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.)

PREPARED DIRECT TESTIMONY OF
MARC W. GOLDSMITH, SUSAN J. HARWOOD, RICHARD B. HUBBARD
AND GREGORY C. MINOR
ON BEHALF OF SUFFOLK COUNTY AND THE
SHOREHAM OPPONENTS COALITION

REGARDING

CONTENTION 7B

April 13, 1982

SUMMARY OUTLINE OF CONTENTION 7B TESTIMONY */

Suffolk County and SOC contend that LILCO has not adequately classified or analyzed Shoreham structures, systems and components (SS&C) which are important to safety, thus violating the NRC's General Design Criteria.

Evidence of deficiencies in LILCO's methodology and implementation of classification is provided in this testimony by means of a review of the following: (a) LILCO's identification of safety-related SS&Cs (Q-List) contained in FSAR Table 3.2.1-1; (b) the components called for in the Shoreham Emergency Operating Procedures compared with those called for in the FSAR Chapter 15 analysis of Design Basis Accidents, and the classification of such components in the Q-List; and (c) examples of systems that fail to satisfy applicable classification criteria.

The methodology used by LILCO to analyze Shoreham SS&Cs to determine their safety importance and thus their quality requirements is very limited, both in approach and effectiveness. LILCO has failed to utilize improved techniques that are now available for safety classification, such as, Probabalistic Risk Assessment (PRA), failure modes and effects analyses, systems interaction analyses, and dependency analyses. As a result, many SS&Cs which are safety-related or important to safety have not been properly recognized, classified, or treated in the Shoreham design.

A further result of LILCO's use of incomplete methodology for analyzing SS&Cs important to safety is that systems

*/ ASLB Memorandum and Order, March 15, 1982, p. 29.

interactions which may have a significant impact on plant safety have gone undetected. For example, unsafe conditions can result from systems interaction involving the water level measurement system at Shoreham. Absent improvements in LILCO's classification and analyses of SS&Cs as discussed in this testimony, there can be no finding that Shoreham complies with the General Design Criteria necessary to ensure safe operation of the Plant.

Exhibits */

1. Memorandum, dated November 20, 1981, from Harold R. Denton to all NRR Personnel re: Standard Definition for Commonly-Used Safety Classification Terms.

2. FSAR Section 3.2, including Table 3.2.1-1 (reflecting revisions set forth in April 8, 1982 letter from T.S. Ellis to Larry Lanpher, see Exhibit 3 below), Table 3.2.1-2, Table 3.2.1-3, and Figure 3.2.2-1.**/

3. Letter, dated April 8, 1982, from T.S. Ellis (Hunton & Williams) to Larry Lanpher.

4. Table of Equipment Relied upon by LILCO for DBA Mitigation Per FSAR Chapter 15 and Shoreham Emergency Operating Procedures.

5. Board Notification 82-08, "Errors in BWR Vessel Water Level Indication," dated February 9, 1982.

6. Weekly Information Report, Week Ending January 22, 1982, dated January 27, 1982--Cover Sheet and Enclosure K.

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*/ ASLB Memorandum and Order, March 15, 1982, p. 29.

**/ Although the FSAR constitutes part of the record in this case, Table 3.2.1-1 has not been formally amended to reflect the changes made on April 8, 1982. Accordingly, for the convenience of the Board and the parties, a copy of the Table with the changes noted thereon, is attached as an Exhibit.

TABLE OF CONTENTS

Page

TABLE OF CONTENTS (Cont'nued)

TABLE OF CONTENTS (Continued)

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I. INTRODUCTION

This testimony was prepared by Marc W. Goldsmith, Susan J. Harwood, Richard B. Hubbard, and Gregory C. Minor.1/ A statement of our qualifications and experience has been separately provided to this Board.

1/ The primary/secondary authors of each section of this testimony are as follows:

Sections I, II, III, and VII -- R.B. Hubbard/G.C. Minor
Sections IV, VIII and IX -- G.C. Minor/R.B. Hubbard
Section V -- S.G. Harwood/M.W. Goldsmith
Section VI -- M.W. Goldsmith/G.C. Minor

G.C. Minor was responsible for overall coordination.

The testimony which follows is organized into six main sections (III through VIII) and a conclusion section (IX). Background is covered in Section III, Inconsistencies in Classification in Section IV, Incompleteness of Classification in Section V, and Examples of Systems Interaction and System Classification Deficiencies in Section VI. Sections VII and VIII cover Generic Issues of Systems Interaction, and Alternate Methodologies, respectively.

II. STATEMENT OF CONTENTION

The purpose of this testimony is to address Contention 7B as restated by the Board:[2/]

> 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, 22, 23, 24, 29, 35, 37 apply, and thus it is not possible to demonstrate compliance with these criteria.

2/ ASLB Memorandum and Order, March 15, 1982, p. 12.

III. BACKGROUND

III.A. SAFETY CLASSIFICATIONS UNDER THE NRC'S REGULATIONS

The NRC's General Design Criteria ("GDC") set forth minimum design criteria requirements for those structures, systems and components ("SS&C") of nuclear power plants which are "important to safety." An Applicant attempting to satisfy the requirements of the GDC must first define the SS&Cs important to safety because only by such definition can the Applicant determine those SS&Cs which must comply with the GDC.

The proper initial determination/definition of SS&Cs important to safety is crucial for a further reason. Under GDC 1, a quality assurance ("QA") program must be established and implemented for SS&Cs important to safety.[3] Such a QA program must cover such matters as design, fabrication, erection and testing of SS&Cs important to safety and appropriate records must be kept. If the SS&Cs to which Appendix A and Appendix B apply are not identified and defined early in the project, QA compliance from the outset of the project is not possible.

3/ The NRC in its recent Regulatory Agenda stated that it intends to issue revisions to GDC 1 to clarify, as originally intended, that the QA requirements of Appendix B to Part 50 would apply to all SS&Cs to which Appendix A applies. *See* 46 *Fed. Reg.* 53,618 (1981).

The NRC has broadly defined SS&Cs "important to safety" as those "systems, structures and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."4/ Harold R. Denton, the Director of the NRC's Office of Nuclear Reactor Regulation, has stated that the foregoing definition:

> Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in an important way to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation).5/

The terms "safety-related" and "safety-grade" are frequently utilized by applicants in classification of SS&Cs. For example, LILCO utilizes the term safety-related in FSAR Section 3.2 wherein it describes its classification system.

Safety-related, when utilized in safety analyses and, indeed, as utilized by LILCO in the FSAR, refers to a more narrow category of SS&Cs than those broad classes of SS&Cs defined in the GDC and by Mr. Denton as "important to safety." Thus, safety-related (or safety-grade, which is synonomous with safety-

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4/ 10 C.F.R. Part 50, Appendix A, Introduction, ¶ 1.

5/ Memorandum from H. R. Denton to all NRR Personnel, November 20, 1981, Subject: Standard Definitions for Commonly-Used Safety Classification Terms, attached as Exhibit 1 hereto.

related), is defined with reference to 10 C.F.R. Part 100, Appendix A, as

> Those structures, systems, or components designed to remain functional for the SSE (also termed 'safety features') necessary to assure 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 this part.6/

Safety-related or safety-grade SS&Cs constitute a subset of the larger category of SS&Cs which are defined as "important to safety" under the GDC.7/

III.B: POWER REACTOR ACCIDENT ANALYSES ARE TRADITIONALLY BASED ON THE SINGLE FAILURE CRITERION

Traditionally, the NRC has approached accident and safety analysis on a system-by-system basis, using the "single failure

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6/ Exhibit 1, p. 3 (emphasis in original); see 10 C.F.R. Part 100, App. A, § III(c).

7/ Exhibit 1, p. 3.

criterion." See 10 C.F.R. Part 50, Appendix A, Criterion 21, for example. Thus, the single failure criterion[8/] requires that a nuclear power plant structure, system or component important to safety be capable of performing its safety function in the presence of: (a) any single detectable failure within that structure, system or component (or its essential auxiliary supporting systems or another safety system) concurrent with all failures resulting from the single failure; (b) all undetectable failures; and (c) all failures that caused or were caused by the accident that require operation of the safety system. In reality, the single failure criterion is

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8/ A "single failure" is defined in 10 C.F.R. Part 50, Appendix A, Definitions and Explanations, as:

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

Also, see NRC Regulatory Guide 1.53 entitled, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

a double failure criterion: it requires that the design must be able to bring the plant to a safe shutdown despite the occurrence of an accident plus the failure of any one additional safety structure, system or component.

Performing an evaluation to determine whether SS&Cs important to safety meet the single failure criterion involves the following steps:

a. Identify SS&Cs that are not safety-related or important to safety, *e.g.*, not seismically and environmentally qualified in accordance with GDC 2 and 4, not physically and electrically separated as required by GDC 17 and 22, or not protected against fire as required by GDC 3.

b. Assume that each such SS&C fails if its failure adversely affects the structure, system or component being analysed, or assume it operates if its operation adversely affects the item being analysed.

c. Assume that all failures which can cause or can be caused by the accident requiring the structure, system, or component being analysed to operate have occurred.

d. Assume that any other single failure has occurred and then determine whether the structure, system or component being evaluated can still perform the required safety function.

LILCO applied the single failure criterion in its Chapter 15 safety analyses which are summarized in the FSAR.[9/]

Applicants are not required to limit their accident analyses to those encompassed by the single failure criterion. They are free to assess multiple failure events since the GDC (wherein the single failure criterion is defined) set forth minimum requirements for the principal design criteria for water-cooled nuclear power plants.[10/] In this regard, the NRC has specifically noted that the GDC are not complete but that the omission of matters from the GDC does not relieve an Applicant from considering matters, important to safe design, which may not be specifically addressed in particular GDC.[11/]

The limitations of the single failure criterion approach to safety evaluations have been widely recognized. For

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9/ Deposition of Robare, et al., March 31, 1982, p. 108 (Dawe) (hereafter cited as "Deposition at _____ (name of speaker)."

10/ 10 C.F.R. Part 50, Appendix A, Introduction.

11/ See GDC 34, 35, 38, 41, and 44, as referenced in 10 C.F.R. Part 50, Appendix A, Introduction.

example, in an August 12, 1980 letter, the NRC's Advisory Committee on Reactor Safeguards informed then NRC Chairman Ahearne that:

> Many current safety evaluations use the single failure criterion as a measure of reliability. Its inadequacy is widely recognized. It should be replaced, where feasible, with criteria that consider the possible contributions to risk of multiple failures. (emphasis added).

The single failure criterion, by definition, ignores the risks resulting from multiple failure accidents.

The application of the single failure criterion is most vividly demonstrated in the Applicant's presentation and the NRC's review of so-called "design basis accidents." A recently published Notice of Proposed Rulemaking for a range of degraded core cooling events describes the design basis approach.12/ Thus, in a Safety Analysis Report the Applicant is required to determine margins of safety for both normal and abnormal operations and to determine the adequacy of structures, systems, and components provided for prevention of accidents and the mitigation of the consequences of accidents. To assist the Applicant in complying with this requirement, the NRC has

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12/ See Advance Notice of Proposed Rulemaking "Consideration of Degraded or Melted Cores In Safety Regulation," Fed. Reg., Vol. 45, No. 193, October 2, 1980, pp. 65474-65477.

published Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, which describes the information to be provided in the Safety Analysis Report.

In particular, Section 15 of Regulatory Guide 1.70 provides guidance to an Applicant concerning "design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 C.F.R. Part 100 criteria," the criteria which define those SS&Cs which are classified as safety-related.[13/] Operating events corresponding to design basis assumptions are termed "design basis accidents," and satisfactory analytical conclusions concerning these specified and predefined events allow a judgment (in the view of the Staff) that the facility can be operated without undue risk to the health and safety of the public.

It should be emphasized that the Regulatory Guide 1.70 events are analyzed primarily for the purpose of establishing the adequacy of engineered safety features. Such features are

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13/ Regulatory Guide 1.70 explains that these design basis assumptions can, for the most part, be derived from Regulatory Guides that deal with radiological releases, and suggests use of Regulatory Guides 1.3 and 1.4, "Assumptions Used for Evaluation of the Potential Radiological Consequences of a Loss-of-Coolant Accident." Regulatory Guide 1.70 further states that "This analysis should be referred to as the 'design basis analysis.'"

those SS&Cs which are designed into a plant to mitigate the consequences of postulated design basis accidents, and which supplement other plant features designed to meet performance specifications for normal operations and anticipated abnormal conditions. However, in the Safety Analysis Report the Applicant is not required to analyze accidents more severe than the design basis accidents. This approach is based on the generic assumption that such accidents are of sufficiently low probability that mitigation of their consequences is not necessary for public safety.

A further weakness of the design basis approach is that it fails to consider in any systematic fashion the potential for adverse systems interactions between safety-related SS&Cs, or between safety-related SS&Cs and SS&Cs which are left unclassified. A systems interaction is an event or sequence of events which causes two or more components to fail to perform their functions, or to perform the functions in a degraded manner, thus increasing the likelihood of an undesired event. The GDC are explicit in requiring Applicants to consider systems interactions in safety analyses and design. Thus, Applicants are directed to give:

Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems.14/

Notwithstanding the directive of the GDC, Applicants and the Staff have failed to perform complete, systematic analyses of the potential for adverse systems interactions.15/

III.C: MULTIPLE-FAILURE ACCIDENTS HAVE OCCURRED AT OPERATING REACTORS

A most serious shortcoming of the design basis approach is its failure to consider multiple failure accidents. The accident at Three Mile Island, Unit 2 (TMI-2), resulted in core damage more severe than that considered in current design basis events, resulting both from multiple failures and adverse systems interactions. TMI-2 has thus demonstrated the need to re-examine the adequacy of the historical design basis approach to analyzing reactor plant design and plant accidents. For example, the January, 1980, Rogovin Report, Three Mile Island: A Report to the Commissioners and to the Public, states on page 150 that:

> [W]e have come far beyond the point at which the existing, stylized design basis accident review approach is sufficient. The process is not good

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14/ 10 C.F.R. Part 50, Appendix A, Introduction.

15/ NUREG/CR-1321, Phase I-Systems Interaction Methodology Applications Program, Nuclear Regulatory Commission, Washington, D.C., April 1980, p. 151.

enough to pinpoint many important design weaknesses or to address all the relevant design issues. Some important accidents are outside or are not adequately assessed within the 'design envelope'; key systems are not 'safety related'; and integration of human factors into the design is grossly inadequate.

The need to assess the safety significance of multiple-failure accidents and the SS&Cs relied upon to respond to such accidents, is set forth in the following paragraphs.

The TMI-2 accident revealed major shortcomings in many of the procedures upon which the NRC has based its approach to safety. First, the accident "involved a sequence of events more severe than those included in current design basis events."[16/] Core temperatures exceeded 3500 degrees F.,[17/] or more than 1300 degrees F. above the level for which emergency core cooling systems are designed. The chemical reaction between water and the zirconium fuel cladding generated five to ten times as much potentially explosive hydrogen as is assumed in the design bases for hydrogen control systems.[18/] Although extensive core damage, with cracking, crumbling, and possible

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16/ NUREG-0585, TMI-2 Lessons Learned Task Force - Final Report U.S. Nuclear Regulatory Commission, Washington, D.C., October, 1979, p. 3-1.

17/ Rogovin Report, Vol. II, p. 18.

18/ NUREG-0683, Draft Programmatic Environmental Impact Statement for TMI-2 Decontamination and Disposal of Radioactive Wastes, 1980, U.S. Nuclear Regulatory Commission, Washington, D.C., p. S-1.

melting and fusing of fuel pellets and parts of fuel assemblies such as occurred at TMI-2 had been foreseen as a possible event, it had been excluded from the design basis of TMI-2 and other plants since plant safety features had been provided to prevent it from occurring.[19]

Second, the TMI-2 accident involved a sequence of several multiple-failures which demonstrated the inadequacy of the single failure criterion. Thus, at TMI-2 the combination of closed auxiliary feedwater valves, stuck open pilot-operated relief valves, and misinformation to the operators, allowed the feedwater failure and the partial blowdown to create voids in the primary coolant. The voids, in turn, produced misleading pressurizer level indications. This resulted in the operators' terminating emergency cooling water, which eventually resulted in failure of the fuel. The release of radioactivity was due to the high sump level causing the pump to turn on and pump radioactive waste to the Auxiliary Building. There, it was released to the environment as a result of additional errors. The radioactivity in the atmosphere fed back through the

19/ NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, 1979, U.S. Nuclear Regulatory Commission, p. 16. The transport of radioactive gases and liquid into the auxiliary building (through normal leakage in water make-up and let-down pumps and tanks deliberately run to cool the reactor coolant pump seals) also fell outside the design basis.

control room ventilation system, raising the levels to the point where special breathing apparatus had to be worn by the operators who were trying to control the accident. After the accident, the high radiation levels in the containment and the primary loop have continued to make it very difficult to perform the necessary maintenance functions. In general, the multiple, interrelated failures involving various systems and their interactions (with and without human intervention) had not been foreseen in the safety analyses conducted as part of the TMI-2 licensing process.

Other BWR accidents, such as the Dresden-2 blowdown in June, 1970 and the Browns' Ferry fire in March, 1975, also involved the effects of one system on another. In addition, the NRC reviews of the June 28, 1980, partial failure to scram incident at Browns' Ferry-3 disclosed a potential for unacceptable interactions between the control rod drive system and the non-safety-grade control air system for BWR's. Other examples of multiple-failure accidents are described in a review of Licensee Event Reports occurring between 1976 and 1978, prepared for the ACRS (see Sections D III and D VI of NUREG- 0572).

