, additional testimony was given by a LILCo panel consisting of the following: Millard S. Pollock, Vice-President - Nuclear LILCo; James Rivello, Shoreham Plant Manager of LILCo; William J. Museler, LILCo's Director, Office of Nuclear; George F. Dawe, Supervisor of Project Licensing for Stone & Webster Engineering Corporation; and Brian McCaffrey, LILCO's Manager of Nuclear Compliance and Safety. Mr. Dawe had testified previously on Contention 7B; Messrs. Museler and McCaffrey had appeared as witnesses on other contentions. (Finding 7B:6C).

\*Two principal points were made in Mr. Conran's affidavit. First, he believed that the Staff's program in support of the resolution of unresolved safety issue A-17 had declined to such an extent over the last several months that it could no longer provide the basis for the finding required by caselaw<sup>5/</sup> that reasonable assurance existed that Shoreham could be operated safely despite the pendency of the unresolved systems interaction issues involved in A-17. Second, Mr. Conran testified that, contrary to his earlier belief and testimony, LILCo's failure to use the term "important to safety" evidenced a substantive defect in LILCo's understanding of the regulations rather than a mere terminological difference. (Findings 7B:191B, 141A).

\*The Staff's position as reflected in earlier testimony did not change as a result of Mr. Conran's affidavit, and the Staff panel reaffirmed the positions taken by the Staff on these issues. (Findings 7B:141D, 191 K). The Staff emphasized, however, that the difference between it and the Applicant over the correct use of the term "important to safety" still existed. (Findings 7B:136). The County's additional testimony essentially agreed with the Conran affidavit and the County's earlier testimony. (Finding 7B:6B). LILCo's panel explained why LILCo continued to oppose adoption of the Staff's definition and argued that Shoreham satisfied regulatory requirements under either construction of "important to safety". (Findings 7B:44A, 138, 138A).

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<sup>5/</sup> See Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2, ALAB-471, 8 NRC 245 (1978)).

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."<sup>6/</sup> 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
- e. Is it necessary to apply the alternative methodologies cited by Intervenor's 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 Intervenor, a "proper systematic methodology" has been used to analyze the reliability

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<sup>6/</sup> 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.

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. Imposition of a license condition embodying the classification definitions and mandating specific steps toward its systematic implementation during plant operation will reduce the potential for confusion in the regulatory relationship and will clarify LILCo's legal obligations in such areas as non-safety-related quality assurance, reporting requirements and inspection.

Intervenors would have us find that various types of additional analytical techniques, 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 assuring adequate safety in 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). Reasonable assurance of no undue risk to public health and safety depends in an important way on the concept of defense-in-depth and the many structures, systems and components required by this principle. Id.

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.<sup>7/</sup>

"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; 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." 10 CFR Part 50, Appendix A, Introduction. The interpretation 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

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<sup>7/</sup> 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).

"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 in 10 CFR Part 50, Appendix A, Introduction as those which provide 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 ~~in 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).<sup>8/</sup> The Denton memorandum explains that safety-related is a subset of the class of important to 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).

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8/ A slightly different formulation of the term "safety-related" was recently provided by the Commission's revision of 10 CFR § 50.49(b). "Safety-related" is defined there as that equipment that is relied upon to remain functional during and following design basis events to ensure: (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines.

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."  
(emphases 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.<sup>9/</sup>

\*Even in the absence of the Commission's recent revision of 10 CFR § 50.49, however, we would reject LILCO's construction of "important to safety". Several factors lead us to this conclusion.

\*First, the General Design Criteria include certain criteria which specifically address non-safety-related items in the plant. For example, GDC-60 requires radioactive effluent control and treatment equipment, which is non-safety-related. (See Finding 7B:26). The Introduction to

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<sup>9/</sup> See, e.g., Northern States Power Company (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 51 51 (1978).

Appendix A states clearly that the General Design Criteria established minimum requirements for the necessary design, fabrication, construction, testing and performance requirements "for structures, systems and components important to safety." Since the General Design Criteria specifically include certain non-safety-related items, the phrase "structures, systems and components important to safety" in the Introduction cannot reasonably be construed as being limited to safety-related items. The broader construction for which the Staff argues eliminates such internal inconsistencies.

\*LILCo argues, nevertheless, that the term "important to safety" which appears in several places in 10 CFR Part 50, Appendix A (and elsewhere) was substituted for several phrases in published draft regulations, each of which phrases referred to safety-related engineered safety features. LILCo suggests that the absence of any clear explanation for the use of a different phrase in the final regulations supports an inference that no substantive change was intended. LILCo's Proposed Opinion at 33.

\*In the draft GDC-1 published in 1967 (32 Fed. Reg. 10213, July 11, 1967), the phrase "essential to the prevention of accidents which could affect public health and safety or to mitigation of their consequences" was used rather than "important to safety"; in common parlance, the latter term would clearly be broader in meaning than the former. As LILCo states, other phrases in the draft criteria are also replaced with the words "important to safety." Neither the statement of consideration published at the time Appendix A was promulgated (36 Fed. Reg. 3255, February 10, 1971) nor the Commission paper discussing the final regulation (SECY-R 143, January 28,



1971) shed any further light on the intent of the Commission in changing the various phrases in the draft criteria to "important to safety". LILCo argues from the lack of explicit discussion of this substitution of phrases that no substantive change was intended. Accepted principles of statutory construction do not support such an inference. If words used in a regulation or statute to express a certain meaning are omitted, the proper presumption is that a change of meaning was intended. See, e.g., Chertkof v. United States, 676 F.2d 984, 987-88 (4th Cir. 1982). While draft regulations were involved here, the operative principle is the same since the Commission issued the draft criteria with the specific statement that they would be used as interim guidance until final criteria issued. See 32 Fed. Reg. 10213, 10,214, July 11, 1967.

\*LILCo seeks to make much of the fact, for example, that "important to safety" has been used at times in the final General Design Criteria where the term used in the draft criteria had been "engineered safety features". Thus, for example, LILCo cites GDC 44, which requires a system to transfer heat for structures, systems and components important to safety to an alternate heat sink. LILCo argues that GDC 44 evolved from proposed criteria 37, 38 and 39, which addressed the general design bases for engineered safety features, and that GDC 44 is only intended to refer to certain safety-related engineered safety features. The fatal flaw in LILCo's argument is that GDC 44 made an important addition to proposed criteria 37, 38 and 39 -- the safety function of the cooling water system is to transfer the combined heat loads of important to safety structures, systems and components under normal operating conditions as well as accident conditions. GDC 44, far from supporting LILCo, makes it clear that "safety

function" refers to normal operation as well as accident conditions and that "important to safety" has broader meaning than the prevention or mitigation of the critical safety function of 10 CFR Part 100, Appendix A. This example supports the Staff rather than LILCo.

\*Appendix B to 10 CFR Part 50 is also cited by LILCo in support of its position that "important to safety" is equivalent to "safety-related." LILCo focuses in particular on that part of 10 CFR § 50.34(c)(7) which states that Appendix B "sets forth the requirements for quality assurance programs" (emphasis added by LILCo), arguing from these words that Appendix B's scope is equivalent to that of GDC-1. This argument ignores the surrounding language and leads to results so obviously inconsistent with public health and safety that it must be rejected. The sentence which precedes that cited by LILCO in § 50.34(a)(7) states that the preliminary safety analysis report must include a description of the quality assurance program applied "to the design, fabrication, construction and testing of the structures, systems and components of the facility." (emphasis added). The underscored words are not modified by either of the phrases "safety-related" or "important to safety" which appear elsewhere. The sentence following that cited by LILCo states that "[t]he description of the quality assurance program . . . shall include a discussion of how the applicable requirements of Appendix B will be satisfied." (emphasis added). Contrary to LILCo's suggestion, these provisions strongly suggest that the quality assurance program applies to more than those plant items covered by the Appendix B program for safety-related items. Again, LILCo's example supports the Staff's interpretation.

\*The fundamental problems with LILCo's argument on the meaning of important to safety generally is that it puts LILCo in the position of arguing that the Commission's regulations impose no quality assurance requirements for the many structures, systems and components of a nuclear power plant which are not safety-related but which play a role in the safe operation of the plant. In concrete terms, LILCo is arguing that such systems as the effluent control systems and fire protection systems are not subject to any present quality assurance requirement under the Commission's regulations. Such a position is inconsistent with the obvious safety significance of these and many other systems which are placed in a nuclear plant for purposes other than the performance of the three critical safety functions of 10 CFR Part 100, Appendix A. LILCo's witnesses attempted to soften this position by arguing that LILCo has good quality assurance measures in place for non-safety-related as well as safety-related items. (See Finding 7B:50A). This misses the point. Even LILCo agrees that what it calls non-safety-related items are necessary to meet the performance-oriented requirements of 10 CFR Part 2 and 10 CFR Part 50, Appendix I. The existence of such performance requirements in the regulations and the need to rely on other than safety-related equipment to meet them demonstrates precisely why it is important that a regulatory requirement exist, as in GDC-1, imposing an obligation to adhere to quality standards and quality assurance measures commensurate with the importance to safety of the particular item. (Finding 7B:50A).

\*The Staff has also pointed to 10 CFR § 50.59 and 10 CFR Part 21 as examples of areas in which LILCo's narrow construction of "important to

safety" can have an impact on safety. Under LILCo's construction of "important to safety" in 10 CFR § 50.59(a)(2), an unreviewed safety question (requiring prior notice to the NRC) would not be presented by a facility modification which increased the "probability of occurrence or the consequences of an accident or a malfunction" of non-safety-related equipment previously evaluated in the FSAR. (Finding 7B:50B). LILCo argues that "using LILCo's interpretation, a § 50.59 review must be done on every plant modification, whether safety-related or non-safety related, to determine whether there is an unreviewed safety question involved." In LILCo's view its interpretation of the words "important to safety" in § 50.59(a)(2) makes no difference since every plant modification is reviewed and reported either before the modification is made or after. See LILCo Proposed Finding B-259V, B-259W. The timing of the report, however, is the critical aspect of § 50.59 affected by LILCo's interpretation of "important to safety." Under LILCo's construction, if the "probability of occurrence or the consequences of an accident or malfunction" of non-safety-related equipment previously evaluated in the FSAR is involved, then an unreviewed safety question is not presented and there is no obligation to report the proposed change prior to making it so that the NRC can evaluate the matter itself before any action is taken. Similarly, LILCo's construction of 10 CFR Part 21 leaves LILCo free (despite any present intention it may have expressed to the contrary) not to report safety problems which the Staff expects licensees to report to the NRC under the Staff's broader reading of Part 21. (See Finding 7B:50C).

\*LILCo makes a further argument based on the "legislative history" of 10 CFR Part 100, Appendix A, taking a change from the draft regulation to

the final regulation as evidence that no change in meaning was intended. As stated above, the opposite inference is equally available and preferred by common rules of statutory construction. At best, the removal of the phrase "important to safety" from the draft rule is ambiguous in terms of evidencing the intent of the Commission. The addition in the final rule of a reference to GDC-2 is not as telling as LILCo suggests. It is not unusual for a general design criteria to be given greater specificity through a regulation as opposed to through Staff regulatory guidance (see, e.g., GDC-4 and 10 CFR § 50.49). Indeed, the specificity which 10 CFR Part 100, Appendix A provides for GDC-2 undercuts the "parade of horrors" argument made by LILCo's witnesses against the Staff's interpretation of important to safety.<sup>10/</sup>

\*LILCo's attempt to use 10 CFR § 50.54 to support its construction of "important to safety" must also be rejected. LILCo quotes language from the statement of consideration (48 Fed. Reg. 1826 (1983)) that "the QA program description becomes a principal inspection and enforcement tool in ensuring

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<sup>10/</sup> LILCo's witnesses testified to their belief that LILCo complied with the intent of the broader definition of "important to safety" but expressed concern about other areas in the regulation where the same term is used, e.g. GDC-2. See Finding 7B:138C. As the TMI-1 Restart decision noted, the language of the regulations typically is broadly drawn so as not to be too prescriptive and to permit flexibility in the implementation of those requirements. Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit No. 1), LBP-81-59, 14 NRC 1211, 1246 (1981). General requirements such as GDC-2 have been given specific content through their application and administration by the NRC, as well as through other regulations as discussed above. See Natural Resources Defense Council v. NRC, 582 F.2d 166 (D.C. Cir. 1978) (administrative interpretation, practice and usage accorded great weight in interpreting statutes); Immigration and Naturalization Service v. Stanisic, 395 U.S. 62, 72 (1969) (administrative agency's interpretation of its own regulation is controlling unless plainly erroneous or inconsistent with the regulation).

that the permit holder or licensee is in accordance with all NRC quality assurance requirements . . . ." (emphasis added by LILCo) and argues from this that implementation of the Appendix B program "constitutes compliance with all NRC quality assurance requirements, including, necessarily, GDC-1." (emphasis LILCO's). However, the use of the words "a principal" in the statement of consideration strongly implies the existence of other inspection and enforcement tools to ensure compliance with quality assurance regulatory requirements. The implication that Appendix B is the only such requirement is conjured out of thin air.<sup>11/</sup>

\*Finally, we need pause only briefly to dispose of LILCo's argument that, unless "important to safety" is equivalent to "safety-related", 10 CFR Part 50, Appendix A was promulgated without adequate notice and in violation of the Section 553(b)(3) of the Administrative Procedure Act. That section requires that the notice of rulemaking include either the terms or substance of the proposed rule or a description of the subjects and issues involved. Here the specific terms of the proposed rule were provided, as required. See 32 Fed. Reg. 10213 et seq., July 11, 1967. New notice and opportunity to comment is not required merely because the terms of the final rule varied from those of the proposed rule:

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<sup>11/</sup> LILCo's citation of 10 CFR § 72.15(a)(14), which uses the terms "important to safety" and "safety-related" in close conjunction, succeeds only in demonstrating that the drafters of the regulations may themselves have confused the terms over the years, a point made by testimony in this proceeding (see Findings 7B:50C, 79). It is not difficult to harmonize this regulation with the body of the remaining regulations by reading the reference to "those safety-related components, systems and structures" as modifying the earlier reference to "structures, systems and components important to safety"; such a reading is consistent with the fact that "safety-related" is a subset of "important to safety" under the Staff's construction.

Simply because a different rule is adopted does not require a new notice and comment procedure if, as required by 5 U.S.C.A. § 553(b)(3), the notice of proposed rulemaking includes the terms or substance of the proposed rule or a description of the subjects and issues involved. This requirement is to sufficiently and fairly apprise interested parties of the issues involved, rather than to specify every precise proposal that the agency may ultimately adopt.

Pennzoil Co. v. FERC, 645 F.2d 360, 371 (5th Cir. 1981). The final rule promulgating the general design criteria was clearly a logical outgrowth of the proposed rule published for comment. See Connecticut Light and Power Co. v. NRC, 673 F.2d 525, 532-34 (D.C. Cir. 1982) (upholding fire protection rule where "final rules were simply more stringent versions of the proposed rules"); Chrysler Corp. v. Department of Transportation, 515 F.2d 1053, 1061 (6th Cir. 1975) (upholding safety regulation where "the regulation as adopted did not embrace any major subjects that were not described in the notice of proposed rulemaking"). Construing "important to safety" broadly, as the Staff does, is not precluded by the Administrative Procedure Act.

\*\*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."<sup>12/</sup> 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 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

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<sup>12/</sup> 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). We reject as unsupported the suggestion by LILCO that "Shoreham record has gone well beyond that in TMI-1" (LILCO Proposed Opinion at 48) and the unspoken inference that the TMI-1 decision sheds no light on the definitional issue presented here. We do not know, and LILCO did not proffer evidence to tell us, what the evidentiary record was upon which the TMI-1 Board based its decision in support of the Staff's definition. We do know, from Mr. Conran's testimony and particularly Attachment R-1 to the Staff's July 1, 1982 rebuttal testimony through Mr. Conran (ff. Tr. 6368), that the position taken by the Staff here is consistent with that taken by the Staff at TMI-1.



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. It was intended, rather, to eliminate a terminological problem which had arisen because individual Staff members had in the past used the terms inconsistently.<sup>13/</sup> It was addressed to the misapplication of the safety classification terms and the potential for confusion that resulted from such misapplication. (Finding 7B:48).

\*LILCo argues that the Denton memorandum expressed a new definition of important to safety contrary to established industry and Staff practice. (See Finding 7B:44A). The weight of evidence is to the contrary. The Denton memorandum was issued to clarify that the regulations require licenses to pay attention to equipment that contributes to safety in ways

beyond the "gold-plated, dedicated, accident-related systems." (Findings 7B:48A, 48B, 48D). Substantive Staff practice (as opposed to the terminology used by particular Staff members) in applying the concept of "important to safety" has been consistent in accordance with the intent now clarified in the Denton memorandum. (Finding 7B:48B). In the licensing review process, for example, it is and has been consistent Staff practice to review particular structures, systems and components important to safety but not safety-related. Staff witness Speis estimated that approximately 25% of the Staff's review effort is directed to this class of plant items. (Finding 7B:35).

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 over-narrowly 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

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13/ LILCo suggests in its proposed initial decision (at 43), without record citation or any evidence whatsoever, that 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.

create a void in the regulations that provide assurance of public health and safety.<sup>14/</sup> (Finding 7B:50).

\*LILCo complains that a definition of important to safety which refers to all plant items that provide reasonable assurance of no undue risk to the public health and safety is unreasonably vague and open-ended. (Finding 7B:138). We cannot agree that the broader definition sets standards which lie beyond LILCo's ability to operate and establish auditable procedures. LILCo witnesses testified that they were comfortable with the term "safety significance." It is clearly no more difficult to work with and audit against the concept of "important to safety" than against "safety significance." (Findings 7B:138, 138A). LILCo already has in place a graded approach to treatment of items in the plant based on LILCo's judgment as to the significance of the item involved in terms of safety, reliability, operability and maintainability. The same judgments that LILCo is already making would be required under GDC-1 using the broader definition of "important to safety." (Findings 7B:138B, 138C). In the final analysis, responsibility for the safety of Shoreham lies with LILCo. (Finding 7B:138).

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<sup>14/</sup> 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 to 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 plan 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<sup>15/</sup> 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.

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<sup>15/</sup> By "deterministic criteria," we mean established qualitative standards or requirements rather than numerical or probabilistic goals. (Finding 7B:206).

(Finding 7B:24).<sup>16/</sup> 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 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:25-35). 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).<sup>17/</sup>

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<sup>16/</sup> LILCo used Regulatory Guide 1.70, Revision 1, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," dated October 1972, and other applicable regulatory guides in the preparation of its FSAR. (Finding 7B:20A).

<sup>17/</sup> 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.

This is the general methodology which has been utilized in the design and review of the Shoreham plant. Intervenor's 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. Intervenor's do not allege that Applicant and the Staff have failed to use any methodology in the analysis and classification of plant structures, systems and components. Rather, Intervenor's 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 those 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 Intervenor's and the adequacy of Applicant's evaluation of systems interactions at Shoreham particularly in relation to a specific system selected by Intervenor's. Finally, we address the alleged need for the alternative methodologies discussed by Intervenor's 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.<sup>18/</sup> 10 CFR 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

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<sup>18/</sup> 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).

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 through 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 requirement 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 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 Intervenor 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, Intervenor's 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 system. (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) rod block monitor (RBM)

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 in 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 treatment 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 interpretation 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, i.e., 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).

\*Nevertheless, an important area of disagreement between the Applicant and the Staff remains because of Applicant's opposition to the Staff's interpretation of "important to safety". In its rebuttal testimony filed through Mr. Conran in July 1982, the Staff identified certain "unacceptable implications" of Applicant's incorrect use of "important to safety". These concerns are of particular importance in the operation of Shoreham. (Finding 7B:136).

\*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, taken together with our finding that the Staff's interpretation of "important to safety" is correct and the license condition we impose below, is sufficient to allay that concern. Applicant's testimony that no substantive differences have resulted from its different usage of the term (Finding 7B:131) stands uncontradicted and is, indeed, reinforced by the record generally. Moreover, Applicant has provided a commitment which is the functional equivalent of the commitment usually provided (here at FSAR § 3.1.2.1) to comply with GDC-1 during the operation of Shoreham. (Findings 7B:136-136F). We see no need to require that the FSAR be re-reviewed or that the scope of the review be expanded. (Finding 7B:135, 1410-141R).



\*The second concern expressed by the Staff was that it was clear under the Staff's interpretation of "important to safety" that there exists in the regulations a requirement under GDC-1 for a quality assurance program for certain non-safety related structures, systems and components (i.e., those important to safety). (Finding 7B:136). LILCo acknowledges no such regulatory requirement. Similarly, the Staff's third area of concern was that Applicant could overnarrowly construe its reporting obligations under such regulations as 10 CFR § 50.59 and 10 CFR Part 21. (Finding 7B:136).

\*As noted in the earlier discussion of the correct interpretation of "important to safety", we find these concerns persuasive. There is also substantial evidence that continued use by LILCo of a definition different than the Staff's will cause confusion and that such confusion will adversely affect safety. (Finding 7B:136A, 136J).