The NRC's Lessons Learned Task Force, after reviewing the TMI-2 accident scenario, formed the following conclusion regarding the potential for adverse systems interactions and multiple-failure accidents:

> The interactions between non-safety-grade and safety-grade equipment are numerous, varied and complex and <u>have not been systematically evaluated</u>. Even though there is a general requirement that failure of non-safety-grade equipment or structures should not initiate or aggravate an accident, <u>there is no comprehensive and systematic demonstration that this has been accomplished</u> . . . .[20]

Thus, comprehensive analyses of multiple-failure accidents, covering both safety and non-safety systems, under normal, transient, and accident conditions at Shoreham appears both necessary and prudent.

IV. THE SHOREHAM CLASSIFICATION METHODOLOGY DOES NOT COMPLY WITH REGULATORY REQUIREMENTS

LILCO presents its conclusions and discussion regarding classification of SS&Cs in Section 3.2 of the FSAR. The following section describes LILCO's classification approach and demonstrates that it does not comply either with GDC requirements or with the Regulatory Guides upon which it allegedly is based.

IV.A. THE SHOREHAM APPROACH TO CLASSIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS

LILCO has classified and listed particular Shoreham SS&Cs in FSAR Table 3.2.1-1, a copy of which is included in Exhibit 2 attached hereto.[21] The classification results set forth in

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20/ NUREG-0585, p. 3-3 (emphasis supplied).

21/ The County would not normally attach Table 3.2.1-1, or any portion of the FSAR, as an Exhibit since the FSAR consti-

(Footnote continued on next page)

the Table allegedly are based on the guidance in NRC Regulatory Guides 1.26, "Quality Group Classifications" and 1.29, "Seismic Design Classification."22/

Regulatory Guide 1.26 discusses the classification system developed by the NRC Staff for safety-related SS&Cs containing water, steam, or radioactive material in water-cooled nuclear power plants. The system consists of four quality groups, A through D. The Regulatory Guide specifies that all four quality groups are safety-related components. Group A, requiring the most stringent quality standards, is described in Section 50.55a of 10 C.F.R. Part 50. Groups B, C, and D, having lower quality standards respectively, are described in Regulatory Guide 1.26.

LILCO describes its use of Regulatory Guide 1.26 as follows:

(Footnote continued from previous page)

tutes part of the record in this case. However, as is discussed in Section IV.C below, the Table has been changed but not yet formally amended by LILCO. Therefore, a copy of the Table with LILCO's changes noted thereon, is included in Exhibit 2 for the convenience of the Board and the parties.

22/ See LILCO FSAR, p. 3.2-1, which is included in Exhibit 2. LILCO does not specify in the FSAR which revision of the Regulatory Guides it has utilized for classification purposes. However, it appears, at least with respect to Regulatory Guide 1.26, that LILCO is using Revision 1, dated September, 1974. Deposition at 57 (Dawe).

> System quality group classifications have been determined for each component of (1) those applicable fluid systems relied upon to prevent, or mitigate the consequences of, accidents or malfunctions originating within the reactor coolant pressure boundary, or to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (2) <u>other associated safety related systems</u>. Regulatory Guide 1.26, "Quality Group Classifications and Standards," was used in assigning (1) quality group classifications and (2) design and fabrication requirements as shown in Tables 3.2.1-1, 2, and 3.23/

Thus, LILCO states that it used the Regulatory Guide's classification scheme for safety-related components not only for fluid systems, but also for "other associated safety related systems."

Regulatory Guide 1.29 outlines a method acceptable to the NRC Staff for identifying and classifying those features of light water cooled nuclear power plants that should be designed to withstand the effects of the Safe Shutdown Earthquake (SSE) and remain functional. Those SS&Cs that must survive the SSE are designated "Seismic Category I" and must meet pertinent Part 50, Appendix B requirements. Those SS&Cs not classified as Seismic Category I are unclassified. Applying Regulatory Guide 1.29, LILCO defined the Seismic Category I SS&Cs in the FSAR as:

23/ LILCO FSAR, p. 3.2-1 (emphasis supplied), included in Exhibit 2.

those necessary to insure:

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 exposure of 10 C.F.R. 100.24/

This is the same category of SS&Cs which the Staff defines as "safety-related," as discussed in Section III.A of this testimony.

There is some confusion in the FSAR regarding LILCO's classification terminology. In Section 3.1, LILCO states that "[s]tructures, systems and components important to safety are listed in Table 3.2.1-1."25/ The FSAR then continues: "The total quality assurance (QA) program is described in Chapter 17 and is applied to the items contained in this table."26/ In fact, LILCO does not utilize a classification category of SS&Cs "important to safety." Rather, where the GDC refer to SS&Cs important to safety, LILCO interprets the item to be safety-related in accordance with the Part 100, Appendix A definition.27/ Further, the full Part 50, Appendix B QA

24/ LILCO FSAR, p. 3.2-1, included in Exhibit 2.

25/ LILCO FSAR, p. 3.1-2.

26/ Ibid.

27/ Deposition, at 60, 71 (Dawe).

requirements are not applied to all items in Table 3.2.1-1, as implied by LILCO's statement. Rather, when reviewing Table 3.2.1-1, one must understand that the full QA requirements of Appendix B are applied only to those SS&Cs marked "LILCO Quality Assurance Category I" or "Seismic Category I."[28/]

IV.B. THE SHOREHAM CLASSIFICATION APPROACH DOES NOT COMPLY WITH PART 50, APPENDIX A

Under the GDC, an Applicant must meet the standards specified in the GDC for all SS&Cs "important to safety." LILCO does not comply with this requirement because it utilizes an inadequate classification scheme.

LILCO equates "important to safety" with "safety-related." However, from the plain words of the Introduction to Appendix A as well as from Mr. Denton's November 1981 Memorandum, it is clear that "important to safety" -- the category to which Appendix A applies -- is broader than that class of items defined as "safety-related."[29/]

28/ FSAR Table 3.2.1-1, note 4, included in Exhibit 2; Deposition, at 54, 70-71 (Dawe).

29/ It bears remembering that the important to safety group:

> Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in important way [sic] to safe operation and protection of the public and all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation).

(Footnote continued on next page)

LILCO's failure to apply the GDC to items important to safety may be illustrated with concrete examples. A review of Table 3.2.1-1 (see Exhibit 2) indicates that neither the main feedwater system nor the main turbine is classified as safety-related. According to the LILCO scheme, this means that this equipment is unclassified and not subject to Part 50, Appendix B requirements. However, in the Shoreham Emergency Operating Procedures, LILCO operators are directed to call upon these systems to mitigate design basis events.[30/] Assuming arguendo that these systems do not need to be safety-related (i.e., meet the Part 100 criteria), it is submitted that equipment relied upon by Shoreham operators in actual practice to mitigate design basis events surely fits the definition of "important to safety" -- that is, a plant feature which contributes in an important way to safe operation and protection of the public.

The effect of LILCO's failure to classify in accordance with the GDC is to raise a significant concern as to the overall safety of the plant. Items important to safety are

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(Footnote continued from previous page)

Exhibit 1 hereto, p. 3 (emphasis in original).

30/ See discussion in Section V.A., infra and in Exhibit 4, hereto, where the Shoreham EOPs are discussed in detail.

supposed to be designed, fabricated, constructed, and tested in accordance with the GDC and "[a] quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions."[31/] These systems, however, at least according to the FSAR Table, are not classified as important to safety and the degree to which QA has been applied to them cannot be determined.

Accordingly, it is concluded that LILCO has failed to comply with Part 50, Appendix A, in that it has failed to classify SS&Cs in accordance with GDC requirements and has failed to apply the QA/QC requirements of Part 50, Appendix B as required by GDC 1.

IV.C. LILCO HAS NOT EVEN COMPLIED WITH THE DIRECTIVES OF THE REGULATORY GUIDES UPON WHICH IT RELIES

As noted in the previous section, it is our opinion that the LILCO classification scheme does not comply with the GDC. In addition, LILCO's scheme does not even comply with the regulatory guidance upon which it allegedly relies. Indeed, LILCO's classification system is fraught with inconsistencies which highlight LILCO's inadequate classification methodology. These inconsistencies are described below.

31/ 10 C.F.R. Part 50, Appendix A, GDC 1.

As noted above, a copy of FSAR Section 3.2, including Table 3.2.1-1 is attached hereto as Exhibit 2. The version of Table 3.2.1-1 included in Exhibit 2 is the most recent version published in the FSAR; at the time of the March 31 deposition, Suffolk County believed it contained an accurate statement of the LILCO position. At the deposition, however, it became clear that the Table was significantly flawed. Suffolk County thereafter was sent a letter, dated April 8, 1982, which detailed numerous changes to the FSAR. That letter is attached hereto as Exhibit 3. The hand notations on Table 3.2.1-1 represent the revisions to the Table necessitated by LILCO's April 8, 1982 letter.[32/]

To assist reviewers, we begin by specifying our understanding of the terminology used by LILCO in FSAR Table 3.2.1-1. First, the "Quality Group Classification" column ("QGC") in the Table refers to the classifications from Regulatory Guide 1.26. Thus, there are letters A-D which correspond to the Regulatory Guide classifications. Where that column is blank, the system is not a fluid system for which a Regulatory Guide 1.26 classification is appropriate.

Second, the "LILCO Quality Assurance Category" column ("QAC") is LILCO's means of specifying whether an item is

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32/ See footnote 21 above.

"safety-related" (if so, it is QAC "I") or non-safety-related (marked QAC "II").[33] Third, the "Seismic Category" column denotes whether an item must meet the Part 100, Appendix A standards for the SSE (marked "I") or is not required to meet those standards (marked "NA"). GE classifies equipment as either Seismic Category I, safety-related, or Non-Seismic Category I, not related to safety and noted in the FSAR Table 3.2.1-1 as "NA".[34]

1. Inconsistencies Between Quality Group (QGC) and Quality Assurance (QAC) Categories

Regulatory Guide 1.26 defines Quality Group D as the least stringent group of safety-related components. Hence items falling into this group, while of lowest priority in the Regulatory Guide, must still be regarded as safety-related. However, according to FSAR Table 3.2.1-1, 50 cases occurred where LILCO and GE QGC and Seismic Category entries do not match the Regulatory Guide Quality Group D standard. That is, in many instances where "D" occurs under the QGC category, LILCO has classified the item with a "II" notation under the QAC column, thus regarding it as not related to safety, and GE has classified the item with an "NA" notation in the seismic

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33/ Deposition at 68 (Dawe).

34/ Deposition at 13 (Robare).

column, again denoting not related to safety. Clearly, a "D" notation, defined in Regulatory Guide 1.26 as safety-related, should call for equally stringent classifications in the LILCO Quality Assurance category and in the GE Seismic category. These inconsistencies with the Regulatory Guide (upon which LILCO allegedly bases its classifications) are not justified in the FSAR.

An inconsistency also exists in LILCO's QGC "C" group. There is no question but that QGC "C" constitutes a safety-related category; even the LILCO witnesses agreed at the March 31 deposition.35/ However, there are three instances where the notations "C", "II" and "NA" are noted in the three respective categories of FSAR Table 3.2.1-1.36/ Here C, a more stringent quality group standard for safety-related components, is matched with LILCO's "II" classification and again with GE's "NA" classification, both of which designate the item as non-safety-related. This is entirely inconsistent with the Regulatory Guide. For convenience of the Board, these errors and the others discussed in Section IV are summarized in Table IV-1 which follows.

35/ Deposition at 58 (Dawe).

36/ The three inconsistencies regarding the "C" classification are on page 9 of Table 3.2.1-1, items 1, 5, and 7a under the Reactor Water Cleanup System. See Exhibit 2.

NUMBER OF INCONSISTENCIES IN FSAR CLASSIFICATION TABLE 3.2.1-1

Inconsistencies in Classification Categories	QGC - QAC [1]		QAC - Seismic [2]		Items Not Classified	Categories Not Defined
Types of Inconsistencies	D,II	C,II	I,NA	II,I		
Components						
1. Piping	12					
2. Valves	11	1				
3. Piping & Valves	3				2	
4. Pump Motor				1		
5. Pumps	4	1				
6. Tanks	4					
7. Cable			22		1	
8. Vessels	2	1				
9. Filters	2					
10. Heat Exchangers	2					
11. Recombiners	1					
12. Steam Jet Air Ejectors	1					
13. Main Steam Leads	1					
14. Blowers					1	
15. Heaters					1	
16. Electric Modules					1	
17. Site Grading						1
18. TSC (bldg)				1		
19. Displays				2		1
20. Subsystems				2		1
21. Firestops			1			
22. D.A. Multiplexer						1
23. Waterproof Doors			1			
24. Dryers				1		
25. Post Accident Sample Skid	1					
26. Other Components	1					
27. Sumps	1					
28. Intake Canal						
29. Feedwater & Condensate System	1					
TOTAL	47	3	24	7	6	4
TOTALS Per Revised data in /8/82 LILCO letter	49	3	24	7	0	0

[1] QGC-QAC = Inconsistency in safety classification between Quality Group Category (QGC) and the Quality Assurance Category (QAC). The primary difference is between QGC "D", which is safety-related according to Regulatory Guide 1.26, and LILCO QAC "II", which LILCO defines as non-safety-related. See, for example, Table 3.2.1-1, item II-9. As an example of a discrepancy of the C,II type, see Table 3.2.1-1, item XIX-8.

[2] QAC-Seismic = Inconsistency in safety classification between the Quality Assurance Category (QAC) and the seismic classification. The primary differences are between QAC I, denoting safety-related, and seismic NA, denoting non-safety-related. See, for example, item II-16. As an example of a discrepancy of the II,I type, see Table 3.2.1-1, item XXXII-1.

2. Inconsistencies Between Quality Assurance and Seismic Categories

There are 31 instances where LILCO QAC classifications and GE seismic classifications are inconsistent. Twenty-four of these occur where LILCO classifies a component as "QAC I," designating the item as safety-related, but GE classifies it as "seismic NA," meaning not-safety-related. If LILCO's classification is correct and these are safety-related devices used to mitigate an accident, then GE must also assume that they are safety-related.

Seven additional inconsistencies occur between LILCO and GE classifications. Each involve LILCO's classifying an item "QAC II," meaning not related to safety, while GE notes "seismic I," thus regarding the item as safety-related. This type of inconsistent classification may be due to a special seismic requirement on a non-safety-related piece of equipment; however, it is not in good quality practice and is not explained in the FSAR.

3. Items Not Classified and Errors in Classification

Under Table 3.2.1-1, Item XXXIV, "Main Steam Isolation Valve Leakage Control System," LILCO originally provided no QAC classification for the six items listed, while GE classified these same items as Seismic Class I. Thus, it is clear that

the basic document used to define quality classification of equipment (the FSAR) does not provide sufficient information to determine the quality class of these components. By letter dated April 8, 1982 (Exhibit 3 hereto), LILCO indicates that the FSAR was in error and that these QAC items should have been "I."

4. Dissimilar Classification of Similar Valves

Under Table 3.2.1-1, Item XIX, Nos. 7a and 7b, "Reactor Water Cleanup System," the valves of the reactor water cleanup system purchased by two different sources (GE and Purchaser), both are classified as Quality Group C and as Seismic Category I. However, these items, which are described identically, are purchased to two different QAC categories (QAC I and II) without adequate description of the difference in their requirements. This inconsistency is additional to that evidenced by the fact that QGC items classified as "C" should clearly have a QAC rating of "I."

5. NRC Requested Expansion of the Shoreham Classification Table

The NRC request No 206.1 asks that the Shoreham Q-List (Table 3.2.1-1), be expanded to include many additional items. Of these items, 31 did not appear in the Q-List and the NRC asked that they be included; the NRC asked that 9 more be

expanded and/or clarified; and it asked that 23 additional items be added as a result of TMI. LILCO's response was to include some items but specifically not to include others for various reasons.[37]

A brief assessment of LILCO's responses with respect to the 31 items requested to be included in the Q-List shows that some were not included because LILCO argued they should not be included. Others were claimed to be part of systems and components already listed in Table 3.2.1-1. However, for several of these items supposedly included in the Table, it is impossible to discern where in Table 3.2.1-1 the items are allegedly included. Table IV-2, which follows, describes examples of these two types of items. LILCO has not seen fit to change the classification Table 3.2.1-1 for these items, and thus their classification remains undiscernable to someone reading the FSAR.

37/ LILCO FSAR Amendment 23, October 1981, FSAR Vol. 15, NRC Requests and Responses.

TABLE IV-2

ITEMS FROM NRC REQUEST FOR EXPANSION OF SHOREHAM Q-LIST[1/]

NRC Item No.	Items NRC Requested be Added to Shoreham Q-List[2/]	LILCO's Reason for not Including in Table 3.2.1-1
A-3.	Combustible gas control system Containment drywell hydrogen monitoring system	This system is part of the primary containment atmosphere control system, Section XXVIII
A-6.	Containment Spray System	This is part of the RHR system, Section IX
A-7j	Onsite Power Systems - AC Control Power inverters	Further clarification provided in 3.2.1-1, Section XXVII
A-7k.	120 v. AC vital bus distribution equipment	Further clarification provided in 3.2.1-1, Section XXVII
A-7f.	Onsite Power Systems - Conduit and cable trays and their supports	These are off the shelf hardware items. Thus they will not be included in 3.2.1-1.
A-8c.	D.C. Power Systems (IE) - Conduit and cable trays and their supports	These are commercial grade hardware items. Not included in 3.2.1-1.
A-13.	Radioactive Contamination measurement and analysis	This is an administrative requirement. Not included in 3.2.1-1.
A-23.	Meteorological data collection programs	This is an administrative requirement. Not included in 3.2.1-1.
A-24.	Expendable and consumable items necessary for the functional performance of safety-related structures, systems & components	This is an administrative requirement. Not included in 3.2.1-1.
A-26.	Measuring and test equipment used for safety-related structures, systems & components	This is an administrative requirement. Not included in 3.2.1-1.
C-13	ADS actuation (II.K.3(18))	There is no hardware change required to this item, thus no modification to Table 3.2.1-1 is required
C-19	ADS valves, accumulators, and associated equipment and instrumentation (II.K.3 (28))	There are no hardware changes required to this item, thus, no modification to Table 3.2.1-1 is required.