\*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. (Finding 7B:137). In order to avoid the confusion inherent in the use of different definitions of the term "important to safety" by LILCo and the Staff, and to minimize difficulties which may otherwise arise in terms of reporting obligations, inspection and quality standards and quality assurance requirements, the following condition shall be made a part of any operating license which may issue for the Shoreham Nuclear Power Station:

"Safety-related" structures, systems and components are those which are relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safety shutdown condition, and (iii) the capability to prevent or mitigate exposures comparable to the 10 CFR Part 100 guidelines. See 10 CFR § 50.49(b)(1). "Important to safety" structures, systems and

components are those which provide reasonable assurance that the facility can be operated without undue risk to public health and safety (See 10 CFR Part 50, Appendix A (Introduction)) and include the "safety-related" structures, systems and components as a lesser subset. LILCo shall take appropriate steps prior to operation of the Shoreham Nuclear Power Station to disseminate these definitions to all employees associated with Shoreham and to instruct all such employees to use these terms properly in all communications within the company, to its contractors and with the NRC and its Staff. LILCo shall also disseminate and require adherence to the commitments contained in LILCo's March 8, 1983 letter to the NRC Staff that all non-safety related structures, systems and components and plant computer software will be accorded, as a minimum, the safety significance given to them in the FSAR, as amended, technical specifications and emergency operating procedures. These structures, systems and components shall henceforth be appropriately termed as "safety-related" or "important to safety" as defined above. LILCo shall further conduct a review of its FSAR, as amended, and correct all uses of the term "important to safety" inconsistent with the definition appearing above. The results of this review, and appropriate amendments resulting therefrom, shall be included in the updated FSAR filed in accordance with 10 CFR § 50.71(e)(3)(i).

\*Mr. Conran and the County would have us go further. Mr. Conran testified that there is a conceptual difference, as well as a terminological difference, between the Staff and LILCo. He believes that "LILCo truly does not understand what is required minimally for safety by NRC under the regulations . . . ." (Finding 7B:141). Mr. Conran argues that the imposition of a definition is not adequate under these circumstances. (Finding 7B:140). He feels that LILCo should be required to develop and demonstrate the requisite understanding of what is minimally required for safety in the operation of Shoreham by preparing

a listing of Shoreham's important to safety structures, systems and components. (Finding 7B:141B). The County agrees.

\*We are satisfied, as is the Staff, that LILCo does understand what is minimally required for safety despite the position it has taken with respect to the Staff's interpretation of "important to safety". The record of this proceeding demonstrates that LILCo has satisfied the deterministic criteria embodied in the Staff's Standard Review Plan, other regulatory guidance documents, and appropriate industry standards and practices. LILCo has also described to the Staff's satisfaction its organization to address facility operation as well as its programs to conduct and audit plant activities in such areas as preventive and corrective maintenance, procurement and storage, and design change control. A further commitment regarding these programs has been placed in the record. (Findings 7B:136C-136F). The evidence of proper design and construction, coupled with LILCo's programs and additional commitments for operating the facility, demonstrates that LILCo understands what is minimally required to operate the facility without undue risk to the health and safety of the public. (Finding 7B:141E).

\*We do not draw from LILCo's resistance to a regulatory definition other than the one it believes to be appropriate the conclusion that LILCo does not understand what is minimally required for safety under the regulations. (Finding 7B:141F). LILCo management testified that LILCo is fully implementing the intent of the Staff's construction of "important to safety" in its programs despite the legal position it has taken before us on the appropriateness of that construction. (Findings 7B:141G-141H).

\*As to Mr. Conran's position that a list should be required of LILCo, listing of non-safety-related plant items which are "important to safety" is not necessary to demonstrate an understanding of what is minimally required for safety nor would it demonstrate such an understanding. (Finding 7B:141I). Mr. Conran himself admitted that in the past he never thought such a list was really necessary. (Finding 7B:141J). Such a list could be generated, to be sure. (Finding 7B:141L). What is important, however, is not the list but the system or process for identifying the important attributes of a structure, system or component and the mechanism for assuring that those attributes are preserved through the life of the plant. (Finding 7B:141M). We believe LILCo has demonstrated that it has such systems and mechanisms in place.

\*Mr. Conran testified that meetings, discussions and the exploration of examples could enable one to determine whether there is really a mutual understanding as to what is required for safety. That is precisely what this Board has undertaken on the record of this proceeding. (Finding 7B:141M). We do not accept Mr. Conran's contention that LILCo lacks essential understanding of what is required for safety and that an "important to safety" list must be required for Shoreham.

4. Analysis of Systems Interactions at Shoreham

a. Applicant's evaluation of systems interaction at Shoreham

One of the important concerns raised by Intervenor's testimony is that no adequate evaluation has been done of potential adverse systems interactions at Shoreham. Intervenor's cited the water level indication system as an example of a system subject to adverse interactions. Extensive testimony has 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.

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, electric 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 both included both deterministic and probabilistic methodologies. Major parts of this evaluation are documented on 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. Water level indication system interactions

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 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. (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,<sup>19/</sup> 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.<sup>20/</sup> 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 solution has been implemented for the particular facility, that a restriction in the level or nature of operation has been

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<sup>19/</sup> Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978).

<sup>20/</sup> 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 Contention 7B. These motions were denied. See Tr. 1093-1103.

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 operating pending resolution of the generic issue may be that the 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).

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 possible 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 had 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:179-182).

\*More recently, however, consideration has been given to applying the Staff's candidate methods to Indian Point Unit 3 in order to provide a comparison with the PASNY method of analysis. This is the preferred alternative at the present time and the Staff has secured the cooperation of PASNY for the comparative methodology demonstration. The Staff expects to receive the results of this study in July 1984. (Findings 7B:182-183).

\*\*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 obligation under North Anna to explain why operation of Shoreham may be permitted despite the pendency of unresolved safety issue A-17. (Finding 7B:202).~~

\*This conclusion is based on the fact that the existing regulatory framework addresses the systems interaction concern by evaluating plant designs against well-established deterministic requirements and criteria which are embodied in regulatory guidance documents. These current requirements are founded on the principle of "defense-in-depth" (see Findings 7B:7-19) and 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, fire and flooding. 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. (Finding 7B:185). The Shoreham application was evaluated against these requirements. (Finding 7B:186). LILCO has given extensive consideration to potential systems interactions at Shoreham, even going beyond Staff requirements for systems interaction analysis in a number of areas. (Findings 7B:191G, 191S)

\*The systems interaction issue is one of the two areas in which Mr. Conran has modified his earlier testimony through the submission of

his February 9, 1983 affidavit. The affidavit expresses Mr. Conran's present view that the Staff's program for resolution of A-17 has declined in recent months to such an extent that he no longer believes that it is currently adequate to provide a basis for the "justification for operation" conclusion required under North Anna. Without adequate progress toward resolution of A-17, Mr. Conran could not conclude that there is reasonable assurance that Shoreham could be operated with no undue risk to public health and safety. (Findings 7B:191A-191D).

\*Mr. Conran's affidavit, which was received in evidence during the reopened hearings (Tr. 20,401), discusses at some length the bases for his change in position in terms of the history of the Staff's systems interaction efforts, events of recent months and his estimation of the schedule on which action toward resolution can proceed. He concludes that a requirement should be imposed by the Staff at this time for limited systems interaction analysis by licensees and operating license applicants. (Finding 7B:191C, 191J). In particular, Mr. Conran's affidavit was precipitated by the cumulative effect of: (1) the loss of the pilot plant demonstration option at selected near-term operating license plants; (2) the delay in availability of data from systems interaction studies undertaken at Diablo Canyon and Indian Point, and (3) what he sees as the lack of any serious indication by Staff management that some other measures would be taken given these circumstances, such as the initiation of limited studies by near-term operating license applicants. (Finding 7B:191D).

\*As a result of the Conran affidavit, the Staff provided supplemental testimony which addressed Mr. Conran's present views on the

systems interaction issue. The Staff's position as reflected in earlier testimony on A-17 and systems interaction has not changed. In brief, that position may be summarized as follows: (1) the Staff's current licensing requirements provide reasonable assurance of no undue risk to public health and safety from potential adverse systems interactions; (2) the A-17 program is confirmatory in nature; (3) the Staff's program on A-17 is progressing toward resolution; (4) Shoreham may be licensed for operation despite the pendency of A-17; and (5) no plant specific systems interaction analyses (other than those now required by regulation or Staff practice) are or should be required until completion of the Staff's program determines whether they are necessary and justified. (Finding 7B:191K).

\*We find that Shoreham may be licensed for operation despite the pendency of unresolved safety issue A-17. (Finding 7B:191X). Several independent bases exist for concluding that the North Anna requirement has been satisfied for Shoreham.

\*First, we agree with the Staff that the nature of the particular issue involved should be factored into the North Anna determination. (Finding 7B:191N). A-17 is a confirmatory task. (See Finding 7B:176). The existing regulatory framework adequately addresses the systems interaction concern and progress in the A-17 program to date has provided no indication that present requirements and review procedures do not provide reasonable assurance that the effects of potential systems interactions on plant safety will be within the effects on plant safety previously evaluated (i.e., within the design basis envelope). (Finding 7B:185-190, 191Q). This is so irrespective of the schedule for

resolution of A-17. (Finding 7B:191L, 191M). Indeed, the Staff is not aware of any major interactions that are not already considered under the regulations. (Finding 7B:191M).

\*Second, in addition to the adequacy of existing regulatory requirements to support the required North Anna finding for Shoreham, LILCo has gone beyond Staff requirements for systems interaction analysis in several areas. (Finding 7B:191S; see Findings 7B:191G, 191H). In Mr. Conran's view, this consideration of systems interactions by LILCo specifically for Shoreham would provide an adequate basis for licensing Shoreham under North Anna if the safety classification issue could be resolved. Mr. Conran's residual concern is that LILCo, because of its allegedly different understanding of the importance of non-safety-related items, might have a different judgment as to the safety significance of interactions identified in its various systems interaction studies. (Finding 7B:191H). Because systems interaction studies are conducted independent of classification and because LILCo has adequately demonstrated its understanding of what is minimally required for safety, the Staff does not share Mr. Conran's concern. (Findings 7B:191U, 7B:141E-141H). Neither does this Board.

\*Third, there has been reasonable progress in the Staff's program for resolving A-17. (Finding 7B:191O; see Findings 7B:177-184). Even with respect to events over the last six months, the time frame focused on by Mr. Conran's affidavit, the Staff believes there has been sufficient progress during that time period to indicate that the Staff is moving toward resolution of A-17. (Finding 7B:191P). The Staff presently expects to complete its review of various systems interaction

studies, assess the efficiency of the methodologies used in these studies, and make a decision on the need for any requirement for plant-specific systems interaction analyses by October 1984. (Finding 7B:1910). The Staff judgment, which we find reasonable on the basis of the evidentiary record, is that A-17 will be resolved within such a time frame that there will not be undue risk to the public from operation of Shoreham in the interim. (Finding 7B:191R). See North Anna, supra, 8 NRC 245, 248 (1978).

\*In sum, we find that an adequate explanation has been provided as to why operation of Shoreham may be permitted despite the pendency of unresolved safety issue A-17. The absence of a declared generic solution does not call into question the safety of current operation at this plant.

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 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."

(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 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. (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.<sup>21/</sup> (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.

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21/ 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") proffered 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. Intervenor 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, 1982. 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. Additional hearings after the record was reopened were held on April 5-8, 1983. Fourteen witnesses appeared during this period.

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 Division 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 Division of Systems Integration;<sup>1/</sup> 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,<sup>2/</sup> 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;

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

<sup>2/</sup> 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).

\*7B:6A After the close of the record on Contention 7B, on January 25, 1983, Staff counsel informed the Board and parties by letter that one of the Staff's witnesses who had testified in the proceeding on Contention 7B, James H. Conran, sought to modify certain of his testimony since he could no longer support some aspects of the testimony previously given by him. Mr. Conran prepared a written statement of his present views which was provided to the Board and parties on February 8.<sup>3/</sup> The Board then directed that the parties file statements of their views

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<sup>3/</sup> The February 8, 1983 statement was provided in unexecuted affidavit form. On February 9, 1983, an executed version of the affidavit was distributed by the Staff.



on the Conran submittal, particularly as to the need for the reopening of the record for receipt of the Conran submittal and for additional testimony by any party.

\*7B:6B Both the Staff and the County favored reopening the record; LILCo opposed such a step. After considering the arguments of the parties, the Board decided on February 24 that the record on Contention 7B should be reopened to receive Mr. Conran's statement in evidence and also to hear such testimony as was necessary in light of Mr. Conran's new testimony. The Board established a schedule for the filing of additional prefiled testimony and for further hearings. Supplemental Staff testimony was filed on March 10 by a panel consisting of the following: Roger J. Mattson, Director of the Division of Systems Integration; Richard H. Vollmer, Director of the Division of Engineering; Charles E. Rossi, a Section Leader in the Instrumentation and Control Systems Branch and a previous witness on this contention; Ashok C. Thadani, Branch Chief of the Reliability and Risk Assessment Branch and also a prior witness on this contention; and Franklin D. Coffman, Jr., Section Leader in the Systems Interaction Section of the Reliability and Risk Assessment Branch. The County filed supplemental testing on March 25 by a panel consisting of Messrs. Goldsmith, Minor and Hubbard, all of whom had testified previously. LILCo decided against offering additional testimony.

\*7B:6C Additional hearing sessions were held on April 5-8, 1983 in Riverhead. Mr. Conran's submittal and the prefiled supplemental testimony of the Staff and the County were received in evidence and cross-examination and Board questioning were conducted. On the

afternoon of April 7, after having heard the oral testimony of Mr. Conran and the staff witnesses, the Board asked the Applicant to provide additional oral testimony on certain aspects of the controversy. On April 8, additional testimony was given by a LILCo panel consisting of the following: Millard S. Pollock, Vice-President - Nuclear LILCo; James Rivello, Shoreham Plant Manager of LILCo; William J. Museler, LILCo's Director, Office of Nuclear; George F. Dawe, Supervisor of Project Licensing for Stone & Webster Engineering Corporation; and Brian McCaffrey, LILCo's Manager of Nuclear Compliance and Safety. Mr. Dawe had testified previously on Contention 7B; Messrs. Museler and McCaffrey had appeared as witnesses on other contentions.

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. Adequate safety depends on this defense-in-depth concept. Speis et al., ff. Tr. 6357, at 18. 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 failure 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.

7BP: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 C.F.R. 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 nonsafety-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 accom-

plishing 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 set-points are assumed. Conservative core parameters (such as heat fluxes, 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. See generally Speis et al., ff. Tr. 6368; Burns et al., ff. Tr. 4346.

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:20A LILCo used Regulatory Guide 1.70, Revision 1, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," dated October 1972, and other applicable regulatory guides in the preparation of its FSAR. Mattson et al., ff. Tr. 20,810, at 10.

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); Tr. 20,825-26 (Rossi); but see Tr. 20,408-09 (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); see Mattson et al. ff. Tr. 20,810, at 10. Adequate defense-in-depth is provided by these items. See Findings 7B:7-11, supra.

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); Tr. 21,026 (Hubbard) (Standard Review Plan includes "the majority" of important to safety items). For example, the turbine bypass is an example of a nonsafety-related system covered in the Standard Review Plan. 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, including safety-related items, are addressed throughout the Standard Review Plan and discussed throughout the FSAR. See Mattson et al., ff. Tr. 20,810, at 10. Dr. Rossi gave examples of design bases for nonsafety-related 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); Tr. 20,822 (Rossi). 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; Tr. 20,414-15 (Conran). 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; see Mattson et al., ff. Tr. 20,810, at 10. These items are essential to adequate safety. See Findings 7B:7-11, supra; 7B:54, 43, 50, infra.

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; see Mattson et al., ff. Tr. 20,810, at 10.

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). With respect to non-safety-related items, the Staff does not concentrate on the terms used to describe them as much as the design requirements and criteria applied. Tr. 20,832 (Rossi).

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; Mattson et al., ff. Tr. 20,820, at 8-9.

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; Mattson et al., ff. Tr. 20,810, at 8-9, 10, 12. 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., 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 defined in 10 CFR Part 50, Appendix A (Introduction) as 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~~ in 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~~ were repeated in a November 20, 1981 memorandum from Harold Denton, Director of the Office of Nuclear Reactor Regulation, to all NRR personnel (Minor et al., ff. Tr. 1113, Attachment 1). 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, 20,607 (Conran). Intervenors concur in these definitions. See Suffolk County Proposed Opinion, at 19.

\*7B:44. Applicant took the position that the terms safety-related and important to safety are synonymous and that both refer to the narrower set of plant items necessary to perform the accident prevention and mitigation functions cited in 10 CFR Part 100, Appendix A rather than the set of structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to public health and safety described in 10 C.F.R. Part 50, Appendix A. Tr. 4790 (Robare); Tr. 7057 (Haass) Tr. 21,047, 21,051 (Pollock). 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:44A. LILCO believes that the definition of "important to safety" argued for by the Staff is new and not equivalent to what has been accepted in the past. Tr. 21,052-53 (Dawe).

\*7B:44B. The improper equating of the safety terms "safety-related" and "important to safety" is something the Staff has seen and recognized both within the Staff and within the industry. Tr. 20,422-23; Tr. 20,453-54; Tr. 20,591-92 (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 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. Conran, ff. 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; 7762 (Conran); but see Tr. 21,144 (Pollock).

\*\*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). It was addressed to the misapplica-  
tion of the safety classification terms and the potential for confusion  
that resulted from such misapplication. Tr. 20,506 (Conran).

\*7B:48A The Denton memorandum was not intended to impose new  
technical requirements on any license or applicant. Minor, et al., ff.  
Tr. 1113, Attachment 1. It was issued because there appeared to be a  
need to clarify the fact that there is equipment beyond the  
"safety-related" that must be considered in terms of its importance to  
safety. Some licenses had failed to recognize that fact. Tr. 20,857  
(Mattson); but see Tr. 7839-40 (Conran).

\*7B:48B The question of what safety significance is to be accorded  
a structure, system or component must be answered on an ad hoc basis for  
the particular item involved. Scrutiny of operating experience and  
equipment failure experience and of overall license performance enables  
one to determine whether those ad hoc judgments are resulting in  
appropriate safety significance being accorded to items in the plant.  
The TMI-2 accident taught the Staff that some of these judgments were  
not "quite right." For this reason, the Denton memorandum was issued to  
clarify the need for licensees to pay attention to the "important to  
safety" equipment. Tr. 20,858 (Mattson). In this sense, the Denton  
definition of "important to safety" was a new definition. Tr. 20,853  
(Mattson). However, Staff practice in applying the concept of  
"important to safety" has been consistent in accordance with the intent  
of the regulations as now clarified in the Denton memorandum. Conran,  
ff. Tr. 6368, at 5-6; Tr. 7736-37 (Conran).

\*7B:48C The definition of the term important to safety has only been presented as an issue and imposed as an explicit requirement in the context of a licensing hearing in the case of the restart of Three Mile Island, Unit 1. Tr. 20,836 (Mattson).

\*7B:48D The regulations address not only the "gold-plated, dedicated, accident-related systems" but also other things that contribute to safety. Tr. 20,460-61 (Conran). Limiting the term "important to safety," for example, to only that equipment the failure of which could prevent the accomplishment of a critical safety function (i.e. 10 CFR Part 100, Appendix A) would exclude some normal reactor controls. Tr. 21,164 (Dawe).

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 that 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 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 A to Part 100. It also includes the lower release 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). This would be true for many items which even LILCo agrees have safety significance. Tr. 21,052-53, 21,078-79 (Dawe); Tr. 21,147, 21,151 (Pollock). 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. Conran, ff. Tr. 6368, 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).

\*7B:50A. LILCo disagrees that its definition of "important to safety" puts certain structures, systems and components beyond the regulations, arguing that such regulations as 10 CFR Part 20 and 10 CFR Part 50 Appendix I impose performance requirements for which non-safety-related equipment is necessary. See Tr. 21,076-77 (Dawe); LILCo Proposed Finding B-210A. The existence of such performance requirements in the regulations and the need to rely on other than safety-related equipment to meet them demonstrates precisely why it is important that the specific regulatory authority exist, as in GDC-1, imposing as a matter of regulatory requirement an obligation to adhere to quality standards and quality assurance measures commensurate with the importance to safety of the particular item. LILCo does not acknowledge that such a requirement now exists. LILCo does not believe

that NRC authority extends to non-safety-related items unless there are such performance requirements or other particularized regulations applicable to them. See Tr. 21,076-79 (Dawe); Tr. 21,102-03, 21,131-32, 21,151 (Pollock). When asked directly by the Board whether LILCo agreed or disagreed that plant structure, systems and components, including those beyond the safety-related, should be designed, fabricated, erected and tested in the future with quality standards commensurate with the importance of the safety factors to be performed, Mr. Dawe agreed that this "should be done." Tr. 21,078 (Dawe). Importantly, Mr. Dawe did not agree that this was mandated by regulation:

[W]e agree you have to do those things for everything in the plant. The 'have to' is not a regulatory requirement for everything in the plant, but it is not only the regulation that make these plants safe . . . .

Tr. 21,079 (Dawe) (emphasis added).

\*7B:50B. With respect to 10 CFR § 50.59, LILCo argues that "using LILCo's interpretation, a § 50.59 review must be done on every plant modification, whether safety-related or non-safety related, to determine whether there is an unreviewed safety question involved." In LILCo's view its interpretation of the words "important to safety" in § 50.59(a)(2) makes no difference since every plant modification is reviewed and reported either before the modification is made or after. See LILCo Proposed Finding B-259V, B-259W. The timing of the report, however, is the critical aspect of § 50.59 affected by LILCo's interpretation of "important to safety." Under LILCo's construction, if the "probability of occurrence or the consequences of an accident or malfunction of equipment" which is non-safety-related is involved, then an unreviewed safety question is not presented and there is no obligation to report the proposed change prior to making it so that the NRC can evaluate the matter itself before any action is taken.