1/ From NRC Request 260.1, FSAR Vol. 15 NRC Requests and Responses. Items not listed were either incorporated, in whole or in part, in Table 3.2.1-1, or were not applicable to Shoreham.

2/ The statement in the NRC Request 260.1, for items A and C are as follows:

"Section 17.1.2.2 of the standard format (Regulatory Guide 1.70) requires the identification of safety-related structures, systems, and components (Q-list) controlled by the QA program. You are requested to supplement and clarify the Q-list in Table 3.2.1-1 of the FSAR in accordance with the following:

A. The following items do not appear on the Q-list (FSAR Table 3.2.1-1). Add these items or justify not doing so.

C. Enclosure 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980) identified numerous items that are safety-related and appropriate for QL application and therefore should be on the Q-list. These items are listed below. Add these items to the Q-list and/or indicate where in the Q-list they can be found. Otherwise justify not doing so."

V. NOT ALL EQUIPMENT USED FOR EMERGENCY OR ACCIDENT MITIGATION IS LISTED IN THE SHOREHAM CLASSIFICATION TABLE

V.A. COMPARISON OF EQUIPMENT CALLED FOR IN FSAR CHAPTER 15 DESIGN BASIS ACCIDENTS AND IN EMERGENCY OPERATING PROCEDURES.

In order to investigate further the methodology used at Shoreham for the classification of "safety-related" SS&Cs, a review was conducted of several FSAR Chapter 15 design basis accident analyses and the Shoreham Emergency Operating Procedures ("EOPs") corresponding to such accident scenarios.

1. Methodology

Specifically, using the Shoreham EOPs that were provided by LILCO to Suffolk County through the discovery process, a correlation was made between the accident scenarios addressed in the EOPs and those analyzed as design basis accidents in FSAR Chapter 15. From this correlation, six design basis accident/Emergency Operating Procedure pairs were assembled, reviewed and evaluated. The six pairs are as follows:

-- Feedwater Controller Failure -- Maximum Demand (FSAR 15A.1.7);
Feedwater/Level Control System Failure Emergency Procedure (SP 29.006.01, Rev. 0).

-- Loss of Feedwater Flow (FSAR 15A.1.18);
Feedwater/Level Control System Failure Emergency Procedure (SP 29.006.01, Rev. 0).

Emergency Procedure (SP 29.006.01, Rev. 0).

-- Loss of Condenser Vacuum (FSAR 15A.1.21); Loss of Condenser Vacuum Emergency Procedure (SP 29.012.01, Rev. 0).

-- Anticipated Transients Without Scram (FSAR 15.1.27); Anticipated Transients Without Scram Emergency Procedure (SP 29.024.01), and Emergency Use of S.L.C. Emergency Procedure (SP 29.004.01, Rev. 0).

-- Loss of AC Power (FSAR 15A.1.19); Loss of Off-Site Power Emergency Procedure (SP 29.015.01, Rev. 1).

-- Pipe Breaks Inside the Primary Containment, or Loss of Coolant Accident (FSAR 15.1.34); Loss of Coolant (Large Break) Emergency Procedure (SP 29.014.02, Rev. 0), and Loss of Coolant (Intermediate Break) Emergency Procedure (SP 29.014.01, Rev. 0).

The Emergency Shutdown Emergency Procedure (SP 29.010.01, Rev. 2) is referenced in each of the EOPs above, except for the ATWS procedures. To the fullest extent possible, operator actions called for by the Emergency Shutdown Procedure were incorporated into the analysis of each scenario and respective procedures cited above.

The following methodology was used in comparing each FSAR Chapter 15 accident analysis with its respective EOP(s). First, each system and/or component relied upon in either the accident analysis or the procedure was identified, regardless of whether such system and/or component was required for passive or active function, or manual or automatic initiation. This phase of the review was intended to determine which systems and/or components were credited with fulfilling their respective functions in the course of design basis accident mitigation, according to the FSAR and/or EOPs.

The second phase of the review involved the correlation of those systems and/or components identified in the process described above, with their respective quality assurance classification requirements as stated in FSAR Table 3.2.1-1. This portion of the review was intended to determine which of those systems and/or components relied upon in either the Chapter 15 analyses or the EOPs had been previously classified as "safety-related" or "non-safety related."

The results of the comparative review of the six design basis accident analyses and their corresponding EOPs are presented in Exhibit 4.

2. Discussion of the Results

As shown in Exhibit 4, the EOP listing of equipment relied upon for mitigation of the six design basis accidents is significantly more comprehensive than that obtained from the review of FSAR Chapter 15. However, the review of both revealed that all the systems and/or components identified in the Chapter 15 analyses are acknowledged for functionability in their respective EOPs. Thus, one may conclude that a systematic review of EOPs is likely to produce a more comprehensive listing of equipment for further safety classification than that produced by a review of the FSAR Chapter 15 accident analyses.

In addition, although numerous items of equipment called upon for accident mitigation were identified from the analysis of EOPs alone (i.e., such equipment was not identified by review of Chapter 15 analyses), these additional systems and/or components are, for the most part, already identified and classified in Table 3.2.1-1. Thus, one may conclude that FSAR Chapter 15 design basis accident analyses, as presented in the FSAR, were not the sole determinant of SS&Cs to be relied upon for the development of the Q-List (Table 3.2.1-1). Furthermore, it appears that the process of evaluating design basis accidents and the equipment relied upon for mitigation of such accidents has not been the sole method of identifying the

applicability of 10 CFR 50 Appendix B quality assurance requirements for those SS&Cs listed in Table 3.2.1-1.

It appears that a non-systematic approach toward safety classification has been applied to SS&Cs identified and analyzed in Table 3.2.1-1. However, such a non-systematic approach has the potential for (a) overlooking certain equipment in terms of its safety function and corresponding classification, and (b) mis-classifying those SS&Cs not specifically identified in the set of design basis accidents, but whose function may, in fact, be relied upon in EOPs. This potential for safety classification and qualification errors, resulting from such non-systematic categorization schemes for equipment not identified in Chapter 15, could be alleviated by the use of systematic methodology.

An example of where the use of a systematic methodology, such as the comparative review described herein, would facilitate the appropriate classification of equipment commensurate with its safety function, is provided by the turbine bypass system. As shown in Exhibit 4, the turbine bypass system is called upon by the operators in each of the EOPs reviewed. This same system was not necessarily identified in all the corresponding Chapter 15 accidents reviewed. The turbine bypass system is important in mitigating pressure transients in the primary system resulting from initiating

events or transients by loss of the main turbine and/or generator. However, this system, like the main turbine and supporting turbine-generator auxiliaries, is not classified nor is it qualified in accordance with the quality assurance requirements of 10 CFR 50 Appendix B.

Similarly, the feedwater control system is another example of a "non-safety related" system identified by the systematic review of Chapter 15 accident analyses and EOPs. While only called upon in one of the six design basis accidents reviewed, the manual operation of this system is relied on by the operators for mitigation of excessive feedwater flow into the reactor vessel caused by a feedwater/level control system failure. Thus, for this non-safety related system, a failure within the automatic control function of the system not only serves as the initiating event for one of the Chapter 15 design basis accidents, but the mitigation of this accident calls for regaining manual control of this non-safety related system. In addition, the need for feedwater control is credited in other Chapter 15 accident analyses, such as the MSIV closure transient, although this analysis has not been specifically considered in this review.

To summarize, the review of EOPs reveals that several key systems and/or components are repeatedly 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. While shutdown of the plant may indeed be demonstrable within the bounding criteria of 10 CFR 100, Appendix A without the use of such "non-safety-related" equipment for mitigation of design basis accidents, a review of both Chapter 15 and the EOPs seems to indicate otherwise; in any event, the operators clearly are directed to utilize these systems. In other words, there are systems and/or components currently classified as "non-safety-related" that face significant demands for availability, control and operability during analyzed accidents. Furthermore, the review of Chapter 15 and EOPs as discussed herein considered the sequence of events and equipment demands for only six accident scenarios. A more complete, systematic method of reviewing operator actions identified in the EOPs for mitigation of all possible transients would likely encompass the evaluation of additional "non-safety-related" equipment.

Thus, one may conclude that a thorough, systematic method of identifying equipment to be used for accident mitigation, will provide a keener insight into what the appropriate safety and quality assurance criteria for specific components should be. An example of such methodology would be the use of probabalistic techniques in which the relative importance of equipment and safety functions in specific accident sequences can be assessed. In this manner, the use of sophisticated analysis techniques for evaluating design basis accidents will likely provide a greater degree of assurance that equipment relied upon for mitigation of such accidents will be identified and will meet the quality levels commensurate with the safety function to be performed. This would result in greater assurance for the operators that equipment relied upon in EOPs would be available to perform such functions.

V.B. SEVERAL IMPORTANT SYSTEMS AND COMPONENTS ARE NOT INCLUDED IN SHOREHAM'S CLASSIFICATION TABLE

In addition to the foregoing review of EOPs, we also have described a further shortcoming of LILCO's classification system: it ignores certain components which clearly play important roles in responding to transient and accident conditions.

For example, GE relies on a review of equipment against regulations (10 C.F.R. 50.55a, RG 1.26 and RG 1.29) to decide

if any item should be classified as safety-related. GE utilizes only two classification levels: safety-grade and non-safety grade. Generally, all equipment that is relied upon to mitigate design basis accidents or transients (those in Chapter 15 of the FSAR) is classified by GE as "safety-grade," with the remainder classified as non-safety grade. However, GE also admits that there are several systems which may be relied upon in whole or in part to mitigate accidents or transients but which are not classified consistent with the GE criteria, and thus do not appear as safety-related classifications in the FSAR Classification Table 3.2.1-1.[38/] GE identified the following systems as examples of this inconsistent classification:

-- Turbine Bypass System

-- Level 8 Trip

-- Rod Block Monitor (RBM: part of the RMCS)

-- RCIC (partial; instruments and controls only)

This situation is not acceptable in view of the important roles which these systems perform. For example:

38/ Deposition at p. 37 (Robare)

(1) The Turbine Bypass Valve allows steam to continue to flow to the main heat sink (the condenser) when the flow to the turbine has to be interrupted, thus avoiding major pressure transients;

(2) The Level 8 Trip is used to warn the operators of possible overfilling of the vessel, a condition which could lead to water in the steam lines and/or opening of the safety/relief valves;

(3) The Rod Block Monitor is used to prevent the out of sequence movement of a high worth rod to avoid serious rod drop accidents; and

(4) The RCIC is the back-up for the HPCI system, one of the major emergency core cooling systems.

Given the importance of these systems and components, they should be classified as safety-related. The fact that they are not, is another example of the inadequate classification methodology utilized by LILCO for Shoreham.

In addition, there are systems called upon in the FSAR analysis of design basis accidents which are not fully included in Table 3.2.1-1. For example, the feedwater control system is called upon in EOPs for some accidents and transients, but the only feedwater component mentioned in the Table is the

feedwater piping connected to the RCPB which is safety-grade. See Table 3.2.1-1, p. 2. Similarly, the automatic depressurization system ("ADS") is used to reduce pressure in order to allow low pressure cooling systems to be utilized in mitigating some loss of coolant accidents; however, only the safety relief valves are listed in the Table, not the ADS controls and instrumentation. See Table 3.2.1-1, p. 2.[39/] Finally, manual insertion of control rods is called for in the ATWS Emergency Operating Procedure, but only the CRD hydraulics is classified as safety-grade; the remainder of the CRD System is not identified in the classification table. The FSAR does not discuss the omission of these systems and components.

These examples of classification deficiencies were discernable with only a brief review; with a detailed analysis, there are undoubtedly other systems or subsystems whose classifications are either inconsistent with LILCO and GE criteria for classification, or inscrutable in the present Classification Table 3.2.1-1.

39/ The NRC request No. 260-1, item C.19, called for the addition to the Shoreham Q-List (Table 3.2.1-1) of "ADS valves, accumulators and associated equipment and instrumentation". LILCO's response was not to add these components to the list stating that "there is no hardware change related to this item, thus no modification to Table 3.2.1-1 is required."

VI. EXAMPLES OF SYSTEMS INTERACTIONS AND CLASSIFICATION ERRORS AT SHOREHAM

Two examples are provided to demonstrate further that the current methods of design and regulatory review at Shoreham are not adequate to detect systems interactions or discrepencies in classification. The interaction example concerns the water level system which is one of the important safety system inputs. Classification problems are discussed with respect to the Standby Liquid Control System.

VI.A. WATER LEVEL SYSTEMS INTERACTION: AN EXAMPLE OF UNDETECTED PROBLEMS AT SHOREHAM

1. Importance of Water Level Measurement

Water level is one of the most important parameters measured and monitored in a boiling water reactor (BWR). An adequate supply of water is critical to assuring that the nuclear fuel is cooled, moderated and that it remains intact. Water level varies during many plant maneuvers (e.g., start-up, power level changes, shutdown, and transients), and, accordingly, prompt response to changes in level is necessary to ensure safe operation.

In many cases the operator must change the water flow to increase or decrease the vessel water level, based on the water level indicators. In other cases, automatic equipment makes the level changes based on water level indicators and steam and

water flow controls. Reliable water level indication is critical to safe plant operation. Water level must always be above the active fuel length to assure fuel integrity and no release of radiation.

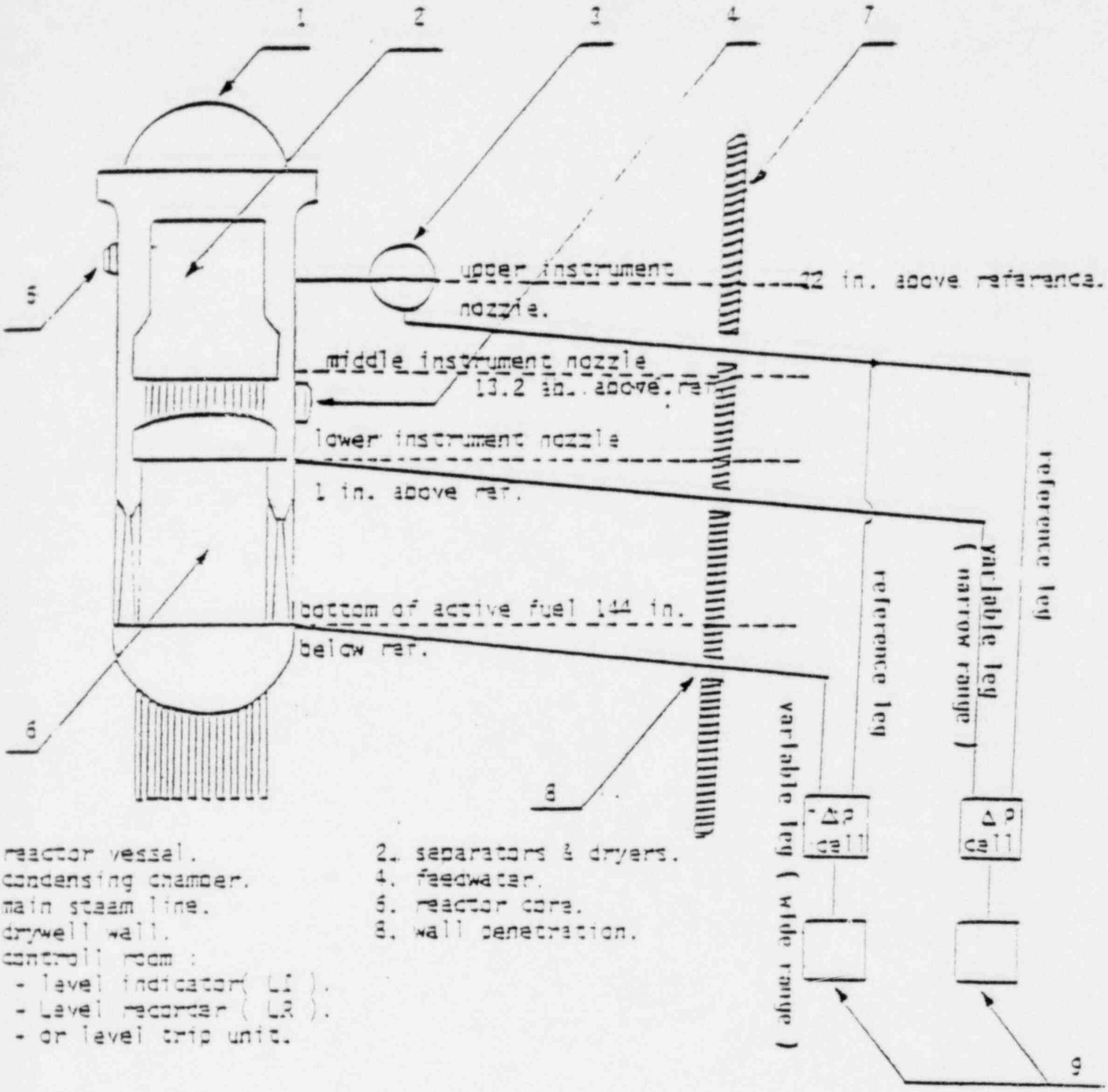
The only direct indication of water level in Shoreham is the cold leg reference water level indicator. This indicator may have flashing in the reference leg during a depressurization and cool down, which could result in a false high indication when core water level actually is low. Such a false reading may lead operators to delay instituting manual action and since automatic ECCS actions are initiated from these water level indicators, emergency core cooling could also be delayed.

2. Shoreham Water Level Instrumentation

All level measurement systems in BWR's employ differential pressure transmitters, a reference leg connected to a condensing pot and, in turn, to the reactor vessel steam space, and a variable leg connected to the vessel at a lower elevation. Several differential pressure cells share common impulse legs. Temperature compensated and uncompensated reference legs are employed. Those level measurement systems which use a temperature compensated reference leg are called Yarways; those which use an uncompensated reference leg are

called cold leg instruments or, often, GEMAC. BWR 1's, 2's, 3's and some 4's use two redundant Yarways to generate engineered safety feature actuation signals and cold leg instruments for indication and control. The remaining BWR 4's and all 5's and 6's use redundant cold leg systems exclusively.

Shoreham uses redundant cold reference leg systems. See Figure VI-1, which follows. These cold reference leg indicators also serve to actuate water level trips which, in turn, activate emergency core cooling systems. There are no other direct water level indicating systems at Shoreham. Other indications of water level and cooling of the nuclear fuel during normal operations can only be inferred (indirectly) from: (1) fission product gases detection; (2) pump flow rates and valve line-ups; and (3) neutron flux monitoring. During transients, these indicators may be isolated or unreliable.



1. reactor vessel.
2. separators & dryers.
3. condensing chamber.
4. feedwater.
5. main steam line.
6. reactor core.
7. drywell wall.
8. wall penetration.
9. controll room :
   - level indicator(LI).
   - Level recorder (LR).
   - or level trip unit.

Figure VI-1 : Cold Reference Leg Design and Location of Instrument Taps for Shoreham.

Note: reference level = top of active fuel.

3. The Water Level Measurement Problem

During some accident conditions and some plant maneuvers, the cold leg reference indicators currently at Shoreham would provide inaccurate water level indication. The indication could be in error by as much as 9 feet.40/

For example, small (e.g., 0.01 ft^2) and intermediate (e.g., 0.04 ft^2) break accidents (LOCA's) that discharge steam into the drywell (at temperatures as high as 340°F) for an extended time period result in substantial heatup of components/air in the drywell (including reactor water level sensing lines). If the reactor is subsequently depressurized, water in the reactor water level sensing lines located in the drywell will flash. This common event can cause a loss-of-direct measurement capability, because the high drywell temperature interferes with all water level measurement. In fact, "even without a break loss of non-safety grade containment coolers would cause the containment to heat up and could cause flashing upon depressurization."41/ Therefore, a system classed as non-safety related could cause errors of significant magnitude in a safety system.

40/ See LILCO Response to Suffolk County Interrogatories and Suffolk County Second Set of Interrogatories, March 26, 1982, p. 14 and referenced attachments.

41/ Board Notification 82-08, Errors in BWR Vessel Water Level Indication, Exhibit 5 hereto.

At Pilgrim Nuclear Power Station, during a routine reactor shutdown and cooling in 1981, a systems interaction event occurred which had the potential to cause a loss-of-all-water level indication. The cause of the loss-of-accurate-level indication ability was totally unrelated to the water level indicators. Temperature of the surrounding drywell had risen to 240 degrees Fahrenheit at a time when the reactor coolant was at 220 degrees Fahrenheit. This caused the water to flash in the heated reference legs (Yarway type indicators - different from those at Shoreham but similar problems could occur with Shoreham cold leg indicators), which in turn, caused an erroneous high reading, then a corresponding erroneous low reading. Thus, there was an oscillation in the indicators which alerted the operators to the problem. The false high level signals could have caused an operator either to terminate safety injection prematurely or to operate at a lower than desireable water level. As the Pilgrim experience demonstrates, no accident is necessary for high containment temperatures to occur; at Pilgrim, the containment coolers, which are classified as non-safety grade, did not function according to design. The containment coolers allowed the containment temperature to rise and since there is no limit on containment temperature, no action was taken.

The systems interaction concerns addressed in this testimony are exemplified by the water level flashing problem discussed above.[42] There is no separate diverse water-level indicator, operating on a different principle, to provide any other indication. There is no mechanical or electrical connection between non-safety grade containment coolers and safety-grade water indicators, even though, as evidenced by the Pilgrim experience, the effect of the containment cooling "failure" can cause safety-related water level indication failures.

There is clearly an effect on reactor safety if water level indication fails. This type of interaction between supposedly unrelated components could be better accounted for by more rigorous systems interaction studies that would assess temperature effects and flooding effects on reliability and functional performance. However, the existing analysis and review techniques as documented in the FSAR and SER failed to discover this problem or other systems interaction problems which may still await detection.

42/ The NRC's Weekly Information Report for the Week Ending Jan. 22, 1982 identifies two additional examples of water level systems interaction at other BWR plants. The NRC document summarizing these events is attached as Exhibit 6.

VI.B. STANDBY LIQUID CONTROL SYSTEM: EXAMPLES OF CLASSIFICATION DEFICIENCIES

Neither the Shoreham FSAR nor the NRC's corresponding Safety Evaluation Report (SER) demonstrates that the Standby Liquid Control System (SLC) is properly designed and qualified. These two reports contain several discrepancies which confuse the issue of whether SLC equipment is properly classified. Further, the FSAR and SER are so general so that it is difficult to determine the standards according to which the SLC was designed and qualified.

The SER lists the SLC as a "system required for safe shutdown."[43/] Therefore, it is a safety system, and components which are required for its function should be classified as safety grade (Seismic Class I). The FSAR seems to agree with that proposition, stating in Section 4.2.3.4.3: "The SLC is a special safety system and is maintained in a standby status whenever the reactor is critical and at all times when it is possible to make the reactor critical."[44/] It goes on to state that "the SLC equipment essential for injection of neutron absorber solution into the reactor is designed as Seismic Category I."[45/]

43/ NUREG-0420, Shoreham Nuclear Power Station, Safety Evaluation Report, U.S. Nuclear Regulatory Commission, Washington, D.C. pp. 7-9 and 7-10.

44/ LILCO FSAR, p. 4.2-84.

45/ *Ibid.*, p. 4.2-85.

The equipment referred to in Section 4.2.3.4.3 can be seen by inspecting FSAR Fig. 7.4.1-3, "Standby Liquid Control System P&ID." The vital components are:

a) Inside Containment Check Valve

b) Explosive Valves

c) Cable to Explosive Valves

d) Keylock Actuator

e) Check Valves Downstream of Isolation Valves

f) Isolation Valves

g) Accumulators

h) Pumps

i) Pump Motors

j) Cable to Pump Motors

k) Storage Tank Valve

l) Storage Tank Heaters

m) Storage Tank

n) System Piping

o) Heater Indicator Lights

p) Pump Indicator Lights

q) Explosive Valve Detonator Circuit Continuity Indicator Lights

Not all the above components appear in the FSAR Table 3.2.1-1, which lists equipment classification. Therefore, in the case of the tank heaters, indicator lights, cables

connecting major components (c and j above), and the keylock actuator, the classification cannot be ascertained.[46/]

The FSAR Fig. 7.1.1-2, "Codes and Standards Applicability Matrix" lists Regulatory Guides (RG's), General Design Criteria (GDC), and IEEE Standards applicable to the SLC instrumentation and controls. From this Figure, one can deduce that some SLC instrumentation and control (I/C) equipment was qualified to seismic requirements because it was designed (by GE's choice) to satisfy IEEE Standards 323-1971 and 344-1971. However, LILCO does not claim that any of the I/C equipment conforms to RG 1.29, and therefore, one cannot tell if this equipment is safety related or whether it meets the 10 CFR Part 50, Appendix B Criteria.

Finally, two statements by the NRC add to the overall confusion over the SLC. First, the SER states that there are two keylock actuators which would indicate redundancy in the design,[47/] whereas the FSAR shows only one.[48/] Second, the SER concludes that "the design of the SLC conforms to the requirements of GDC 26."[49/] In the FSAR Fig. 7.1.1-2, LILCO

46/ FSAR Table 3.2.1-1, pp. 3 and 4. See Exhibit 2.

47/ Ibid. 43, p. 9-8.

48/ Ibid. 44, Fig. 7.4.1-3.

49/ Ibid. 43, p. 9-8.

does not claim that the SLC I&C equipment conforms to either GDC 20 (Protection System Function) or GDC 26 (Reactivity Control System). This inconsistency in design basis is not explained.

In summary, the FSAR and SER do not demonstrate that the SLC is properly designed, classified, and qualified. This may be due to the ambivalent and general wording used in these reports, but at this time, a finding that the SLC is safe and reliable and properly classified and qualified is unsupported.

In a brief review of two key systems, we have concluded that there are major deficiencies in the ability of the present classification and analyses techniques, as applied to Shoreham, to produce a consistent and reliable technique for classifying safety-related components. The techniques also are deficient in their ability to discover major systems interactions, amongst existing systems, which may degrade their abilities to perform their safety functions.

VII. UNRESOLVED SAFETY ISSUES ARE NOT PROPERLY EVALUATED FOR SHOREHAM

In previous sections of this testimony we have addressed our concerns for LILCO's inadequate and non-systematic approach to systems interactions problems at Shoreham. In this Section, we address the Staff's failure to address the unresolved safety issues resulting from systems interaction and the Staff's

failure, consistent with Appeal Board guidance, to explain why it is safe to proceed with operation absent resolution of these issues.[50]

Unresolved Safety Issues ("USIs") A-17 and A-47 address the safety implications of multiple-failure accidents. In this testimony, we refer to these related deficiencies in accident and safety analysis -- the lack of systems interaction analysis, the lack of multiple or "common-cause" failure analysis, and the tendency of the "single-failure criterion" to exclude a large number of potential accident-causing events -- as the "systems interaction issue."

VII.A. THE NEED TO ADDRESS THE SYSTEMS INTERACTION ISSUE

This issue has become extremely significant after the TMI-2 accident, which itself involved not a single failure but rather a series of failures, or domino effect, which included both dependent and independent multiple-failures. The Kemeny Commission found that "[t]he accident at TMI-2 was a multiple-failure accident,"[51] as did the NRC's Special Inquiry Group,[52] but "[i]n the licensing process, applicants are only

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50/ See Virginia Electric & Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, CCH Nuc. Reg. Rptr. ¶30,321 (1978).

51/ Kemeny Report, p. 52.

52/ Ibid., p. 148.

required to analyze 'single failure' accidents. They are not required to analyze what happens when two systems fail independently of each other,"[53] nor to assess possible adverse interactions among systems.

As a result of the foregoing findings, the Kemeny Commission called upon the NRC to emphasize a systems engineering examination of overall plant design and performance which would include interaction among major systems and increased attention to the possibility of multiple failure accidents.

The NRC's Special Inquiry Group also criticized the NRC's safety and accident analysis approach as inadequate, noting that "one of the obvious lessons" of TMI-2, "is the critical need for overall plant and systems analysis," and, with particular regard to the concentration of engineering design and analysis teams on single specialized systems, "[t]here is as much or more of a chance that safety matters will 'fall in the cracks' between two or more highly proficient technical groups as there is for a safety error to be made in any of the specific groups."

53/ Ibid., pp. 19-20.

In addition, in August 1980, the ACRS recommended an investigation of the safety implications of control system failures as follows:

> Recent experience has indicated that more attention must be given to reactor control system reliability. Most safety analyses in the past have given minimum attention to control system reliability based partly on the assumption that failure of the system makes it unavailable and ignores the fact that this failure may actually produce an unsafe mode of reactor behavior. This problem should receive further study to determine appropriate reliability standards for control systems. Appropriate reliabilty of nonsafety system information displayed for use of the reactor operator is a related important issue.

VII.B. THE STAFF RESPONSE TO SYSTEMS INTERACTION ISSUES IS INADEQUATE

In response to the ACRS, and other Staff research activities, the systems interaction issue was approved as a USI by the NRC on December 24, 1980 and assigned the designation as Task A-47. However, as of February 19, 1982, nearly a year after initiation of the issue, and River Bend (ALAB-444), no Task Action Plan has yet received Staff concurrence and approval.[54/] No detailed description of the lack of a Task Action Plan is provided by the Staff in the Shoreham SER. This is a serious omission, which leaves unanswered the North Anna inquiry of why it allegedly is acceptable to permit operation

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54/ NUREG-0606, Vol. 4, No. 1, February 19, 1982, p. 41.

of Shoreham with this issue unresolved and with no apparent progress toward resolution.

Task A-47 concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. The Staff has determined, as set forth in Appendix B of the Shoreham SER, that neither the Staff nor LILCO has conducted a systematic evaluation of the Shoreham control system to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than those analyzed. In addition, no plan or schedule for such a Shoreham-unique study has been developed. The Staff has stated, however, and we concur, that "it is not likely that it will be possible to develop generic answers to these concerns but rather plant-specific reviews will be required."55/ Clearly, if plant-specific reviews are needed and none are even scheduled yet for Shoreham, a basis for finding adequate progress toward resolution of this issue cannot be found.

A subtask of A-47 is to assess the need for preventative and/or mitigative design measures to preclude or minimize the consequences of a reactor overfill transient. In response to this potential problem, Shoreham has installed and relies upon

55/ NUREG-0606, Vol. 4, No. 1, February 19, 1982, p. 41.

commercial grade, rather than safety-grade, high level trips (level 8) to terminate flow from appropriate systems.56/ Such reliance on less than complete safety-related systems and components for accident mitigation, conflicts with LILCO's responsibility to provide reasonable assurance that the facility can be operated without undue risk to public health and safety.

In addition, as stated in the SER the Staff has requested LILCO to provide additional information on control system failures. The Applicant has committed to conduct a review (a) to identify any power sources or sensors which provide power or signals to two or more control systems, and (b) to demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences outside the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems. Further, the Staff has requested a review by LILCO to determine whether the harsh environments associated with high energy line breaks might cause control system malfunctions and result in consequences more severe than those of FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.57/ While LILCO has committed to

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56/ NUREG-0420, Shoreham SER, p. 8-15 and 8-16.

57/ NUREG-0420, Shoreham SER, Suppl. 1, p. 7-3 and 7-4.

conduct such a review, we do not believe that this limited and piecemeal commitment constitutes an acceptable resolution of this issue. Rather we believe the Staff and Applicant have failed to demonstrate that: (a) a solution satisfactory for Shoreham has been implemented; (b) a restriction on the level or nature of operation adequate to eliminate this type of problem has been imposed; or, (c) the safety issue does not arise until the later years of operation. Thus, we conclude that the Staff has not provided in the Shoreham SER an explanation of either the present status of the Task A-47 generic studies, or the measures employed at Shoreham to compensate for the current absence of the answers sought by these studies.

The Staff's treatment of Task A-17 is even more deficient. The Staff's cursory summary and general description of the systems interaction issue, as set forth in Task A-17 and as reviewed by the Staff in Appendix B of the Shoreham SER, is totally devoid of the required Shoreham-specific information. First, the Staff has made no attempt to describe any actions, other than the normal regulatory reviews, that have been implemented at Shoreham to resolve the issue of multiple-failures. Likewise, the Staff has failed either to impose a restriction on the level or nature of operation adequate to eliminate the problem which may be encountered as a

result of systems interaction, or to demonstrate, factually, that the safety issue does not arise until the later years of Shoreham's operation.

Further, the status of Task A-17, as reported in the Shoreham SER, is replete with errors and omissions. For example, the NRC Staff has not demonstrated a determination to discover an answer to the systems interaction issue. The nine-man Systems Interaction Branch formed in April 1980 (see p. B-11 of the SER), was "dissolved 5/81 and 2 people transferred to RRAB/DST to perform (the) SI program for NRR." The reallocation of resources "will delay 6 pilot LWR reviews by at least 6 months" and "the characterization of SI experience at operating reactors was postponed."58/

In summary, the systems interaction issue, as set forth in Tasks A-17 and A-47, has been recognized again and again as a high-priority, high-risk-potential, unresolved safety issue. The history of the systems interaction issue has been one of repeated recognition, classification, re-classification, and

58/ The Staff also states that recent review concluded "that no single method presently exists in a form that can be used to perform an adequate review for adverse SI," and thus the Staff stated "the task of developing interim guidance must define a program combining the applicable methods." TMI Task II.C.3. However, the definition and implementation of such a composite resolution is not being pursued in a timely manner for Shoreham.

re-listing without much real progress. Eight years ago, in 1974, the ACRS first requested that the NRC Staff give attention to the problem. It was a Category A high-priority generic technical issue in NUREG-0410, published in January, 1978.[59/] It was classed as a "potential high risk" issue in the NRC's 1978 risk-based evaluation of unresolved safety issues.[60/] It was one of the top 20 unresolved safety issues listed in the NRC's January, 1979, Report to the U.S. Congress.[61/] In fact, in the Denton/ Steering Committee "point value" NRC manpower allocation directives, no issue had a higher point value.[62/] It is part of the TMI Action Plan (as "Task" II.C.3), with a priority 1 classification and a point value indicating that it was again determined to be of high safety significance for the Action Plan as well.[63/]

59/ NUREG-0410, NRC Program For the Resolution Of Generic Issues Related to Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., January 1978.

60/ Taylor, et al, Summary Report On A Risk Based Categorization of NRC Technical and Generic Issues, preliminary draft issued by NRC, page 1. In addition, see NUREG-0510.

61/ NUREG-0510, Identification of Unresolved Safety Issues Relating to Nuclear Power Plants, Report to Congress, U.S. Nuclear Regulatory Commission, Washington, D.C., January, 1979.

62/ See NRC Document SECY-79-76 (January 30, 1979), a memorandum from Harold R. Denton, Director, Office of Nuclear Reactor Regulation, to the NRC Commissioners. In addition, see memorandum from Harold R. Denton to Roger S. Boyd, et al., January 23, 1979 (attached to SECY-79-76).

63/ NUREG-0660, NRC Action Plan Developed as a Result of the TMI-2 Accident, Vols. I, II, U.S. Nuclear Regulatory Commission, Washington, D.C., May 1980.