\*7B:50C. As to 10 CFR Part 21, LILCo asserts that "LILCo and the industry approach the reporting requirements in § 50.55e and Part 21 by including in the evaluation systems and equipment that could have an adverse effect on safety whether or not that equipment is classified as safety-related or non-safety-related." See LILCo Proposed Finding B-259U. It is not an adequate response to the gap in 10 CFR Part 21 created by LILCo's construction of important to safety to say "we always do that anyway." The concern is that LILCo or any other regulated entity may consider itself free to change its reporting practices in the future to stop reporting that which it considers itself not legally bound to report because of an improperly narrow construction of 10 CFR Part 21.<sup>4/</sup>

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<sup>4/</sup> 10 CFR Part 21 includes the term "important to safety" and the Staff testified that it construes that regulation to apply to important to safety but not safety-related items (as well as safety-related items). Tr. 20,627 (Conran). During cross-examination, Mr. Conran was confronted with a staff document (NUREG-0302, marked for identification as LILCo Ex. 68 but not received in evidence) which suggests that Part 21 is intended to apply only to safety-related items. However, Mr. Conran testified from personal knowledge that the document "was put together by people in the regulatory standard[s] organization who understood the term 'safety-related' to be the same as 'important to safety', but in the broad sense." Tr. 20,628-30 (Conran). Because of the uncertainty over the intent of the author, LILCo Ex. 68, which was not received in evidence, would not be probative of any Staff practice of applying Part 21 only to safety-related items. Accordingly, LILCo's argument in note 33 at page 73 of its Reply (dated February 22, 1983) is without evidentiary support and must be rejected.

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<sup>5/</sup> 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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5/ 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

3R, 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. Intervenor's 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

78:74. 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,<sup>6/</sup> 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 in 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 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

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<sup>6/</sup> 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); Tr. 20,630 (Conran)), 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); Tr. 20,631-32 (Conran). 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).

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); Tr. 20,527-28 (Conran). In the specific case of GDC-1 quality assurance requirements for important to safety items, the Staff regards an acknowledgment of the requirement under the regulations (i.e., a commitment) to be necessary and sufficient evidence of compliance without additional guidance being given. Tr. 20,414, 20,547 (Conran).

b. Assessment of specific systems

7B:83. These general descriptions of the treatment of nonsafety-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 Intervenor's 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, Intervenor's 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. 4880-81 (Ianni); Tr. 4880 (Robare).

7B:87. An analysis of the 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 et al., 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 et al., 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 et al., 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 PWR 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.

7R: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.

7R: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.

7R: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:107. 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 condenser 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 overflow 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. Peis 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.<sup>7/</sup> 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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<sup>7/</sup> 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); Mattson et al., ff. Tr. 20,810, at 10; Tr. 20,834, 20,859 (Mattson).

\*\*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. Mattson et al., ff. Tr. 20,810, at 8-9; GERFAR, 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 (GERFAR); see Tr. 6537 (GERFAR); Speis et al., ff. Tr. 6357, at 10; Mattson et al., ff. Tr. 20,810, at 8-9, 10; Tr. 20,818, 20,821, 20,825-26 (Rossi); Tr. 20,872 (Mattson).

\*\*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); Mattson et al., ff. Tr. 20,810, at 10; Tr. 20,825-26 (Rossi).

\*\*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 or to expand the scope of its audit review despite the difference in Applicant's use of the language of the regulations. Tr. 7121-23 (Rossi, Hodges, Haass, Kirkwood); Tr. 20,860-61 (Mattson). 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. Nevertheless, an important area of disagreement between the Applicant and the Staff remains because of Applicant's refusal to recognize that the term "important to safety" is defined differently in the regulations and is considerably broader than "safety-related." Tr. 20,833 (Mattson). The Staff's concern is with respect to operation of Shoreham. Tr. 20,834 (Mattson). 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.<sup>8/</sup>

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 quality assurance program for certain non-safety-related structures, systems and components (i.e., those important to safety) which provide reasonable assurance that the facility can be operated without undue risk to the public health and safety. See 10 C.F.R. Part 50, Appendix A, Introduction.

3. Under Applicant's construction of "important to safety," the obligations imposed by 10 CFR. 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. See also 10 CFR § 50.59(a)(2). The Staff is concerned about LILCo's compliance with these reporting requirements. Tr. 20,852 (Mattson).

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<sup>8/</sup> 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). 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:136A. The Staff has also expressed concern about confusion which is likely to result from the interchangeable use of "safety-related" and "important to safety." See Tr. 20,591 (Conran). The potential for confusion is significant if LILCo does not use the correct definition of important to safety. Tr. 20,848 (Mattson); Goldsmith et al., ff. Tr. 20,903, at 28-29. LILCo management agreed "positively" that there is going to be confusion if LILCo continues to use "important to safety" one way and the Staff uses it another way. Tr. 21,127 (Pollock); but see Tr. 21,128 (Dawe), Tr. 21,129 (Pollock). Use of a common definition will lead to a decrease in confusion and better performance by the licensee and will make agency-licensee relations more efficient and better from the regulator's viewpoint. Tr. 20,835-36, 20,853 (Mattson). For example, there is a need to avoid confusion when an inspector has an interest in an important to safety item but a licensee objects that the item is not safety-related and therefore not within the inspector's purview. Tr. 20,853 (Mattson).<sup>9/</sup>

\*7B:136B. The Staff testified that it considered it necessary to obtain reconfirmation of LILCo's commitment to comply with GDC-1 during operations at Shoreham using the correct definition of important to safety. Tr. 7122-23 (Haass). After an exchange of letters failed to provide an acceptable commitment, the Staff requested a meeting with LILCo to discuss

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<sup>9/</sup> LILCo conceded that an NRC inspector has a legal right to access every place in the plant, including a right to inspect in program areas that are non-safety-related. Tr. 21,137-38 (Pollock). LILCo reserved its right, however, to contest inspection findings if an item is not in its view covered by the regulations. Tr. 21,137 (Pollock). In LILCo's view, areas other than the safety-related are not specifically covered by regulation. Tr. 21,141 (Pollock).



LILCo's plans for compliance with GDC-1 during operations. The Applicant described to the Staff's satisfaction its organization to address facility operation as well as its programs to conduct and audit plant activities including its preventive and corrective maintenance program, its procurement and storage programs, as well as its design change control program. Mattson et al., ff. Tr. 20,810, at 10-11. LILCo believes its programs address every piece of equipment in the plant from the point of view of safety significance. Tr. 21,134 (Pollock).

\*7B:136C. During the February 18, 1983 meeting, the Staff asked LILCo to make a suitable commitment in the FSAR itself that LILCo will comply with GDC-1 during operations. Following that meeting, the Staff's request for the FSAR amendment was formally issued to LILCo in a letter from D. Eisenhut to M. Pollock of the same date. Mattson et al., ff. Tr. 20,810, at 11; Staff Ex. 14, ff. Tr. 20,812. The specific request was as follows:

Amend the FSAR to commit for non-safety related structures, systems, and components, to include in the preventive and corrective maintenance program the design change control program, the procedures for procurement of equipment, the procedures for modifications and removal of equipment from service, and the QA program, a provision that, as a minimum, the equipment and associated software shall be accorded the safety significance given to it in the FSAR, the technical specifications and the emergency operating procedures. The charters and decisions of the Review of Operations Committee, the Offsite Nuclear Review Board, and the Manager of Quality Assurance shall also reflect these considerations.

Staff Ex. 14, ff. Tr. 20,812.

\*7B:136D. The Applicant's commitment to the Staff was made by letter on March 2, 1983. LILCo Ex. 69, ff. Tr. 20,654. In order to ensure that there was no misunderstanding as to the exact meaning of the Applicant's commitment, the Staff requested by letter on March 7, 1983 that the FSAR

amendment be sent to the Staff for review as soon as possible. Staff Ex. 15, ff. Tr. 20,812. On March 8, 1983, LILCo submitted examples of the language it intends to incorporate in the FSAR. LILCo Ex. 70, ff. Tr. 20,654. The Staff reviewed that language and found it acceptable. Mattson et al., ff. Tr. 20,810, at 12.

\*7B:136E. Through this and other such FSAR amendments, the Shoreham FSAR will reflect the commitment that, during operations, Shoreham's structures, systems and components will be accorded as a minimum the safety significance given to them in the FSAR, the technical specifications and the emergency operating procedures. The same commitment will be reflected in the Shoreham preventive and corrective maintenance program, the design change control program, procedures for procurement of equipment, procedures for modification and removal of equipment from service, and the applicable portions of the Quality Assurance program. This corporate policy will be present in the charters and decisions of the Review of Operations Committee, the Nuclear Review Board and the Independent Safety Engineering Group. Mattson, et al., ff. Tr. 20,810, at 13.

\*7B:136F. This commitment by LILCo is a supplementary statement which commits to continue the quality standards and quality assurance measures already in place. Tr. 21,123 (Dawe). The commitment extends to all structures, systems and components in the FSAR, technical specifications and emergency operating procedures. Tr. 21,124 (Dawe). The commitment has been documented in the FSAR in order to satisfy the Staff's concern over whether the philosophy and sensitivity to safety discussed by LILCo at the February 18, 1983 meeting would be carried forward at all times

by people in the company. Tr. 21,071 (McCaffrey). This FSAR commitment will be implemented and LILCo believes that its programs thereby meet the intent of the Staff's interpretation of important to safety. Tr. 21,097, 21,144 (Pollock).

\*7B:136G. LILCo has not, through its FSAR commitment or otherwise, committed to recognize that "important to safety" is broader than "safety-related." Tr. 20,833 (Mattson); Tr. 21,054-58 (Pollock, Museler).

\*7B:136H. Without an acknowledgement that important to safety is broader than safety-related, the FSAR amendment does not provide an acceptable basis for licensing. Mattson et al., ff. Tr. 20,810, at 11; Tr. 20,848, 20,850 (Mattson). When the Staff obtained the FSAR commitment from LILCo, the Staff was operating on the assumption that the correct interpretation of "important to safety" would be accepted by or imposed on LILCo. Tr. 20,848, 20,849-50, 20,851 (Mattson).

\*7B:136I. The FSAR commitment will ensure that plant items are "flagged" in a way that will permit a future maintenance person or other employee to consider and assess the safety significance of a given item. Tr. 20,874-75 (Mattson).

\*7B:136J. Adoption or imposition of the correct interpretation of "important to safety", however, would add to the FSAR commitment in such important areas as the reporting of information and inspection. Tr. 20,854 (Mattson). Future regulators and future plant operators will have less difficulty communicating on safety matters by subscribing to a common and correct definition in the future. Tr. 20,836 (Mattson). Use of such a definition would avoid the need to expend time and resources (as, for example, in this proceeding) to ensure that what LILCo says it is doing

is equivalent to what should be done under the Staff's definition.

Tr. 20,855-86 (Mattson). There is a need for a common basis for communication and understanding about the flags that are attached to equipment because of its safety significance. Tr. 20,853 (Mattson).