Further, the issue is directly applicable to Shoreham. A number of possible evaluation approaches are available, but not planned for Shoreham, including methodologies based on analytical techniques, physical walkdowns, and operating experience. We believe that the Staff and Applicant have failed to demonstrate why it is acceptable to permit Shoreham to operate in the face of the safety issues under study. We also believe that the Staff has failed to explain properly either the present status of generic studies A-17 and A-47 or the measures employed at Shoreham to compensate for the current absence of the answers sought by these studies. Finally, we believe that the Shoreham plant and operating procedures need a thorough, systematic review from the standpoint of the systems interaction issue. This is necessary in order (a) to assess potential design modifications; (b) to assess corrective programs; and (c) to minimize the safety risks due to interactions between safety and control systems.

VIII. OTHER METHODOLOGIES ARE AVAILABLE TO SUPPLEMENT THE DESIGN BASIS ANALYSIS APPROACH UTILIZED BY LILCO

LILCO has utilized the design basis analysis approach to classification of SS&Cs at Shoreham. This Section discusses the deficiencies of this approach -- particularly, its failure to identify potential systems interactions and its failure to classify appropriately all components which may be relied upon to mitigate design basis events.64/ This Section also

64/ The discussion herein of deficiencies in methodology is purposely kept brief to avoid repetition of earlier discussion.

demonstrates that there are alternative methodologies available by which the foregoing deficiencies can be largely eliminated, thereby improving the safety of the plant.

VIII.A: DEFICIENCIES IN CURRENT METHODOLOGY

Current review procedures for nuclear power plants, as contained in the recent revision of the NRC's Standard Review Plan (SRP), do not adequately address the issues of systems interaction and classification of equipment important to safety. Thus in a recent report to the NRC on systems interaction, Sandia Laboratories reported:

> [T]he review of the SRP identified several areas which might be characterized as soft spots. These were identified in the application of the methodology where (1) the potential cause for an interaction could be identified, (2) if it occurred, it would increase the likelihood of core damage, and (3) the potential was not explicitly covered in the SRP.[65]

The Sandia systems interaction report was also critical of present classification practices, stating:

> The safety classification of systems used in the SRP does not include all systems important with respect to systems interactions. The fault tree identified "non-safety" components, which if caused to fail by systems interactions would increase the likelihood of the plant being unable to perform one of the three functions covered in the study. Therefore, completeness

65/ NUREG/CR-1321, Phase I - Systems Interaction Methodology Applications Program, Nuclear Regulatory Commission, Washington, D.C., April 1980, p. 131 (emphasis supplied).

of any systems interaction study requires that systems traditionally labelled "not required for safety" be included in the models.66/

Actual events at operating reactors also have highlighted the limitations of reliance solely on a design basis accident or an SRP approach to detection and prevention of adverse systems interactions. For example, the TMI-2 accident involved several examples of systems interaction and human error. Similarly, in 1980, both Crystal River 3 and Browns' Ferry 3 suffered systems interaction events. Thus, these three plants all exhibited the potential for adverse systems interaction, despite having been subject to SRP Review.

Past experience also supports the need for a new means of classifying equipment. During one part of the Three Mile Island accident, the reactor core was cooled by the reactor coolant pumps. These pumps, not being "safety grade," had no backup power source to keep them operating in case of a loss of offsite power. Fortunately, the accident did not include loss

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66/ Ibid. The Sandia findings confirmed results reached earlier by the NRC's Lessons Learned Task Force which concluded:

> The interactions between non-safety-grade and safety-grade equipment are numerous, varied, and complex and have not been systematically evaluated.

NUREG-0585, TMI-2 Lessons Learned Task Force - Final Report, U.S. Nuclear Regulatory Commission, Washington, D.C., October, 1979, p. 3-3.

of offsite power. However, the TMI accident did illustrate that systems which are not subject to the most stringent quality and review criteria may nonetheless be called upon to prevent or mitigate the effects of nuclear accidents. This clearly is not desirable.

Alternative methods exist which would supplement and improve the existing design basis/SRP approach and thus reduce the likelihood of adverse systems interaction. These methods also would lead to more systematic classification of equipment which may be relied upon to mitigate the effects of an accident. These methods represent a significant improvement over reliance solely on the SRP and the present classification of equipment.

VIII.B: ADDITIONAL METHODS TO IDENTIFY POTENTIAL SYSTEMS INTERACTIONS

The NRC's GDC direct Applicants to consider systems interactions in design of nuclear facilities. Thus, Applicants must consider "the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems."[67/]

67/ 10 C.F.R. Part 50, Appendix A, Introduction, ¶ 4.

LILCO has not attempted systematically to identify potential systems interactions during the design and construction of Shoreham. While LILCO identified certain systems interaction efforts at the March 31 deposition,68/ these efforts were sporadic and in almost every case responded to a NRC request for more data. These efforts lacked in-depth, detailed analyses of the potential for interactions.

There are, however, methods to approach the systems interaction question which provide far greater assurance that most adverse interactions will in fact be identified. Three such currently available approaches to systems interaction are: dependency analysis, fault tree/event tree diagrams, and "walkdowns."

Dependency analysis looks at the various ways that components and systems depend upon one another. Some examples of this approach are binary matrices, failure modes and effects analyses (FMEA's), and auxiliary safety systems commonality diagrams.

Binary matrices are matrix representations of mechanical flow diagrams. They show which components in a system are functionally dependent upon other components. The entries in a

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68/ Deposition at 113, *et seq.* (Kascsak).

binary matrix are either zero (components are independent) or one (one component dependent on the other).

FMEA's describe, for a given component, the possible modes of failure, and which other components would be affected by each failure mode. For instance, a block valve might fail by sticking open, sticking closed, or leaking. Each mode of failure would be expected to cause differing problems which then would be analysed in detail in the FMEA.

Auxiliary safety systems commonality diagrams illustrate the commonalities (power source, lubrication, location, actuation, etc.) among redundant auxiliary systems. By defining the commonalities the reviewer is able to identify the potential for adverse interactions and to pursue design alternatives which minimize these effects. This technique is more specific than the two preceding techniques. Commonality diagrams can be applied to main safety systems, electrical systems, or to any system of special interest.

Using commonality diagrams, dependency matrices, and FMEA's, a reviewer can obtain a thorough understanding of the interactions possible in a power plant. That knowledge is best then applied through use of fault tree/event tree diagramming, such as is used in PRA (Probabilistic Risk Assessment).

Unfortunately, the PRA currently being done at Shoreham excludes systems interactions, as stated on page 1-18:

> [T]he results of this analysis are valid to the extent that the systems have been successfully designed as independent and redundant. Externalities and other internal initiators such as sabotage, seismic events, or fire have not been postulated as initiators.69/

Interactions were excluded because they have not been systematically identified at Shoreham -- by commonality diagrams, by dependency matricies, or by FMEAs.

This approach to the scope of a PRA -- that is, failing, as a first step, to identify potential interactions -- has been criticized by the ACRS, which wrote to the NRC in January, 1982,

> The Committee notes that the current probabilistic risk assessments (PRAs) do not usually include a systematic examination of systems interactions and improvements in safety and reliability.70/

A PRA recently completed at Indian Point 3 (IP-3) may require revision when the IP-3 Systems Interaction Review is complete.71/ Like the study currently being done at Shoreham,

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69/ Probabilistic Risk Assessment, Shoreham Nuclear Power Station, Preliminary Draft, p. 1-18.

70/ Transcript of February 3, 1981, ACRS Subcommittee meeting, "Plant Features Important to Safety," U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, Washington, D.C., p. 4.

71/ "Inside NRC," McGraw-Hill Publications, April 5, 1982, p. 9.

the IP-3 PRA did not address dependent failures and between-systems dependencies.

However, a PRA which uses the results of dependency analysis could be valuable in increasing plant safety and reliability. Such a PRA could successfully identify potential systems interactions which could result from system dependencies. The PRA would then be able to model fault trees and event trees for all aspects of the plant which might be affected by an accident or transient. This would result in a far more complete study than the Shoreham PRA which effectively excludes systems interaction.

The dependency analysis/PRA should be complemented by a physical survey or "walkdown," such as was performed at Diablo Canyon. Pacific Gas & Electric Co. (PG&E), owner of the Diablo Canyon Nuclear Plant, has undertaken a "walkdown" or physical survey program to investigate potential systems interactions at the plant. The program, conducted by multi-disciplinary teams of engineers, seeks as its end product the identification of SS&Cs which could adversely affect other plant systems during operation. Although the scope of PG&E's review is limited to seismic events (due to the proximity of that facility to the Hosgri Fault), PG&E has made 300 modifications to the plant as a result of its walkdown program.[72/] PG&E considers this review

72/ PG&E Letter to NRC, March 9, 1982

a very worthwhile venture, which will in time save much more than it will cost, and will also increase plant safety.[73] Walkdowns provide an opportunity to review and improve the actual, as-built plant.[74] The concrete knowledge gained from a walkdown is critical in checking the more abstract results of analytical reviews. Thus, the best method of resolving the systems interaction issue at any given plant is a combined approach, using dependency analysis/PRA, balanced by a walkdown study.

73/ Transcript of October 8, 1980 ACRS Subcommittee meeting, "Diablo Canyon Systems Interaction Study," Advisory Committee on Reactor Safeguards.

74/ GE stated in the March 31 deposition of Robare (at page 31) that it has performed "walkdowns" of the emergency systems at Shoreham. However, based on the description of those "walkdowns," they were in the nature of design or "as-built" verifications, where the lead systems engineers "visit the site to ensure that the as-built design is harmonious with the original design." A system review of this nature is a narrow view of the plant, focussed on a particular system that is of interest to a specific engineer.

A walkdown of the systems interaction type would involve extensive preparation in the office, including planning, plotting the locations of sources and targets, defining zones, and making out check lists. The actual walkdown would then be conducted by an interdisciplinary team of reviewers. This is considerably different from an inspection of the progress and accuracy of construction on a particular system or subsystem.

VIII.C: ADDITIONAL CATEGORIES OF CLASSIFICATION

The classification of equipment at Shoreham could also be improved significantly if additional methodologies were used. First, the systems interaction methodologies just described would identify accident sequences and equipment which is not considered in the traditional design basis analysis approach. The use of such methodologies would enable LILCO, before an accident, to reclassify equipment or take other safety precautions to ensure that plant operation complies with the regulations.

Further, the Shoreham equipment classifications require additional revision beyond the use of the interaction analyses described above. Shoreham equipment is allegedly classified according to regulatory guidance and traditional industry practice.[75/] That is, safety-related equipment at Shoreham is classified by methods which have been used in the industry for the past few years at least.[76/] The Shoreham classifications are safety-related (safety grade, important to safety) and non-safety-related (not important to safety).[77/]

75/ Deposition 28-31, 78 (Robare, Dawe).

76/ Deposition at 23 (Robare).

77/ Deposition at 71-76 (Dawe).

Several authoritative groups in the nuclear regulatory field agree that the traditional classification of equipment is inadequate in two respects. First, the dichotomy of safety/non-safety equipment does not account for the fact that some equipment is in-between, requiring significant assurance of reliability but not quite safety grade. Second, certain equipment has historically been classified as non-safety-grade even though it may be necessary to prevent or mitigate accidents, or its failure could adversely affect safety systems.

The ACRS has termed the safety/non-safety classification as an "all or nothing" approach to safety assurance, and has urged the NRC to change the approach.[78/] The NRC is working on a proposed rule to expand the list of components, structures, and systems subject to 10 C.F.R. Part 50, Appendix B Criteria.[79/] Finally, the IEEE is currently preparing Standard P827, aimed at standardizing a new approach to classification of instrumentation, controls, and electrical equipment.[80/]

78/ Transcript of February 3, 1981, ACRS Subcommittee meeting, "Plant Features Important to Safety," U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, Washington, D.C., p. 4

79/ NUREG-0660, NRC Action Plan Developed as a Result of the TMI Accident, Item I.F.1.

80/ IEEE Trial Use Guide: A Method for Determining Requirements for Instrumentation, Control, and Electrical Systems and Equipment Important to Safety, Draft, January 1981.

Basically, the NRC, ACRS, and IEEE appear to have decided that changes are needed in the classification of equipment. The intent of all these groups seems to be to create a designation between safety and non-safety-related. Equipment in this class would be more stringently designed, qualified, tested, and inspected than when classified as non-safety-related, thus ensuring greater reliability if, in an accident scenario, operators were required or chose to rely upon it.

Two procedures are available for assigning systems to the intermediate category. The first is straight-forward. Safety studies such as the PRA, systems interaction studies, operating procedures, Chapter 15 of the FSAR, and so on, should be reviewed to identify any non-safety equipment which actually affects safety or which is relied upon to mitigate Chapter 15 transients or accidents. The systems interaction studies could be especially effective in identifying equipment having previously unsuspected impact on plant safety. Once such equipment has been identified, LILCO will be in a position, based upon adequate knowledge, to determine what additional requirements (qualification, QA, etc.) should be applied to particular equipment to ensure reliable operation in an emergency.

Alternatively, one could use the approach being developed by the IEEE P827 working group. This method involves setting a safety goal in terms of probability of a consequence (such as 10^{-6} early deaths per reactor year), then reviewing each piece of equipment, using PRA techniques, to determine the contribution made by that equipment to the safety goal.[81/] Equipment making a significant contribution to safety, but classified as non-safety, would be reclassified.

IX. CONCLUSION

The General Design Criteria (GDC) for nuclear power plants establish criteria which are used to ensure the quality and qualification of components which are important to safety. Thus, the GDC establish requirements for quality standards and records commensurate with the importance of the safety function to be performed (GDC-1) and specify that components important-to-safety be designed for accident environments (GDC-4). Absent a systematic and thorough method of component classification, there is no assurance that the full and necessary set of SS&Cs has been subjected to the requirements of these GDC or, conversely, that compliance with the GDC has been assured.

81/ Ibid.

The GDC also establish criteria to guide the design, redundancy, separation and analysis of SS&Cs importance to safety in order to ensure their safety function will be accomplished. These include design for protection against the effects of natural disasters, fires and missiles (GDC-2, 3 and 4). These conditions must be applied to the design and analysis of components to be considered and the possible interactions resulting from the accident initiators of concern.

Three of the GDC concern themselves with the necessity of redundancy and/or design features to protect against vulnerability to single failures in protection systems (GDC-21), the diversity of protection system design (GDC-22), and the assurance of failsafe modes for protection systems (GDC-23). Other GDC address the need for careful design of reactor protection and control systems to ensure their separation and thus prevent unfavorable interactions (GDC-24) and assurance that the reactivity control and protection systems will be capable of performing their functions under all anticipated operational occurrences (GDC-29). For each of these GDC, it is necessary to assure that the classification of systems is accurately and systematically performed to identify all systems which should be included and their independence from interactions with other systems.

There are criteria for fuel temperature limits which must be met to ensure integrity of the fuel cladding under worst case conditions. These cover the design of the reactor and the ECCS and the necessity that cooling water sources be available under the most adverse conditions (GDC-10 and 35), plus the need for testing of ECCS under a range of conditions and power sources (GDC-37). Part of the protection system, cooling system and testing system requirements is that there be a thorough set of instrumentation and control for the range of accident conditions (GDC-13). There is a clear need to have a full description of the safety-related components which must be tested and the system conditions which could interact with the emergency systems to create adverse operation conditions.

It is not possible to find that Shoreham has met the above criteria until there has been a systematic analysis to identify the SS&Cs necessary for important to safety and safety-related functions. This analysis must also include a systems interaction analysis of Shoreham to find all systems and components whose actions may have importance to safety.

The Applicant's FSAR, together with the Staff's SER, are supposed to consistute the definitive documents in support of licensing a nuclear plant. At Shoreham these documents are deficient and do not in fact provide a basis for issuance of an operating license. Thus, as documented in preceeding sections:

-- The Shoreham classification system is not consistent with the GDC or even with the Regulatory Guides upon which it allegedly is based;

-- The EOPs direct operators to utilize equipment which has not been classified or qualified commensurate with the functions performed;

-- The problems of systems interactions have not been systematically analysed, as exemplified by recent water level indicator problems; and

-- LILCO has failed to supplement its analyses with alternative methodologies that would assist classification of equipment and identify adverse systems interactions.

This Board, accordingly, should find that LILCO has failed to demonstrate compliance with he NRC's regulations.

EXHIBIT 1

Memorandum, dated November 20, 1981, from Harold R. Denton to all NRR Personnel re: Standard Definition for Commonly-Used Safety Classification Terms.

ATTACHMENT A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 2 0 1981

MEMORANDUM FOR: All NRR Personnel

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: STANDARD DEFINITIONS FOR COMMONLY-USED SAFETY CLASSIFICATION TERMS

Litigation of one of the principal issues in the TMI-1 Restart Hearing brought to light the fact that there is not complete consistency among all elements of the NRR staff in the application of safety classification terms used frequently in the conduct of NRR's safety review and licensing activities. More specifically, it appears that terms "important to safety," "safety grade," and "safety-related" have been used at times interchangeably, or in ways not completely consistent with the definitions and usage of such terms in the regulations, and which do not fully reflect the intent of the regulations or current licensing practice.

Efforts have been underway for some months now to develop guidance for the consistent usage of these terms. These efforts have included: (a) review of a large number of Reg Guides and SRP's, in conjunction with parts of the regulations upon which they are based, for consistency in the application of safety classification terminology, (2) extensive discussions among cognizant NRR, RES (Stds. Devel.) and ELD representatives regarding proper interpretation and application of such terms, including consideration of alternative "standard" definitions and (3) consultation with the cognizant ACRS Subcommittee regarding these matters, and consideration by the full ACRS as well.