\*\*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. LILCo complains that a definition of important to safety which refers to all plant items that provide reasonable assurance of no undue risk to the public health and safety is "vague", "indefinite", and "open-ended" and does not provide LILCo with bounds within which to operate and to establish auditable procedures. Tr. 21,047; Tr. 21,053-54; Tr. 21,082-83 (Pollock). The Staff agrees that the outer boundary of "important to safety" is not very clear. Tr. 20,845, 20,876 (Mattson). However, that question is left to a licensee's judgment and is not the boundary of greatest importance. Tr. 20,846 (Mattson). Ultimate responsibility for the safety of Shoreham lies with LILCo. Tr. 21,132 (Pollock).

\*7B:138A. LILCo's argument that it is unable to audit for compliance with a broader definition of important to safety than it used must be rejected. LILCo finds the FSAR commitment "workable" and "auditable". Tr. 21,108 (McCaffrey). Mr. Pollock and Mr. Dawe both testified that they were comfortable with the term "safety significance" (which is used in the March 8 FSAR commitment). Tr. 21,099-100 (Pollock); Tr. 21,102 (Dawe); see Tr. 21,057 (Museler). It is clearly no more difficult to work with and audit against the concept of "important to safety" than against "safety significance".

\*7B.138B. LILCo has in place a graded approach to treatment of items in the plant based on LILCo's judgment as to the significance of the item involved in terms of safety, reliability, operability and maintainability. See Tr. 21,051, 21,147 (Pollock). LILCo maintains that it understands "what the requirements are to apply to nonsafety-related equipment in terms of its safety significance." Tr. 21,057, 21,072-73 (Museler). The same judgments that LILCo is already making would be required under GDC-1 using the broader definition of "important to safety."

\*7B:138C. The credibility of LILCo's vagueness objection was also undercut by LILCO's testimony rejecting different limitations on the outer bound of "important to safety" (for example, that equipment specifically addressed in the FSAR, technical specifications and emergency operating procedures). Tr. 21,125 (Pollock); but see Tr. 21,126-27 (Dawe) (no objection to such a definition of important to safety in relation to GDC-1).

\*7B:139. In order to avoid the confusion inherent in the use of different definitions of the term "important to safety" by Applicant and the Staff, and to minimize difficulties which may arise in terms of reporting obligations, inspection and quality standards and quality assurance requirements, the following conditions shall be made a part of any operating license which may issue for the Shoreham Nuclear Power Station:

"Safety-related" structures, systems and components are those which are relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down

the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. See 10 CFR § 50.49(b)(1). "Important to safety" structures, systems and components are those which provide reasonable assurance that the facility can be operated without undue risk to public health and safety (see 10 CFR Part 50, Appendix A (Introduction)) and include the "safety-related" structures, systems and components as a lesser subset. LILCo shall take appropriate steps prior to operation of the Shoreham Nuclear Power Station to disseminate these definitions to all employees associated with Shoreham and to instruct all such employees to use these terms properly in all communications within the company, to its contractors and with the NRC and its Staff. LILCo shall also disseminate and require adherence to the commitments contained in LILCo's March 8, 1983 letter to the NRC Staff that all non-safety related structures, systems and components and plant computer software will be accorded, as a minimum, the safety significance given to them in the FSAR, as amended, technical specifications and emergency operating procedures. These structures, systems and components shall henceforth be appropriately termed as "safety-related" or "important to safety" as defined above. LILCo shall further conduct a review of its FSAR, as amended, and correct all uses of the terms "safety-related" and "important to safety" inconsistent with the definitions appearing above. The results of this review, and appropriate amendments resulting therefrom, shall be included in the updated FSAR filed in accordance with 10 C.F.R. § 50.71(e)(3)(i).

\*7B:140. The Conran affidavit takes the position that the imposition of a definition upon LILCo is not adequate under the circumstances present here. Conran, ff. Tr. 20,401, at 31-32.

\*7B:141. In his affidavit, Mr. Conran states that it was not clear to him at the time of his previous testimony, but is now, that LILCo's stated position regarding the safety classification term "important to safety" is more than a terminological difference. Mr. Conran now believes that there is a conceptual difference as well, and stated that "LILCo truly does not understand what is required minimally for safety by NRC under the regulations . . . ." Conran, ff. Tr. 20,401, at 28 (emphasis in the

original). In Mr. Conran's view, LILCo has "a fundamentally different way of thinking about the degree of importance to safety" than the Staff and other utilities. Tr. 20,778 (Conran). His primary concern is whether LILCo will accord proper safety significance to non-safety-related items during operation of the plant. Tr. 20,674-75 (Conran); see Tr. 20,513-14 (Conran).

\*7B:141A. As bases for his changed testimony, Mr. Conran states that he has had the opportunity to consider longer and review more thoroughly the testimony of LILCo's witnesses, particularly at Tr. 5425-5449, that he has been struck by LILCo's continued resistance to using the Staff's definition of "important to safety," and that these two considerations have a synergistic effect when considered together. Conran, ff. Tr. 20,401, at 28-30; see Tr. 20,454-55, 20,457, 20,460, 20,571-72 (Conran). Mr. Conran's concern arises basically out of LILCo's refusal to agree that non-safety-related structures, systems and components are covered by regulation, particularly GDC-1. Tr. 20,482 (Conran).

\*7B:141B. Mr. Conran believes that LILCo should be required to develop and demonstrate the requisite understanding of what is minimally required for safety in the operation of Shoreham by preparing a listing of Shoreham's important to safety structures, systems and components. Conran, ff. Tr. 20,401, at 32-33.

\*7B:141C. Mr. Conran also suggests that the scope of the audit review conducted by the Staff be expanded to examine more examples in order to determine whether Shoreham is designed and constructed in compliance with regulatory requirements. Tr. 20,438, 20,450-51, 20,519-20, 20,672-73 (Conran). Contrary to the County's proposed finding S7B:26, Mr. Conran did not recommend that the Shoreham application "should be

re-reviewed". Rather, Mr. Conran testified that he "would be inclined to expand the scope of my audit . . ." (Tr. 20,438 (Conran)) and would give the Shoreham application a "more thorough review" than it has been given (Tr. 20,451 (Conran)). Nowhere does Mr. Conran say or intimate that the Shoreham application should be re-reviewed. It is the County that wants the application re-reviewed. Goldsmith et al., ff. Tr. 20,903, at 41-42.

\*7B:141D The Staff's position as reflected in testimony given previously by Staff witnesses in this proceeding on the subject of safety classification did not change as a result of Mr. Conran's affidavit or for any other reason. Mattson et al., ff. Tr. 20,810, at 8-9.

\*7B:141E. The Staff is satisfied that LILCo understands what is minimally required for safety. Mattson et al., ff. Tr. 20,810, at 10. LILCo has satisfied the deterministic criteria embodied in the Staff's Standard Review Plan, other regulatory guidance documents, and appropriate industry standards and practices. In addition, the Applicant has described to the Staff its organization to address facility operation as well as its programs to conduct and audit plant activities including its preventive and corrective maintenance program, its procurement and storage programs, as well as its design change control program. The evidence of proper design and construction, coupled with LILCo's programs for operating the facility, demonstrate that LILCo understands what is minimally required to operate the facility without undue risk to the health and safety of the public. Mattson et al., ff. Tr. 20,810, at 10-11. Contrary to Suffolk County proposed finding S7B:66, the conclusion by the Staff witnesses (other than Mr. Conran) that LILCo understands the importance of non-safety-related structures, systems and components (Mattson et



al., ff. Tr. 20,810, at 12) is not dependent on LILCo's acceptance of the broader definition of important to safety contained in 10 CFR Part 50, Appendix A, Introduction. In the Staff's view, it is the ultimate finding of reasonable assurance of no undue risk that cannot be made absent acceptance or imposition of the current definition. See Tr. 20,850 (Mattson).

\*7B:141F. It does not necessarily follow from LILCo's resistance to adopting a definition of a regulatory term other than the one it believes to be appropriate that LILCo does not understand what is minimally required for safety under the regulations. Mr. Conran conceded that one can ascribe tremendous safety significance to a particular structure, system or component and still deny that it is covered by a particular regulatory phrase such as "important to safety." Tr. 20,477 (Conran).

\*7B:141G. LILCo management testified that LILCo does not disagree with the philosophy underlying the Staff's definition of "important to safety." Tr. 21,050, 21,053 (Pollock). Rather, LILCo has a problem with the words of the definition in that LILCo perceives a lack of specificity which makes it difficult to establish auditable procedures for compliance. Tr. 21,053-54, 21,067, 21,070, 21,130 (Pollock). LILCo believes that it is fully implementing the intent of the broader definition of "important to safety" in its programs. Tr. 21,097, 21,151 (Pollock). LILCo does not, by contesting the meaning of "important to safety," intend to say that it believes that what it calls non-safety-related items require no attention because of a lack of safety significance. Tr. 21,161 (Museler).

\*7B141H. The testimony of LILCo witnesses cited by Mr. Conran as one source of his concern about LILCo's understanding (Tr. 5425-5449) does not

support Mr. Conran's conclusion that LILCo does not adequately acknowledge or recognize the safety significance of important to safety items in the plant. See Conran, ff. Tr. 20,401, at 30; Tr. 20,464-65 (Conran). To the contrary, that testimony affirms that LILCo's understanding of safety goes well beyond the mere performance of the critical safety functions of Appendix A to Part 100. It also represents an affirmation of LILCo's belief that it is in compliance with GDC-1 even under a broader construction of the term "important to safety" than it used in designing and constructing the plant.<sup>10/</sup>

\*7B:141I. A listing of non-safety-related plant items which are "important to safety" is not necessary to demonstrate an understanding of what is minimally required for safety nor would it demonstrate such an understanding. Mattson et al., ff. Tr. 20,810, at 11.

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<sup>10/</sup> During this testimony, the Board expressed its concern that the questions being posed resulted in a tautology which would not assist the record. See Tr. 5434-36, 5442-45, 20,472 (Brenner, J.) The same point was later made by Mr. Conran. Tr. 20,457-59, 20,489-90, 20,613 (Conran). While it is clear that the exchange at Tr. 5425-5449 does not directly assist in the resolution of the ultimate issues involved in this contention, it is useful nevertheless. As stated above, the testimony affirms that LILCo's understanding of safety goes well beyond the mere performance of the critical safety functions of Appendix A to Part 100. Moreover, it represents a sworn statement that LILCo believes it meets the requirements of GDC-1 even under a broader construction of "important to safety" than LILCo has used. This is significant in that it negates the possible inference that LILCo has used a narrower construction of "important to safety" because the narrower construction permitted LILCo to do something which GDC-1 would not permit if it applied to a broader set of structures, systems and components. See also Tr.21,084-85 (Dawe).

\*7B:141J. Mr. Conran himself admitted that he never thought the construction of a list was really necessary to understand the concept of "important to safety." Rather, others had suggested to him that it might be helpful. Tr. 20,660, 20,669-70 (Conran).

\*7B:141K. Mr. Conran also suggested that the give and take of meetings and discussions between the Staff and the utility could also enable one to determine whether there was really a mutual understanding as to what is required for safety. Tr. 20,477, 20,662 (Conran). Precisely such discussions have been conducted by LILCo and the Staff and on the evidentiary record of this proceeding. The Staff is satisfied with LILCo's demonstrated understanding on that basis (see Mattson et al., ff. Tr. 20,810, at 10-13). The Board is satisfied as well.

\*7B:141L. A list of non-safety-related items that are important to safety could probably be generated from a program such as the preventive maintenance program for the plant. Tr. 20,843 (Vollmer). Indeed, LILCo management testified that it has such a list. Tr. 21,134 (Pollock).

\*7B:141M. The Staff does not believe that review of an "important to safety" list is a way to improve safety. Tr. 20,840 (Mattson). Over-emphasis on a list can cause one to fail to do more than rely on that list. If the list is not adequate for whatever reason, serious problems may result. The recent Salem breaker failure event is an example of such a situation. Tr. 20,843 (Mattson). What is important is not the list but the system or process for identifying the important attributes of a structure, system or component and the mechanism for assuring that those attributes are preserved through the life of the plant. Tr. 20,840-41 (Vollmer); Tr. 20,844 (Mattson).

\*7B:141N. The generation of a list and the review of it is not a particularly important ingredient in the regulatory process, at least not sufficiently important to justify allocating resources to it. Tr. 20,843 (Vollmer).

\*7B:141O. Contrary to Mr. Conran's suggestion, it is not necessary to expand the audit review conducted by the Staff and to look at more examples to determine whether LILCo has properly treated structures, systems and components important to safety. Tr. 20,860-61 (Mattson).

\*7B:141P. The Staff's review in accordance with the Standard Review Plan has turned up no evidence that a substantive difference exists between LILCo and the Staff on the treatment to be accorded equipment important to safety. Mattson et al., ff. Tr. 20,810, at 10.

\*7B:141Q. While Mr. Conran has reservations about the effectiveness of the review process for Shoreham because of the conceptual difference he perceives, the question whether Shoreham meets GDC-1 for the past is a matter Mr. Conran preferred to leave to the expert reviewers. Tr. 20,448, 20,431-32 (Conran). Mr. Conran freely acknowledged that the expert reviewers who had conducted the Shoreham review had not changed their position as to the adequacy of the review. Tr. 20,448-50; 20,430; 20,481; 20,524 (Conran). He suggested that the way to evaluate compliance with GDC-1 is to look at examples with the appropriate experts and come to a conclusion as to the appropriateness of what has been done. Tr. 20,500 (Conran). That is precisely what this Board has done.

\*7B:141R. Mr. Conran's use of the qualifier "perhaps" in his affidavit discussion of the backstop provided by the existence and use of regulatory guidance documents meant only that he is unable to verify the

operation of this backstop at Shoreham. Tr. 20,430 (Conran). The Staff's expert reviewers have done so and maintain that Shoreham's compliance is satisfactory. Tr. 20,430 (Conran). Insofar as he has knowledge of the quality standards and quality assurance applied by LILCo to non-safety-related plant items, Mr. Conran knows of no examples where LILCo has not applied proper quality standards and quality assurance measures or has deviated from regulatory requirements. Tr. 20,436, 20,509, 20,523, 20,526 (Conran); but see Tr. 20,706 (Conran) (concern that systems interaction studies by LILCo may be affected by different understanding of "important to safety) and Finding 7B:191H, infra; see also Finding 7B:191U, infra.

\*7B:141S. Mr. Conran believes that the FSAR commitments provided by LILCo are a tautology since LILCo only promises to accord to non-safety-related items in the future the safety significance accorded them in the past. Tr. 20,617 (Conran). The County's witnesses agree. Goldsmith et al., ff. Tr. 20,905, at 26-28, 38. This position ignores the lengthy review and the ample evidentiary record compiled in this proceeding which demonstrate the adequacy of what has been done at Shoreham in the design and construction phase. See Findings 7B:20-39, 51-130, 131-135. When taken in the context of this review and hearing process, the FSAR commitment has substantial meaning and content and is not tautological.

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));

7R: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 non-safety-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., 0.00 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. Intervenor's 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."<sup>11/</sup> 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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<sup>11/</sup> 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 Intervenors 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-scam). 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; Mattson et al., ff. Tr. 20,810, at 3, 4, 8, 14.

\*\*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; Mattson et al., ff. Tr. 20,810, at 4, 5, 14. Its object is to develop and evaluate specific methods to see if there are interactions which may have gone undetected and to see if there is a need to revise present requirements. Tr. 20,830 (Thadani).

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's program plan for its Indian Point-3 study has been approved and endorsed for performance by both the Staff and the ACRS. The actual study effort got underway in April 1982. The Staff expects to receive the results of the PASNY methodology study in August 1983. Mattson, et al., ff. Tr. 20,810, at 7.

\*7B:182. The Staff has considered several alternatives in applying available candidate methods for systems interaction analyses. Consideration was given to using the activities which include: 1) the Power Authority of the State of New York (PASNY) study of Indian Point Unit 3, 2) the Pacific Gas and Electric Co. study of Diablo Canyon, 3) pilot studies on a limited number of light water reactors to test the candidate methodologies, and 4) the Consumers Power Co. program on Midland 2. Alternative consideration was given to an efficient integration of the proposed Systematic Evaluation Program (SEP) Phase III and the National Reliability Evaluation Program (NREP), together

with the systems interaction methodology demonstration.<sup>12/</sup> Finally, consideration has been given to applying the Staff's candidate methods to Indian Point Unit 3, to provide a comparison with the PASNY method of analysis.<sup>13/</sup> Mattson et al., ff. Tr. 20,810, at 6.

7B:183. The alternative of comparing the Staff's methods with the PASNY method in a study of Indian Point Unit 3 is the preferred one at the present time because it will allow the most efficient use of resources for a comparison that is less complicated by plant-wide variations. Mattson et al., ff. Tr. 20,810, at 6. The Staff has secured the cooperation of PASNY to use the Indian Point-3 facility for a demonstration of comparative analyses. The Staff effort to implement

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<sup>12/</sup> 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 commercial 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.

<sup>13/</sup> The Staff originally planned to select four near term operating license (NTOL) applications as pilot plants to apply the methods being studied. Cost consideration led Staff to coordinate the effort under NREP, but later delays in NREP led to reconsideration of the approach and to placing emphasis on Indian Point. See 7B:183, infra. These steps represent progress in the A-17 program in the sense that the Staff is going through the necessary steps before the issue can be resolved. Tr. 20,866-67 (Thadani).

the demonstration analyses on Indian Point-3 is now being prepared. Speis et al., ff. Tr. 6357, at 39-40; Mattson et al., ff. Tr. 20,810, at 6, 8. The three methodologies to be compared are digraph matrix analysis, fault tree interactive failure modes and effects analysis and the method developed by PASNY which involves the use of dependency tables. Tr. 21,013-014 (Minor). The schedule for the comparative analysis at Indian Point includes initiation of the Staff Methodology Comparison Study in April 1983 and receipt of results of the Staff Study in July 1984. Mattson et al., ff. Tr. 20,810, at 7.<sup>14/</sup>

\*\*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. Another part of the reason is the difficulty in merging the NREP program and the systems interaction program. Tr. 7151 (Conran). During the time Mr. Conran was responding to Contention 7B, the established schedule for the systems interaction program was as given in Enclosure 3 in a memorandum from W. J. Dircks, Executive Director for Operations, to P. Shewmon, Chairman of the ACRS, dated February 12, 1982. That schedule called for selection of the plants for demonstration analyses by February 1982, and a Staff decision on whether to issue a plant-specific systems

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<sup>14/</sup> Suffolk County's experts argued that the data obtained from the Indian Point study will not be useful for Shoreham since the former plant is a pressurized water reactor and the Staff has no plans to test the methods on boiling water reactors. Goldsmith, ff. Tr. 20,903, at 15-16. They conceded, however, the three methods being studied at Indian Point "may be applicable to BWR studies". Id. at 15. Use of the three methods at Indian Point, which are used for systems analysis in contexts other than nuclear power plant evaluation, will provide comparative data for the evaluation of the relative value of these methodologies. Tr. 20,015-16 (Minor).

interaction requirement by January 1984. Recognizing the large costs involved in demonstration analyses, the selection of the plants was delayed to consider the additional alternative of integrating the systems interaction demonstrations with the proposed SEP Phase III/NREP effort. This delay in the established schedule was described as a potential delay within the Dircks to Shewmon memorandum of February 12, 1982. In November 1982, the Commission decision on a proposal to combine and implement SEP III/NREP was deferred until after a review on the safety benefits of the SEP Phase II was completed. The decision by the Commission on SEP Phase III is now scheduled for Summer 1983. Mattson, et al., ff. Tr. 20,810, at 7-8.

\*\*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; Mattson et al., ff. Tr. 20,810, at 3; Tr. 20,815 (Thadani); Staff Ex. 2A, at B-9, B-10.

\*7B:186. The Shoreham application was evaluated against licensing requirements that were founded on the principle of defense-in-depth. The Shoreham design was reviewed against the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (originally issued as NUREG 75/087 in December 1977, and reissued as NUREG-0800 in July 1981 with the addition of the TMI-2 accident-related requirements), which requires interdisciplinary review of equipment and addresses different types of potential systems interactions. Use of the Standard Review Plan in the review process results in safety requirements such as physical separation and independence of redundant safety systems and protection against hazards such as high energy line ruptures (Section 3.6.1 of the SRP), missiles (Sections 3.5.1 and 3.5.2), high winds (Section 3.3), flooding (Sections 3.4 and 3.6), seismic events (Section 3.2.1, 3.4 and 3.9.2) and fires (Section 9.5.1). Mattson et al., ff. Tr. 20, 810, at 4-5; see Tr. 6659, 6779, 20,831 (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); Mattson et al., ff. Tr. 20,801, at 5; but see Finding 7B:191C, infra.

\*\*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; Mattson et al., ff. Tr. 20,810, at 3-4, 5, 8, 13; Tr. 20,816 (Thadani); Staff Ex. 2A, at B-9 through B-11; ~~Tr. 7141-(GERRAN)~~; 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 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; Tr. 20,816-17 (Thadani).

\*\*7B:189. The same conclusion was expressed ~~earlier-this-year~~ in 1982 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; Tr. 20,816-17 (Thadani). 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-(GERRAN)~~; Tr. 6779 (Thadani); Staff Ex. 2A, at B-11; Mattson et al., ff. Tr. 20,810, at 3.

\*\*7B:190. Contrary to Intervenor's 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 (Thadani); see Finding 7B:188, supra.

\*\*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).

\*7B:191A. On January 25, 1983, the Staff informed the Board and parties by letter that one of the Staff's witnesses who testified on this contention, James H. Conran, had informed Staff counsel that he sought to modify certain of his testimony since he no longer supported some aspects of that testimony. The Board was further informed that Mr. Conran was preparing a written statement of his present views on the matters discussed in his testimony. Mr. Conran's affidavit was filed on February 8, 1983. The Conran affidavit was received in evidence during reopened proceedings on April 5, 1983. Tr. 20,401. One of the subjects of this affidavit was the Staff's efforts in the systems interaction area and Mr. Conran's present view as to the inadequacy of those efforts.