As a result of these efforts, I am endorsing and prescribing for use by all NRR personnel the standard definitions set forth in the enclosure to this letter. It should be noted that in connection with long-term efforts to develop means for ranking reactor plant systems with respect to degree of importance to safety, and in connection with related efforts to develop a graded Q.A. approach in reactor licensing, the general question of safety classifications and safety classification terminologies will be reexamined; and this could result in changes to the definitions set forth in the enclosure or perhaps in development of a completely new scheme in this regard. For the time being, however, the definitions in the enclosure should be considered "standard" and should be applied consistently by all NRR personnel in all aspects of our safety review and licensing activities and should be appropriately reflected in our regulatory guidance documents.

DEFINITION OF TERMS

Important to Safety

- Definition - From 10 CFR 50, Appendix A (General Design Criteria) - see first paragraph of "Introduction."

  "Those 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."

- Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in important way to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation).

- Includes Safety-Grade (or Safety-Related) as a subset.

Safety-Related

- Definition - From 10 CFR 100, Appendix A - see sections III.(c), VI.a.(1), and VI.b.(3).

  Those structure, systems, or components designed to remain functional for the SSE (also termed 'safety features') necessary to assure 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 off-site exposures comparable to the guideline exposures of this part.

- Subset of "Important to Safety"

- Regulatory Guide 1.29 providesan LWR-generic, function-oriented listing of "safety-related" structures, systems, and components needed to provide or perform required safety functions. Additional information (e.g., NSSS type, BOP design A-E, etc.) is needed to generate the complete listing of safety-related SSC's for any specific facility.

  Note: The term "safety-related" also appears in 10 CFR 50, Appendix B (Q.A. Program Requirements); however, in that context it is framed in somewhat different language than its definition in 10 CFR 100, Appendix A. That difference in language between the two appendices has contributed to confusion and misunderstanding regarding the exact meaning of "safety-related" and its relationship to "important to safety" and "safety-grade." A revision to the language of Appendix B has been proposed to clarify this situation and remove any ambiquity in the meaning of these terms.

It is expected that minor editorial revisions will have to be made to some existing Reg Guides and SRP's in order to make their wording consistent with these definitions. You should review the regulatory guidance documents within your purview in this regard and recommend the necessary changes; it is not expected that this will involve extensive revision efforts. I want to make clear that my interest here is only in establishing consistency in the language used by all cognizant groups within NRR in expressing our technical requirements. It is not my intention by this action to dictate new technical requirements, to modify existing technical requirements, or to broaden the existing scope of NRR licensing review.

Harold Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Definition of Terms

<u>Safety-Grade</u>

- Term not used explicitly in regulations but widely used/applied by staff and industry in safety review process.
- Equivalent to "Safety-Related," i.e., both terms apply to the same subset of the broad class "Important to Safety."

EXHIBIT 2

FSAR Section 3.2, including Table 3.2.1-1 (reflecting revisions set forth in April 8, 1982 letter from T.S. Ellis to Larry Lanpher, see Exhibit 3 below), Table 3.2.1-2, Table 3.2.1-3, and Figure 3.2.2-1.

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 Seismic Classification

Seismic Category I structures, systems, and components are those necessary to ensure:

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 exposure of 10CFR100.

Seismic Category I structures, systems, and components are designed to remain functional during a design basis earthquake (DBE). Strain limits in excess of yield are allowed provided safety functions are maintained. Seismic Category I structures, systems, and components are [illegible] designed to be within the elastic limit for an operating [illegible] earthquake (OBE) having a vibratory motion of 50 percent [illegible] the DBE.

The DBE and the OBE are described in Section 2.5. The seismic design of Seismic Category I systems and components is described in Sections 3.7 and 3.10 and of structures in Sections 3.7 and 3.8. Seismic Category I structures, systems and components are listed in Table 3.2.1-1.

The structures, systems, and components listed in the Regulatory Position of Regulatory Guide 1.29 are Seismic Category I, as applicable, and are listed in Table 3.2.1-1.

3.2.2 System Quality Group Classifications

System quality group classifications have been determined for each component of (1) those applicable fluid systems relied upon to prevent, or mitigate the consequences of, accidents or malfunctions originating within the reactor coolant pressure boundary, or to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (2) other associated safety related systems. Regulatory Guide 1.26, "Quality Group Classifications and Standards," was used in assigning (1) quality group classifications and (2) design and fabrication requirements as shown in Tables 3.2.1-1, 2, and 3.

Figure 3.2.2-1 is a diagram which depicts the relative location of components along with their associated quality group classification.

3.2.2.1 Exceptions to Quality Group Classification

3.2.2.1.1 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system is designed as two separate subsystems: the fuel pool cooling subsystem and the fuel pool cleanup subsystem. The fuel pool cleanup subsystem and its components have no safety functions and are classified Quality Group "D". The fuel pool cooling subsystem complies with Regulatory Guides 1.13, 1.26, and 1.29. All headers and connections common with the fuel pool cooling subsystem are Quality Group C. The fuel pool cooling and cleanup system and the interconnections with other systems are discussed in Section 9.1.3. Its instrumentation and controls required for safety are discussed in Section 7.6.

3.2.3 Quality Assurance

Structures, systems, and components whose safety functions require conformance to the quality assurance requirement of 10CFR50, Appendix B, are summarized in Table 3.2.1-1 under the heading, LILCO Quality Assurance Category.

TABLE 3.2.1-1

EQUIPMENT CLASSIFICATION

Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
I. Reactor System								
1. Reactor vessel	GE	PC	A	I	I	2/67	ASME III A, 1965 Winter 66	
2. Reactor vessel support skirt	GE	PC	-	I	I	2/67	ASME III A, 1965 Winter 66	
3. Reactor vessel appurtenances, pressure retaining portions	GE	PC	A	I	I	12/69	ASME III A, 1965 Winter 66	
4. CRD housing supports	GE	PC	-	I	I		X	
5. Reactor internal structures, engineered safety features	GE	PC	A	I	I		X	
6. Core support structures	GE	PC	-	I	I	2/67	X	
7. Other internal structures								
a. Shroud head & separator assembly	GE	PC	-	~~I~~ II	~~I~~ NA		X	
b. Dryers	GE	PC	-	II	~~I~~ NA		X	(9)
8. Control rods	GE	PC	-	I	I		X	
9. Control rod drives	GE	PC	-	I	I		X	
10. Power range detector hardware	GE	PC	B	I	I		ASME III-2	
11. Fuel assemblies	GE	PC	-	I	I		X	
12. Reactor vessel stabilizer	GE	PC	-	I	I		X	
13. Reactor vessel star truss	P	PC	-	I	I		X	
14. Reactor vessel insulation	GE	PC	-	II	NA		X	
II. Nuclear Boiler System								
1. Vessels, instrumentation condensing chambers	GE	PC	A	I	I		ASME III-1	
2. Vessels, air accumulators	P	PC	B	I	I		ASME III-2	

TABLE 3.2.1-1 (CONT'D)

Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
3. Piping, relief valve discharge (including ramshead and supports)	P	PC	C	I	I		ASME III-3	
4. Piping, main steam within outer isolation valve	GE	PC	A	I	I	11/69	B31.1.0	
5. Pipe supports, main steam within outer isolation valve	GE	PC	~~A~~	I	I	1/75	ASME III	
6. Pipe whip restraints, main steam	P	PC	-	I	I		X	
7. Piping feedwater, within outermost isolation valves	P	PC,RB	A	I	I		ASME III-1	
8. Other primary coolant pressure boundary piping within isolation valves	P	PC	A	I	I		ASME III-1	
9. Piping, instrumentation beyond outermost isolation valves	P	RB	D	II	NA		See Note 8	
10. Safety/Relief Valves	GE	PC	A	I	I	12/69	ASME I, III, & IX 1968 Winter	
11. Valves, main steam isolation valves	GE	PC,RB	A	I	I	10/69	B31.1.0/ASME VIII	
12. Valves, feedwater isolation valves and within	P	PC,RB	A	I	I		ASME III-1	(10)
13. Valves, other, isolation valves and within	P	PC,RB	A	I	I		ASME III-1	
14. Valves, instrumentation beyond outermost isolation valves	P	RB	D	II	NA		See Note 8	
15. Electrical modules with safety function	GE	PC	-	I	I		X	
16. Cable, with safety function	P	-	-	I	NA		X	
III. Recirculation System								
1. Piping	GE	PC	A	I	I	10/69	B31.1.0	
2. Piping suspension, recirculation line	GE	PC	-	I	I	12/74	ASME III	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope of Supply[2]	Location[3]	Quality Group Classification[4a]	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	3. Pipe restraints, recirculation line	GE	PC	-	I	I			
	4. Pumps	GE	PC	A	I	I	11/69	ASME IIIC	
	5. Valves	GE	PC	A	I	I	10/69	ASME VIII/MSS Sp66	
	6. Pump motor	GE	PC	-	~~II~~ I	I	10/69	X	
	7. Electrical modules, with safety function	GE	RB,R	-	I	I		X	
	8. Cable with safety function	P	-	-	I	NA		X	
IV.	CRD Hydraulic System								
	~~1. Valves, isolation, water return line~~	~~P~~	~~PC,RB~~	~~A~~	~~I~~	~~I~~		~~ASME III-1~~	
	2. Valves, scram discharge volume lines	GE	RB	B	I	I	12/69	B31.1.0	
	3. Valves insert and withdraw lines	P	RB	B	I	I		ASME III-2	(11)
	4. Valves, other	P	RB	D	II	NA		B31.1.0	
	~~5. Piping, water return line within isolation valves~~	~~P~~	~~PC~~	~~A~~	~~I~~	~~I~~		~~ASME III-1~~	
	6. Piping, scram discharge volume lines	P	RB	B	I	I		ASME III-2	
	7. Piping, insert and withdraw lines	P	PC,RB	B	I	I		ASME III-2	
	8. Piping other	P	RB	D	II	NA		B31.1.0	
	9. CRD pumps, filters, and strainers	GE	RB	D	II	NA	1970	X	
	10. Hydraulic control unit	GE	RB	-	I	I		Special	(12)
	11. Cable, with safety function	P	-	-	I	NA		X	
V.	Standby Liquid Control System								
	1. Standby liquid control tank	GE	RB	B	I	I	2/74	ASME II, III, IX & API 620/650	
	2. Pump	GE	RB	B	I	I	12/69	B31.1.0 & HIS	
	3. Pump motor	GE	RB	-	I	I	12/69	X	
	4. Valves, explosive	GE	RB	B	I	I	12/67	ASME VIII-1	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope of Supply[2]	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	5. Valves, isolation and within	P	RB	A	I	I		ASME III-1	
	6. Valves, beyond isolation valves	P	RB	B	I	I		ASME III-2	
	7. Piping, within isolation valves	P	PC,RB	A	I	I		ASME III-1	
	8. Piping, beyond isolation valves	P	RB	B	I	I		ASME III-2	
	9. Electrical modules, with safety function	GE	RB,R	-	I	I		X	
	10. Cable, with safety function	P	-	-	I	NA		X	
	11. Accumulators	GE	RB	C	I	I	8/71	See Note 13	(13)
	12. Valves and piping other	P	RB	D	~~NA~~ II	NA		B31.1.0	
	13. Drain tank	P	RB	D	~~NA~~ II	NA		ASME VIII	
VI.	Neutron Monitoring System								
	1. Piping, TIP	GE	PC	D	II	I		B31.1	
	2. Valves, isolation, TIP subsystem	GE	RB	B	I	I		ASME III-2	
	3. Electrical modules, IRM and APRM	GE	RB,R	-	I	I		X	
	4. Cable, IRM and APRM	P	-	-	I	NA		X	
VII.	Reactor Protection System								
	1. Electrical modules	GE	PC,RB,R,T	-	I	I		X	
	2. Cable	P	-	-	I	NA		X	
VIII.	Fixed Process, Airborne, and Effluent Radiation Monitors								
	1. Electrical modules, main steam line and reactor building ventilation monitors	GE	RB,R,T	-	I	I		X	
	2. Cable, main steam line and reactor building ventilation monitors	P	-	-	I	NA		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	3. Electrical modules for process liquid, process ventilation, air ejector offgas, and offgas treatment radiation monitoring systems	GE	R,T,RW,RB	–	II	NA		X	
		P	R,T,RWW,RB	–	II	NA		X	
	4. Electrical modules for effluent radiation monitoring system	P	R,T,RW,RB	–	I & II	I & NA	9/76		
IX.	RHR System								
	1. Heat exchangers, primary side	GE	RB	B	I	I	8/69	ASME III-C & TEMA C	
	2. Heat exchangers, secondary side	GE	RB	C	I	I	8/69	ASME VIII & TEMA C	
	3. Piping, connected to RCPB within outermost isolation valves	P	PC, RB	A	I	I		ASME III-1	
	4. Piping, ~~beyond outermost isolation valves~~ (other)*	P	RB, PC	B	I	I		ASME III-2	
	5. Pumps	GE	RB	B	I	I	8/69	B31.1.0/ ASME III-C	
	6. Pump motors	GE	RB	–	I	I	10/69	X	
	7. Valves, isolation, LPCI and shutdown lines	P	PC,RB	A	I	I		ASME III-1	
	8. Valves, isolation, other	P	PC,RB	B	I	I		ASME III-2	
	9. Valves, beyond isolation valves	P	RB	B	I	I		ASME III-2	
	10. Electrical modules, with safety function	GE	RB,R	–	I	I		X	
	11. Cable, with safety function	P	–	–	I	NA		X	
X.	Core Spray System								
	1. Piping connected to RCPB Piping, within outermost isolation valves	P	PC, RB	A	I	I		ASME III-1	

* everything except piping connected to RCPB

TABLE 3.2.1-1 (CONT'D)

Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
2. Piping, ~~beyond outermost isolation valves~~ (other)*	P	RB, PC	B	I	I		ASME III-2	
3. Pumps	GE	RB	B	I	I	9/69	B31.1.0/ ASME III-C	
4. Pump motors	GE	RB	-	I	I	7/69	X	
5. Valves, isolation and within connected to RCPB	P	PC, RB	A	I	I		ASME III-1	
6. Valves, ~~beyond outermost isolation valves~~ (other)*	P	RB	B	I	I		ASME III-2	
7. Electrical modules with safety function	GE	RB, R	-	I	I		X	
8. Cable, with safety function	P	-	-	I	NA		X	
XI. HPCI System								
1. Piping, within outermost isolation valves	P	PC, RB	A/B	I	I		ASME III-1, ASME III-2 (28)	
2. Piping beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	
3. Piping return test line to condensate storage tank beyond reactor building	P	O	D	II	NA		B31.1.0	
4. Vacuum pump discharge line	P	RB	B	I	I		ASME III-2	
5. Pump	GE	RB	B	I	I	6/69	B31.1.0/ ASME III-C	
6. Valves, isolation and within	P	PC, RB	A/B	I	I		ASME III-1, III-2 (28)	
7. Valves, return test line to condensate storage	P	RB	B	I	I		ASME III-2	
8. Valves, other	P	RB	B	I	I		ASME III-2	
9. Turbine	GE	RB	-	I	I	6/69		(14)
10. Electrical modules with safety function	GE	RB, R	-	I	I		X	
11. Cable, with safety function	P	-	-	I	NA		X	

* Everything except piping connected to RCPB

(28) Component is quality group A when connected to RCPB

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XII.	RCIC System								
	1. Piping, within outermost isolation valves	P	PC, RB	A/B	I	I		ASME III-1, III-2	(28)
	2. Piping, beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	
	3. Piping, return test line to condensate storage tank beyond reactor building	P	O	D	II	NA		B31.1.0	
	4. Vacuum pump discharge line from vacuum pump to containment isolation valves	P	RB	B	I	I		ASME III-2	
	5. Pump	GE	RB	B	I	I	6/69	B31.1.0/ ASME III-C	
	6. Valves, isolation and within	P	PC, RB	A/B	I	I		ASME III-1, III-2	(28)
	7. Valve, return test line to condensate storage	P	RB	B	I	I		ASME III-2	
	8. Valves, other	P	RB	B	I	I		ASME III-2	
	9. Turbine	GE	RB, R	-	I	I	6/69		(14)
	10. Electrical modules, with safety function	GE	RB, R	-	I	I		X	
	11. Cable, with safety function	P	-	-	I	NA		X	
XIII.	Fuel Service Equipment								
	1. Fuel preparation machine	GE	RB	-	I	I		X	
	2. General purpose grapple	GE	RB	-	I	I		X	
XIV.	Reactor Vessel Service Equipment								
	1. Steam line plugs	GE	RB	-	I	I		X	
	2. Dryer and separator sling and head strongback	GE	RB	-	I	I		X	

(28) Component is quality group A when connected to RCPB.

TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
	3. Drywell head lifting rig	P	RB	-	I	I		X	
XV.	In-Vessel Service Equipment								
	1. Control rod grapple	GE	RB	-	I	I		X	
XVI.	Refueling Equipment								
	1. Refueling platform	GE	RB	-	I	I	4/71	AISC	
	2. Refueling bellows, drywell	GE	PC	-	II	NA		X	
	3. Refueling bellows, reactor cavity	P	RB	-	II	See Note (15)		X	(15)
	4. New fuel inspection stand	GE	RB	-	II	NA		X	
XVII.	Storage Equipment								
	1. Fuel storage racks	GE	RB	-	I	I		X	
	2. Defective fuel storage container	GE	RB	-	I	I		X	
	3. Spent fuel pool, dryer/sep. pool, Rx cavity liners	P	RB	-	I	I		X	
XVIII.	Radwaste System								
	1. Tanks, Atmospheric	P	RW	D	II	NA		X	RG 1.143
✓	2. Heat exchangers	P	RW	D	II	NA		ASME VIII	
	3. Piping, containment isolation	P	PC, RB	B	I	I		ASME III-2	
	4. Valves, containment isolation	P	PC,RB	B	I	I		ASME III-2	
✓	5. Piping, other	P	RB,O,T,RW	D	II	NA		B31.1.0	
	6. Pumps	P	RB,RW	D	II	NA		X	
	7. Valves, flow control and filter system	P	RW	D	II	NA		B31.1.0	
	8. Valves, other	P	RB,RW	D	II	NA		B31.1.0	(26)

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XIX.	Reactor Water Cleanup System								
	1. Vessels: filter/ demineralizer	GE	RB	C	II	NA	3/70	ASME III-C	
	2. Heat exchangers	GE	RB	C	I	I	12/69	ASME III-C/ TEMA R	(16)
	3. Piping within outermost isolation valves	P	RB	A	I	I		ASME III-1	
	4. Piping, bey nd outermost isolation valves	P	RB	C	I	I		ASME III-3	
	5. Pumps	GE	RB	C	II	NA	12/69	ASME III-C/ B31.1.0	(16)
	6. Valves, isolation valves and within	P	RB	A	I	I		ASME III-1	
	7. Valves, beyond outermost isolation valves	a. GE	RB	C	II	I	12/69	B31.1.0	
		b. P	RB	C	I	I		ASME III-3	
XX.	Fuel Pool Cleanup Subsystem								
	1. Demineralizer vessel	P	RW	D	II	NA		ASME VIII	
	2. Filters	P	RW	D	II	NA		ASME VIII	
	3. Pumps, purification	P	RB	D	II	NA		X	
	4. Piping	P	RB,RW	D	II	NA		B31.1.0	
	5. Valves	P	RB,RW	D	II	NA		B31.1.0	
XXI.	Fuel Pool Cooling Subsystem								
	1. Pumps, cooling	P	RB	C	I	I		ASME III-3	
	2. Heat exchangers	P	RB	C	I	I		ASME III-3	
	3. Piping	P	RB	C	I	I		ASME III-3	
	4. Valves	P	RB	C	I	I		ASME III-3	
XXII.	Control Room Panels								
	1. Electrical modules, with safety function	a. P	R	-	I	I		X	
		b. GE	R	-	I	I		X	
	2. Cable, with safety function	a. P	-	-	I	NA		X	
		b. GE	-	-	I	NA		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
XXIII.	Local Panels								
	1. Electrical modules, with safety function a.	P	RB	-	I	I		X	
	b.	GE	RB	-	I	I		X	
	2. Cable, with safety function	P	-	-	I	NA		X	
	3. Remote shutdown panel	GE	RB	-	I	I		X	
XXIV.	Offgas System								
	1. Atmospheric glycol tanks	P	T	D	II	NA		X	
	2. Heat exchangers	P	T	D	II	NA		ASME VIII	
	3. Piping	P	T,RW	D	II	NA		B31.1.0	
	4. Valves, flow control	P	T,RW	D	II	NA		B31.1.0	
	5. Valves, other	P	T,RW	D	II	NA		B31.1.0	
	6. Steam jet air ejectors	P	T	D	II	NA		ASME VIII	
	7. Charcoal vessels	P	RW	D	II	NA		ASME VIII	
	8. Recombiners	P	T	D	II	NA		ASME VIII	
	9. Filters	P	RW	D	II	NA		ASME VIII	
XXV.	Service Water System								
	1. Piping, Safety related	P	RB,O,P,R	C	I	I		ASME III-3	
	2. Piping, other	P	P,O,R,T	D	II	NA		B31.1.0	
	3. Pumps, safety related	P	P	C	I	I		ASME III-3	
	4. Pump motors, safety related	P	P	-	I	I		X	
	5. Valves, ~~[illegible]~~ safety related	P	P,R,RB	C	I	I		ASME III-3	
	6. Valves, other	P	T,O,P,R	D	II	NA		B31.1.0	
	7. Electrical modules, with safety function	P	R,P	-	I	I		X	
	8. Cable, with safety function	P	-	-	I	NA		X	
	9. pumps (other)	P	P	D	II	NA			
	10. motors (other)	P	P	—	II	NA			
XXVI.	Compressed Air System								
	1. Vessels, accumulators, supporting safety-related systems	P	PC,RB	C	I	I		ASME III-3	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	2. Piping in lines between accumulators and safety-related systems	P	PC,RB	C	I	I		ASME III-3	
	3. Valves in lines between accumulators and safety-related systems	P	PC,RB	C	I	I		ASME III-3	
	4. Piping, containment isolation	P	PC,RB	B	I	I		ASME III-2	
	5. Valves, containment isolation	P	PC,RB	B	I	I		ASME III-2	
	6. Electrical modules with safety function	P	PC,RB,R	-	I	I		X	
	7. Cables with safety function	P	-	-	I	NA		X	
	8. Valves and piping, other	P	-	D	II	NA		B31.1.0	
XXVII	Onsite Power Systems								
a.	Diesel Emergency Power Systems								
	1. Day tanks	P	R,O	-	I	I		ASME III-3	
	2. Piping, fuel oil system	P	R,O	-	I	I		ASME III-3	
	3. Valves, fuel oil system	P	R,O	-	I	I		ASME III-3	
	4. Pumps, fuel oil system	P	R,O	-	I	I		ASME III-3	
	5. Pump motors, fuel oil system	P	R,O	-	I	I		X	
	6. Diesel-generators	P	R,O	-	I	I		X	
	7. Electrical modules with safety functions	P	R,O	-	I	I		X	
	8. Cable, with safety functions	P	R,O	-	I	NA		X	
	9. Diesel fuel storage tanks	P	/O	-	I	I		ASME III-3	
	10. Diesel air compressors	P	R	-	I	I		X	
b.	A/C Power Systems, safety related								
	1. 4160 V switchgear	P	R	-	I	I	3/73	X	
	2. 480 V switchgear	P	R	-	I	I	1/74	X	
	3. 480 V MCC	P	R,RB,P	-	I	I	4/74	X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope of Supply[2]	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	4. Cables (instrumentation control and power)	P	-	-	I	NA	-	X	
	5. Transformers	P	R,RB,P,~~PH~~	-	I	I	1/76	X	
c.	Containment electrical penetration and assemblies	P	RB	-	I	I	1/75	ASME III RG 1.63	
d.	Fire stops	P	-	-	I	~~NA~~ I	2/80	X	
e.	DC power systems								
	1. 125 V batteries battery charges	P	R	-	I	I	11/74 3/75	X X	
	2. Cables	P	-	-	I	NA	-	X	
	3. Battery racks	P	R	-	I	I	11/74	X	
	4. Protective relays and control panels	P	R	-	I	I	~~11/75~~ 4/75	X	
XXVIII.	Primary Containment Atmospheric Control System								
	1. Piping	P	RB, PC	-	I	I		ASME III-2	
	2. Valves	P	RB, PC	-	I	I		ASME III-2	
	3. Fans	P	RB	-	I	I		X	
	4. Hydrogen recombiners	P	RB	-	I	I		X	
	5. Electrical modules with safety functions	P	RB	-	I	I		X	
	6. Cables with safety function	P	-	-	I	NA		X	
XXIX.	Reactor Building Standby Ventilation System								
	1. Ducting and isolation valves with safety function, including filter trains	P	RB	-	I	I		X	
	2. Blowers	P	RB	-	I	I		X	
	3. Unit coolers	P	RB	-	I	I		X	(17)
	4. Chilled water system	P	RB,R	-	I	I		X	(18)

TABLE 3.2.1-1 (CONT'D)

Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
5. Electrical modules with a safety function	P	RB	–	I	I		X	
6. Cable with a safety function	P	–	–	I	NA		X	
XXX. Primary Containment Purge System								
1. Containment isolation valves and associated piping	P	PC,RB	~~B~~ –	I	I		ASME III-2	
2. All other components	P	RB	–	II	NA		X	
XXXI. Power Conversion System								
1. Main steam piping between outermost isolation valves up to turbine stop valves	P	RB,T	B	I	I		ASME III-2	
2. Main steam branch piping to 1st valve capable of timely actuation	P	T	B	I	I		ASME III-2	
3. Main turbine bypass piping up to bypass valve	P	T	B	I	I		ASME III-2	
4. First valve that is either normally closed or capable of automatic closure in branch piping connected to main steam and turbine bypass piping	P	T	B	I	I		ASME III-2	
5. Turbine stop valves, turbine control valves and turbine bypass valves	P	T	D	II	NA		Special	(19)
6. Main steam leads from turbine control valve to turbine casing	P	T	D	II	NA		Special	(19)

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	7. Feedwater and condensate system beyond 3rd isolation valve	P	RB,T	D	II	NA		B31.1.0	(10)
XXXII.	Condensate Storage and Transfer System								
	1. Condensate storage tank	P	O	D	II	NA		API-650	(20)
	2. Piping, suction line to HPCI, RCIC	P	O,RB	B	I	I		ASME III-2	
	3. Piping & Valves other	P	O	D	II	NA		B31.1.0	
	4. Other components	P	O	D	II	NA		X	
XXXIII.	Permanent Emergency Support Facilities								
	1. Technical Support Center (TSC)	P	TSC Bldg.	-	II	I	Δ	X	Building Seismic, Instruments Nonseismic
	2. Emergency Operations Facility (EOF)	P	*	-	II	NA	Δ	X	
	3. Operational Support Center (OSC)	P	*	-	II	NA	Δ	X	
	~~4. Safety Parameter Display System (SPDS)~~	~~P~~	~~R,TSC Bldg.~~		~~I~~	~~I~~	~~Δ~~	~~X~~	
XXXIV.	Main Steam Isolation Valve Leakage Control System								
	1. Piping and Valves up to the first isolation valve of the inboard subsystem	P	RB	A	I	I		ASME III-1	
	2. Piping and Valves to second isolation valves	P	RB	B	I	I		ASME III-2	
	3. Blowers	GE	RB	~~NA~~ —	I	I		X	
	4. Heaters	GE	RB	~~NA~~ —	I	I		X	
	5. Cable with Safety Function	P	-	~~NA~~ —	I	~~I~~ NA		X	
	6. Electrical Modules with Safety Functions	GE	RB	~~NA~~ —	I	I		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XXXV.	Miscellaneous Components								
	1. Reactor building polar crane	P	RB	-	I	I		X	
	2. ECCS loop level pumps	P	RB	B	I	I		ASME III-2	
XXXVI.	Reactor Building Closed Loop Cooling Water System								
	1. Pumps and heat exchangers	P	RB	C	I	I		ASME III-3	
	2. Valves, containment isolation	P	PC, RB	B	I	I		ASME III-2	
	3. Piping and valves for spent fuel pool HX; Reactor Recirc. pump cooler, ECCS pump coolers	P	RB	C	I	I		ASME III-3	
	4. Pumps and piping for motor generator MG set coolers	P	R, RB	D	II	NA		B31.1.0	
	5. Piping, other	P	PC, RB	D	II	NA		B31.1.0	
	6. Valves, other	P	PC, RB	D	II	NA		B31.1.0	
	7. Recirc Pump Cooler inline Flow Meter	GE	PC	-	II	NA			(27)
XXXVII.	Equipment and Floor Drainage Systems								
	1. Sumps	P	RB, T, RW	D	II	NA		X	
	2. Pumps	P	RB, T, RW	D	II	NA		X	
	3. Piping, containment isolation	P	RB, PC	B	I	I		ASME III-2	
	4. Valves, containment isolation	P	RB, PC	B	I	I		ASME III-2	
	5. Cable, with a safety function	P	-	-	I	NA		X	
	6. Piping, other	P	RB, T, RW	D	II	NA		B31.1.0	
	7. Valves, other	P	RB, T, RW	D	II	NA		B31.1.0	
XXXVIII.	Miscellaneous Ventilation Systems								
	1. Battery room H & V[22]	P	R	-	I	I		X	

Safety related

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope of Supply[2]	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	2. Screenwell pumphouse H & V	P	P	-	I	I		X	
	3. Relay and emergency switchgear H & V	P	R	-	I	I		X	
	4. Control room air conditioning, including filter trains	P	R	-	I	I		X	
	5. Diesel generator room ventilation	P	R	-	I	I		X	
XXXIX.	Area Radiation Monitoring System								
✓	1. All components	GE	RW,T,R,RB	-	II	NA		X	
	2. High range area	P	~~RB~~ PC	-	I	I		X	
XL.	Leak Detection System								
	1. Temperature element	GE	PC,RB	-	I	I		X	
	2. Temperature switch	GE	PC,RB	-	I	I		X	
	3. Differential temperature switch	GE	PC,RB	-	I	I		X	
	4. Differential flow switch	GE	PC,RB	-	I	I		X	
	5. Pressure switch	GE	PC,RB	-	I	I		X	
	6. Differential pressure switch	GE	PC,RB	-	I	I		X	
	7. Differential flow summer	GE	PC,RB	-	I	I		X	
✓	8. Reactor building floor drain sumps	P	RB	-	II	NA		X	(21)
✓	9. Reactor building floor drain pumps and piping	P	RB	~~-~~ D	II	See Note (22)		X, B31.1.0	(22)
XLI.	Fire Protection System								
	1. Water spray deluge systems	P	-	-	II	NA		X	
	2. Sprinklers, carbon dioxide systems	P	-	-	II	NA		X	
	3. Portable and wheeled extinguishers	P	-	-	II	NA		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XLII.	Civil Structures								
	1. Reactor building	P	RB	-	I	I			
	2. Office and service building	P	-	-	II	NA			
	3. Screenwell	P	P	-	I	I			
	4. Control building	P	C	-	I	I			
	5. Turbine building	P	T	-	II	NA		ACI-318-71	(23)
	6. Intake Canal	P	-	-	~~NA~~ II	I		ACI-301-66&72	
	7. Discharge tunnel	P	-	-	II	NA		AISC-70	
	8. Discharge pipe and diffuser	P	-	-	II	NA			
	9. Radwaste building	P	RW	-	I	I			
	10. Auxiliary boiler and MG set building	P	-	-	II	NA			
	11. Biological shielding	P	PC,R,RW T,TSC	-	I	I		ACI-318-71 ACI-301-66&72	
	12. Missile barriers	P	RB,R	-	I	I		ACI-318-71 ACI-301-66&72	
	13. Waterproof doors	P	P,R Diesel fuel Pump house	-	I	NA		X	
	14. Site grading	P	O	-	~~I~~ II	NA		~~X~~	
	15. Safety related Masonry walls	P	RB,R	-	I	I		~~NA~~	
XLIII.	Primary Containment Structure								
	1. Reinforced concrete	P	PC	-	I	I		ACI-301	
	2. Liner	P	PC	-	I	I	8/70		(24)
	3. Penetrations	P	PC	-	I	I	8/70	B31.7,1969	
	4. Drywell head and drywell equipment, CRD removal and suppression chamber access hatches	P	PC	-	I	I	8/70	ASME III-B Summer 1969	
	5. Drywell personnel hatch	P	PC	-	I	I	8/70	ASME III-B Winter 1969	
	6. Personnel hatch for drywell equipment hatch (Emergency air lock)	P	PC	-	I	I		ASME III-MC Winter 1972	
	7. Downcomers	P	PC	B	I	I		ASME III-2 Winter 1972	(25)

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XLIV.	Plant Safety Parameter Display System								
	1. SPDS subsystem of ERF	P	TSC bldg.	–	II	I	Δ		
	2. Electrical modules with safety function	P	R	–	I	I	Δ	X	
	3. SPDS display	P	R,	–	II	I	Δ	X	
			EOF, TSC		II	NA	Δ	X	
XLV.	Post Accident Sample System								
	1. Post accident sample system control panel	P	Sample Bldg	–	II	NA	Δ	X	
	2. Post accident sample skid	P	Sample Bldg.	D	II	NA	Δ		
XLVI	Valve Position Indication								
	1. Valve position switch	P,GE	PC,RB	–	I	I			
	2. Data acquisition multiplexer	P	Sample Bldg.	–	II □	~~II~~ I	Δ	X	
	3. SPDS display	P	R	–	II	I	Δ	X	
	4. TSC/EOF display	P	TSC Bldg. EOF Bldg.	–	II	~~II~~ NA	Δ	X	
	5. SPDS subsystem of ERF	P	TSC Bldg.	–	II	I	Δ	X	
	6. TSC/EOF subsystem of ERF	P	TSC Bldg. EOF Bldg.	–	II	~~II~~ NA	Δ	X	
XLVII.	Accident Monitoring Instrumentation (NUREG 0578)								
	1. Containment pressure reactor water level hydrogen concentration high range radiation	GE,P	RB,R,PC		I	I	Δ	X	

Δ Not yet purchased
* Not yet finalized
□ Actually bought to QA Cat. I/Seismic Cat. I standards and specifications, credit is taken for ~~Class~~ QA Category II only.

TABLE 3.2.1-1 (CONT'D)

Notes:

1. A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors; and mechanical modules include turbines, strainers, and orifices.

2. GE = General Electric   P = Stone & Webster

3. PC = within primary containment   R = control building
   RB = within reactor building   T = turbine building
   O = outdoors on site   RW = radwaste building
   P = screenwell house

4. a. Quality group classification per Regulatory Guide 1.26.

   b. I - The equipment meets the quality assurance requirements of 10CFR50, Appendix B.

      II - The equipment meets the quality assurance requirements defined in the purchase specification.

5. I - The equipment is designed in accordance with the seismic requirements for the DBE/OBE.

   NA - The seismic requirements for the DBE/OBE earthquake are not applicable to the equipment.

6. Equipment with purchase order dates listed are constructed in accordance with the codes listed in Table 3.2.1-2. The QA requirements specified at the time of purchase are comparable to the current requirements for the class listed as defined in Regulatory Guide 1.26.