\*7B:191B. The Conran affidavit (ff. Tr. 20,401) expresses Mr. Conran's present view that the Staff's program for resolution of A-17 has declined to such an extent that he no longer believes that it is currently adequate to provide a basis for the "justification for



operation" conclusion required under North Anna. Conran, ff. Tr. 20,401 at 2, 10-12; Tr. 20,696 (Conran). He believes that reasonable progress is a necessary element of the North Anna finding. Tr. 20,698-99 (Conran). Without progress toward resolution of A-17, Mr. Conran could not conclude that there was reasonable assurance that Shoreham could be operated with no undue risk to public health and safety. Tr. 20,718, 20,781 (Conran).

\*7B:191C. Mr. Conran's affidavit discusses at some length the bases for his change in position in terms of the history of the Staff's systems interaction efforts, events of recent months and his estimation of the schedule on which action toward resolution can proceed. Conran, ff. Tr. 20,401, at 3-12. He makes a particular point that the occurrence of unanticipated interactions, if permitted to happen often enough for long enough can make the likelihood of a serious accident unacceptably high. Conran, ff. Tr. 20,401, at 6; see Tr. 20,788 (Conran). He concludes that a requirement should be imposed by the Staff at this time for limited systems interaction analyses by licensees and operating license applicants. Id., at 12. The testimony of Suffolk County's witnesses is in fundamental agreement with Mr. Conran on these matters. Goldsmith, et al., ff. Tr. 20,903, at 1-22.

\*7B:191D. Mr. Conran's affidavit was precipitated by the cumulative effect of: (1) the loss of the separate NTOL pilot plant option; (2) the delay in availability of data from systems interaction studies undertaken at Diablo Canyon and Indian Point; and (3) the lack of any serious indication by Staff management that some other measures would be taken to offset these losses, such as the initiation of limited

system interaction studies by all NTOL applicants. Conran, ff. Tr. 20,401, at 10-12, 19-22; Tr. 20,716-17 (Conran).

\*7B:191E. Mr. Conran's concern was with the weight that the Board might give to the adequacy of the Staff's program in arriving at a finding that Shoreham could be licensed for operation despite the pendency of A-17. Tr. 20,685, 20,785 (Conran).

\*7B:191F. Mr. Conran would not go so far as to say that, because of the concern about A-17, no plant should be licensed and existing plants should be shut down. Tr. 20,688 (Conran).

\*7B:191G. Mr. Conran acknowledged that LILCo had given "rather extensive consideration" to potential systems interactions at Shoreham. Tr. 20,686, 20,782-84 (Conran).

\*7B:191H. In Mr. Conran's view, this consideration of systems interaction by LILCo specifically for Shoreham would provide an adequate basis for licensing Shoreham under North Anna if the safety classification issue of LILCo's refusal to use the term "important to safety" as the Staff uses it could be resolved. Tr. 20,687, 20,782-84, 20,787 (Conran). This residual concern was that LILCo, because of its different understanding of the importance of non-safety-related items, might have a different judgment as to the safety significance of interactions identified in its various systems interaction studies. Tr. 20,705 (Conran). This was referred to by Mr. Conran as a "synergistic" concern. Conran, ff. Tr. 20,401, at 26-27; Tr. 20,686 (Conran).

\*7B:191I. In the one Shoreham systems interaction analysis that Mr. Conran had examined, the high energy line break study, LILCo had addressed non-safety-related as well as safety-related equipment. Tr. 20,701-02 (Conran).

\*7B:191J. Mr. Conran also questioned the Staff's present schedule, and suggested that resolution of A-17 "is still 2-3 years off without significant re-ordering of priorities and reconstitution of the . . . program . . ." Conran, ff. Tr. 20,401, at 10.

\*7B:191K. The Staff's position as reflected in the earlier Staff testimony on A-17 and systems interaction has not changed as a result of Mr. Conran's affidavit or for any other reason. Mattson et al., ff. Tr. 20,801, at 3. That position includes: (1) that the Staff's current licensing requirements provide reasonable assurance of no undue risk to public health and safety from potential adverse systems interactions; (2) the A-17 task is confirmatory in nature; (3) the Staff's A-17 program is progressing toward resolution; (4) Shoreham may be licensed for operation despite the pendency of A-17; and (5) no plant specific systems interaction analyses (other than those now required by regulation or Staff practice) are or should be required until completion of the Staff's program determines whether they are necessary and justified. Mattson et al., ff. Tr. 20,810, at 3-4.

\*7B:191L. The judgment made by the Staff for Shoreham that it can operate safely pending resolution of A-17 is not tied to the schedule

for resolution of A-17. Rather the basis for this judgment is provided by the present reviews that are conducted in accordance with the guidance given in the Standard Review Plan. It is based on what has been done rather than what will be done in the future. Tr. 20,878-79 (Coffman). Mr. Coffman is the Staff member responsible for management of the Staff's program for resolution of A-17. Mattson et al., ff. Tr. 20,810, at 3.

\*7B:191M. The schedule for resolution of A-17 is not critical in reaching reasonable assurance for Shoreham. The actions taken by the Commission in terms of new requirements have most likely taken care of many systems interaction issues. Tr. 20,867-68 (Thadani). Standard Review Plan requirements have already increased in areas where one might expect interactions of some significance; Staff management testified that current criteria would identify most, if not all, of the significant interactions related to safety. Tr. 20,862-63 (Thadani); see Tr. 20,917 (Goldsmith). Indeed, the Staff is not aware of any major interactions that are not already considered under the regulations. Tr. 20,830 (Thadani); but see Tr. 20,788 (Conran) (concern about operating experience).

\*7B:191N. The nature of the particular issue should be factored into the North Anna determination. With formal unresolved safety issues as broad as A-17 (which says, in essence, "go look" throughout plants for interactions), changes in requirements and equipment are made over time

which address the underlying concern. As a result of this process of the improvement of plants and requirements over time, the safety issue is addressed incrementally; it is easier for such issues to make the North Anna finding that the plant can operate without undue risk to the public health and safety because intervening changes have ameliorated the problem. Tr. 20,863-65 (Mattson); Tr. 20,879 (Thadani).

\*7B:1910. Moreover, there has been progress in the Staff's program for resolving A-17. Mattson et al., ff. Tr. 20,810, at 4. By October, 1984, the Staff expects to complete a review of various systems interaction studies, assess the efficiency of the methodologies used in these studies, and to make a decision on the need for any requirement for plant-specific systems interaction analyses. This expectation is based on the following schedule:

- 1) Initiate Staff Methodology Comparison Study on Indian Point Unit 3 in April 1983;
- 2) Receive PASNY Methodology results in August 1983;
- 3) Receive results of Staff Study on Indian Point Unit 3 in July 1984;
- 4) Develop Safety Significance of Identified Interactions in July 1984;
- 5) Develop Basis for new licensing requirements, if any, as a result of the A-17 program in October 1984. Mattson et al., ff. Tr. 20,810, at 7.

\*7B:191P. Even with respect to events over the last six months, which was the time frame focused on by Mr. Conran's affidavit, Mr. Thadani expressed his judgment that there has been sufficient progress during that time period to indicate that the Staff is moving toward resolution of A-17. Tr. 20,814 (Thadani).

\*7B:191Q. Progress in the program to date has provided no indication that present review procedures and criteria do not provide reasonable assurance that the effects of potential systems interactions on plant safety will be within the effects on plant safety previously evaluated (i.e., within the design basis envelope). Mattson, et al., ff. Tr. 20,801, at 5.

\*7B:191R. The Staff judgment is that A-17 will be resolved within a time frame such that there will not be undue risk to the public from operation of Shoreham in the interim. Tr. 20,877 (Thadani); see Tr. 20,913 (Goldsmith) (low probability of systems interaction events); but see Tr. 20,788 (Conran) (concern about operating experience). Even if five years elapsed without resolution, the judgment of no undue risk for Shoreham would be valid. Tr. 20,878 (Thadani).

\*7B:191S. In addition to the adequacy of existing regulatory requirements to support the required North Anna finding for Shoreham, LILCo has gone beyond Staff requirements for systems interaction analysis in several areas, including the probabilistic risk assessment done for Shoreham. Tr. 20,869 (Thadani). The Shoreham PRA, even though it did not include external initiators, will identify some of the major interactions. Tr. 20,869 (Thadani); Tr. 20,975-76 (Goldsmith).

\*7B:191T. The systems interaction analyses done by LILCo for Shoreham are likely to be sufficient even after the Staff is done with its A-17 program. Tr. 20,877 (Thadani).

\*7B:191U. The Staff does not share the "synergistic" concern expressed by Mr. Conran relating the safety classification issue to the systems interaction issue. Tr. 20,828-29 (Thadani). Systems interaction studies are conducted independent of classification. Tr. 20,828 (Thadani); Tr. 20,927 (Goldsmith).

\*7B:191V. Additional plant-specific systems interaction studies are not necessary to provide reasonable assurance of public health and safety as a predicate to licensing Shoreham. Systems interaction analyses are very expensive (even limited ones would cost over \$500,000 each). The Staff's program to resolve the A-17 issue is now at the stage where the next step is an application of the known and documented methods. The application of these methods will provide a basis to answer the questions of the efficiency of a specific methodology: 1) to discovery unforeseen intersystems dependencies within the plant, 2) to rank-order such systems dependencies that are safety significant, and

3) to establish the resource efficiency from a safety-significance basis. Mattson et al., ff. Tr. 20,801, at 5-6, 14.

\*7B:191W. No plant-specific systems interaction analyses (other than those now required by regulation or by Staff practice) are or should be required until completion of the Staff's program determines whether they are necessary and justified. Mattson et al., ff. Tr. 20,810, at 4; Tr. 20,831 (Thadani).

\*7B:191X. Shoreham may be licensed for operation despite the pendency of unresolved safety issue A-17. Mattson et al., ff. Tr. 20,810, at 4; Tr. 20,831-32 (Mattson).

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 opera-

tional 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<sup>15/</sup> 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).

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<sup>15/</sup> 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).<sup>16/</sup> 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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<sup>16/</sup> 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 reproducibility 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; Tr. 20,870 (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; Tr. 20,870 (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 takes 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. Tr. 6627-28 (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 unwieldy 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. Conclusion on reliance on the Shoreham draft PRA

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  
(Shoreham Nuclear Power Station, ) (OL)  
Unit 1) )

CERTIFICATE OF SERVICE

I hereby certify that copies of revised "NRC STAFF PROPOSED OPINION, FINDINGS OF FACT, AND CONCLUSIONS OF LAW IN THE FORM OF A PARTIAL INITIAL DECISION," dated May 16, 1983, in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, or, as indicated by a double asterisk, by hand delivery, this 16th day of May, 1983:

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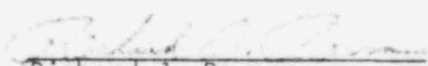
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