7. In general the principal code represents both the purchase (fabrication) code and the erection code; however, in instances where system upgrading may have occurred after purchase, applicable engineering documents should be consulted for the appropriate erection code.

   "X" = Manufacturer's standard.

8. a. Lines equivalent to a 3/4 in. or smaller liquid line which are part of the reactor coolant pressure boundary are Quality Group B and ASME III, Class 2.

   b. All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate safety systems are Quality Group B and ASME III, Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the instrumentation.

TABLE 3.2.1-1 (CONT'D)

c. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate safety systems are Quality Group D and B31.1.0 from the outer isolation valve or the process shutoff valve (root valve) to the instrumentation.

d. All other instrument lines:

through the root valve are of the same classification as the system to which they are attached

beyond the root valve, if used to actuate a safety system, are of the same classification as the system to which they are attached

beyond the root valve, if not used to actuate a safety system, are Quality Group D and B31.1.0

e. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system are Quality Group D and B31.1.0.

f. All instrumentation lines and associated instrumentation which penetrate the primary containment are designed to Seismic Category I requirements.

9. In the seismic analysis, the dryers are carried in the analysis as lumped mass with the separator assembly.

10. The outermost valve of the three isolation valves in the feedwater lines ~~and the control rod drive system water return line~~ is a positive acting motor operated valve of high leak-tight integrity. The check valve outside the containment is similar to a boiler feed pump check valve and has a spring-loaded piston operator held open by air pressure during normal operation. A fail-open solenoid valve is used to release air pressure and to permit the check valve piston operator to close.

The classification of the feedwater lines from the reactor vessel to and including the third isolation valve is Quality Group A; beyond the third valve is Quality Group D. Feedwater piping within the main steam line tunnel is seismically analyzed for the DBE.

11. The control rod drive insert and withdraw lines from the drive flange up to ~~and including~~ the first valve on the hydraulic control unit are Quality Group B.

12. The hydraulic control unit (HCU) is a General Electric factory-assembly engineered module of valves, tubing, piping and stored water which controls a single control rod drive

TABLE 3.2.1-1 (CONT'D)

by the application of precisely timed sequences of pressures and flows to accomplish slow insertion or withdrawal of the control rods for power control, and rapid insertion for reactor scram.

Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Groups A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components and instruments).

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, (1) all welds are LP inspected, (2) all socket welds are inspected for gap between pipe and socket bottom, (3) all welding is performed by qualified welders, (4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Groups A, B, or C. This was supplemented by the QC techniques described above.

13. The design, construction, materials, inspection, and testing of the accumulator are in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, Division I. An ASME stamp is required. Other codes applied to the accumulator are:

    a. ANSI B16.11 Forged Steel Fittings, Socket Welded and Threaded; and

    b. 10050 Bosses, Standard Dimensions for Gasket Seal Straight Thread.

14. The RCIC and HPCI turbines do not fall within the applicable design codes. To assure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, General Electric has established specific design requirements for this component, which are as follows:

TABLE 3.2.1-1 (CONT'D)

a. All welding is qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.

b. All pressure containing castings and fabrications are hydrotested to 1.5 x design pressure.

c. All high-pressure castings are radiographed according to:

ASTM E-94		
	E-142	20% coverage, minimum
	E-71, 186 or 280	Severity level 3

d. As-cast surfaces are magnetic particle or liquid penetrant tested according to ASME, Section III, Paragraph N-323.3 or N-323.4.

e. Wheel and shaft forgings are ultrasonically tested according to ASTM, A-388.

f. Butt-welds are radiographed according to ASME Section III, N-624, and magnetic particle or liquid penetrant tested according to ASME Section III, N-626 or N-627.

g. Notification is made on major repairs, and records maintained thereof.

h. Record system and traceability are according to ASME Section III, IX-225.

i. Control and identification are according to ASME Section III, IX-226.

j. Procedures conform to ASME Section III, IX-300.

k. Inspection personnel are qualified according to ASME Section III, IX-400.

15. Bellows seal is designed to withstand all normal displacements combined with displacements associated with DBE/OBE.

16. Reactor Water Cleanup

a. Reactor water cleanup, regenerative heat exchanger and primary side of nonregenerative heat exchanger are per ASME B&PV Code Section III Class C; Secondary side of nonregenerative HX is to Section VIII. TEMA class R standards apply to both HX's.

TABLE 3.2.1-1 (CONT'D)

b. Reactor water cleanup pump applies Hydraulic Institute standards for pump design and ASME Section III Class C for sizing of pump thickness.

17. ASME III-3 is applied to cooling coils of unit coolers.

18. ASME III-3 is applied to chilled water system piping, valves, pumps, tanks, and water side of water chiller heat exchangers.

19. These valves are per manufacturer's standards and GE-LSTG QA Requirements as outlined in GEZ-4482A (Quality Assurance Program).

20. The condensate storage tank is designed, fabricated, and tested to meet the intent of API Standard 650. Although the tank is not Seismic Category I, the lower portion of the tank (a volume of 100,000 gal) is encased in a Seismic Category I concrete retention basin.

21. Sumps are encased in Seismic Category I concrete.

22. Reactor building floor drain pumps are designed in accordance with Seismic Category I requirements. Piping, except that encased in Seismic Category I concrete, is analyzed to Seismic Category I requirements.

23. Turbine building structure is analyzed for DBE to ensure integrity of Category I equipment and structure.

24. The reactor containment liner field erection utilized procedures, personnel, and nondestructive examination methods in accordance with the requirements of Regulatory Guide 1.19. The Applicant utilized pertinent sections of ASME III, Class MC, Summer 1972 Addenda in lieu of ASME Section V. The major portion of the shop fabrication of the liner was accomplished prior to the issuance of this guide. Nondestructive examination for shop fabrication was accomplished utilizing as guide ANSI B31.1, ANSI B31.7, and the ASME III, Class B, 1969 Summer Addenda code.

25. Fabricated in accordance with ASME III-2, Winter 1972, except not code stamped.

26. Knife gate valves are per manufacturer's standards for intended service.

27. The inline flow switches (Rotameters) mounted in the discharge lines from each of the reactor recirculation pump motor winding and seal coolers do not fall within the applicable design codes. To assure that these instruments are fabricated to the standards commensurate with their

✓ 28. (see p.6, 7 for text)

TABLE 3.2.1-1 (CONT'D)

safety and performance requirements, General Electric has established specific design requirements for these components, which are as follows:

a. The modules are designed, fabricated, examined, and tested in accordance with the provisions of ANSI B31.1.0 Power Piping.

b. All welding is qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.

c. Notification is made on all weld repairs to butt welds, and records maintained thereof.

d. Acceptance testing and inspection is performed, and results submitted, in accordance with General Electric approved procedures.

TABLE 3.2.1-2

CODE GROUP DESIGNATION

Industry Codes and Standards for Mechanical Components

Quality Group Classification	ASME III Code Classes 1968 Ed.	1971 Ed.	Components Ordered Prior to Jan. 1, 1970	Components Ordered on or After Jan. 1, 1970 to July 1, 1971	Components Ordered on or After July 1, 1971
A	A	1	ASME III, A ANSI B31.1.0 TEMA C Note (a)(g)	ASME III, A ANSI B31.7 I NP and VC I TEMA C Note (a)(g)	ASME III, I NA and NB Subsections TEMA C Note (c)(g)
B	B*,C	2, MC*	ASME III, C,B* ANSI B31.1.0 TEMA C Note (a)(f)	ASME III, C ANSI B31.7, II NP and VC, II TEMA C TANKS (e) Note (a)(f)	ASME, 2 NA and NC Subsections NA and NE Subsections TEMA C TANKS (e) Note (c)(f)
C	-	3	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C Note (b)	ASME VIII, Div. 1 ANSI B31.7, III NP and VC III TEMA C TANKS (e) Note (b)	ASME III, 3 NA and ND Subsections TEMA C TANKS (e) Note (c)
D	-	-	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C, R Note (b)	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C TANKS (e) Note (b)	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C, R TANKS (e) Note (d)

*Metal containment vessel only; see Note (f).

TABLE 3.2.1-2 (CONT'D)

Notes:

(a) Pumps Classified A and B The requirements of ASME Section III, C, Boiler and Pressure Vessel Code, are used as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing cover bolting.

(b) Pumps Classified C or D and Operating Above 150 psig or 212 F The requirements of ASME Section VIII, Div. 1, Boiler and Pressure Vessel Code, are used as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing cover bolting. Pumps classified D and operating below 150 psig and 212 F use manufacturer's standard pump for service intended.

(c) Pumps Classified A, B, and C Use applicable ASME Section III Subsections NB, NC, or ND respectively for vessel design as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing cover bolting. The analytical and/or experimental analysis method that is required for Class I pumps for pressure retaining portions of the pump is furnished by GE-NED to meet the requirements of Subarticles NB-3100 and NB-3200.

(d) Pumps Classified D and Operating Above 150 psig or 212 F The requirements of ASME VIII, Div. 1 are used as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing cover bolting. Pumps operating below 150 psig and 212 F use manufacturer's standard pump for service intended.

(e) Tanks are not fully covered by ASME codes. Groups B and C tanks ordered on or after July 1, 1972, apply Winter 1971 Addenda of ASME Section III, 1971 Edition.

Other tanks are designed, constructed, and tested to meet the intent of API Standards 620/650, AWWA Standard D100 or ANSI B96.1 Standard for Aluminum Tanks, NBS PS-15-69, ASTM Proposed Standard Recommended Practice for Filament Wound Glass Fiber Reinforced Polyester Chemical Resistant Tanks.

(f) Metal containment vessel and penetrations (extensions of containment) are ASME stamped Class B or MC and third party inspected by a qualified inspector.

(g) In-service inspection is required for components including vessels, piping, pumps, and valves that are considered essential portions of the Reactor Coolant Pressure Boundary (RCPB) up to and including the outermost containment isolation valve and these components meet the provisions of In-Service Inspection of Nuclear Reactor Coolant Systems, Section XI of the ASME Boiler and Pressure Vessel Code.

(h) Components required to be stamped to ASME Boiler and Pressure Vessel Code are stamped with the applicable ASME code symbol and third party inspected by a qualified inspector.

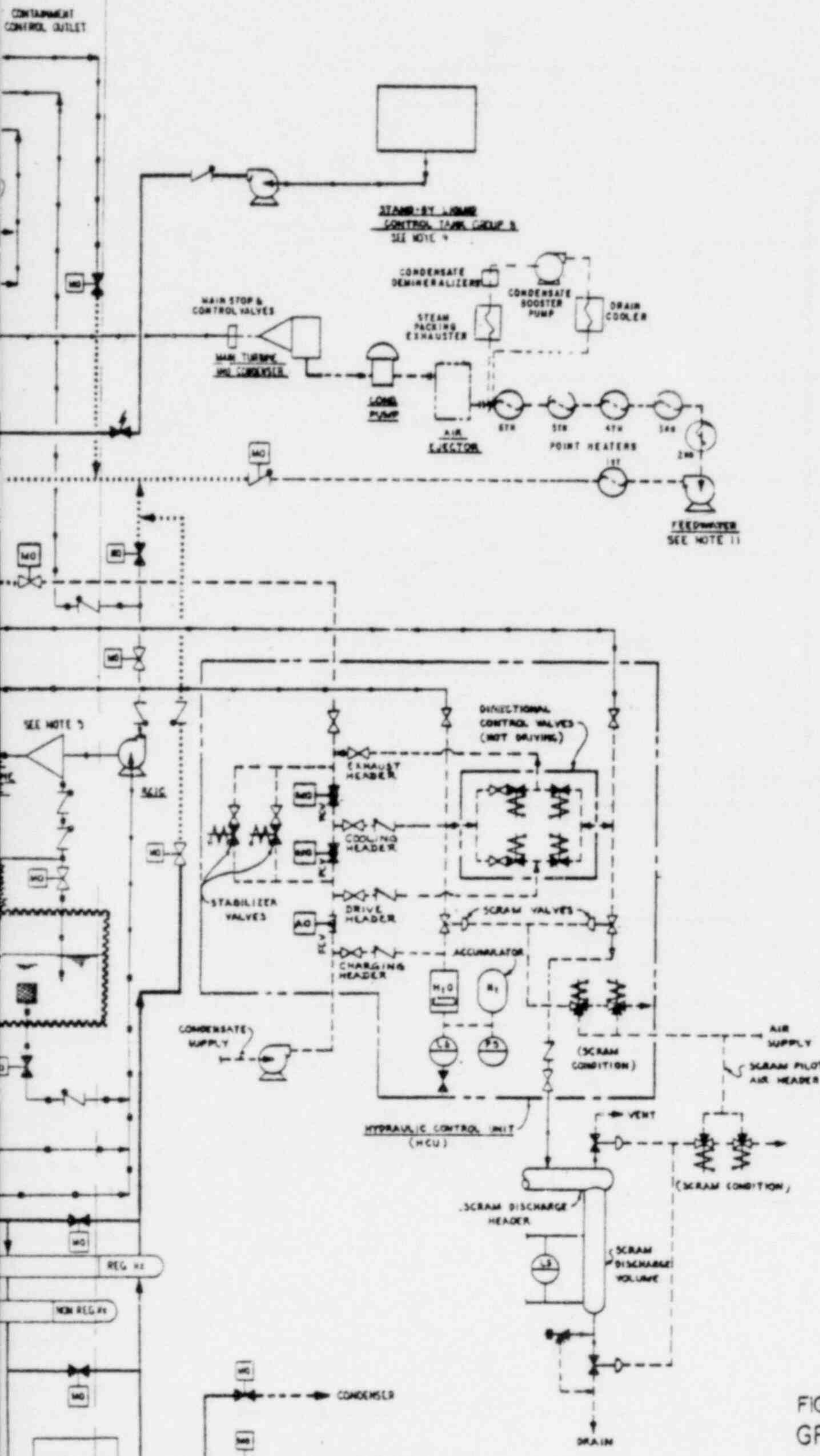
(i) Unless otherwise noted, piping, valves, pumps, tanks, heat exchangers, and miscellaneous equipment for safety related systems are purchased in accordance with the edition and addenda in effect on the date of order. In general, these components in safety related systems are installed to the 1971 Edition of ASME Section III through Winter 1972 Addenda, by virtue of the purchase order date for shop fabricated piping. By Section NA-1140 of ASME Section III, newer edition, addenda, and cases which have not become mandatory on the contract date for a component may be used by mutual consent of the owner or his agent and manufacturer or installer. When using newer requirements, the Applicant (as owner) will ensure that they are acceptable to the enforcement authorities having jurisdiction, in accordance with Section NA-1140.

TABLE 3.2.1-3

CLASSIFICATION & CODE COMPLIANCE REQUIREMENTS
FOR SMALL DIAMETER LINES

Process System to which Instrumentation Connects	Reactor Coolant Pressure Boundary Instrument & Sampling Lines through Root or Isolation Valves		Safety Related Lines beyond Root or Isolation Valve to Sensing Instrumentation	Nonsafety Related Other Instrument Line and Sample Lines beyond Root or Isolation Valve to the Sensing Instrumentation
Quality Group Classification	Over 3/4 in.	3/4 in. or under		
A	A	B	B	D
B	B	B	B	D
C	C	C	C	D
D	D	D	D	D

Note: ANSI B31.1 Code Case 78 may be applied to all lines 3/4 in. and smaller for Classifications A&B or C&D. Code Case 78 has been included in ASME Section III, 1971 Edition.



LEGEND:

GROUP A
GROUP B
GROUP C
GROUP D
UNCLASSIFIED

OTHER SYSTEMS	GROUP
1. OFF GAS	D
2. FUEL POOL (FP) COOLING	C
3. FP FILTER/DEMIN. PRECOAT LOOPS BEYOND ISOLATION VALVE	D
4. HPCI TURBINE VACUUM PUMP & DRAIN POT DISCHARGE SUPPLY	D
5. SERVICE WATER SYSTEM (AUX.)	D
6. SAFETY RELATED PORTIONS OF SERVICE WATER SYSTEM	C
7. REACTOR BLDG. CLOSED COOLING WATER SYS SAFETY RELATED PORTIONS	C
8. FIRE PROTECTION SYS	UNCL
9. HEAT VENT. & AIR COND. SYS	UNCL
10. MAKEUP WATER TREAT SYS	UNCL
11. INSTR. & SERVICE AIR SYS	D (SEE NOTE 8)
12. POTABLE & SANITARY WATER SYS	UNCL
13. STANDBY DIESEL FUEL STORAGE TANK/PIPING	C
14. DRYWELL PENETRATION & EXT. OF CONTAINMENT (SEE NOTE 9) INSTRUMENT LINES (SEE NOTE 8)	B
15. SAFETY RELATED SERVICE/INSTRUMENT/AIR CONDITIONING	(SEE NOTE 7)
16. EQUIPMENT DRAINS DOWNSTREAM OF SHUTOFF VALVES, INCLUDING SAMPLE SYSTEM	D

NOTES.

1. CLASSIFICATION CHANGE AT VALVES. VALVES MUST COMPLY TO HIGHER GROUP.
2. GROUP CLASSIFICATIONS ARE DEFINED IN DESIGN SPEC. "PRESSURE INTEGRITY OF PIPING" SUPP. DOC 1.
3. FOR DETAIL CLASS CHANGE BOUNDARIES AND SYSTEM SHUTOFF VALVING SEE INDIVIDUAL SYSTEMS'S P&ID'S.
4. STANDBY LIQUID CONTROL TEST TANK "GROUP D".
5. THE HPCI & RCIC TURBINE ARE CATAGORIZED AS MACHINERY-UNCLASSIFIED. DESIGN & QUALITY CONTROL REQUIREMENTS SPECIFIED ARE COMMENSURATE WITH THE APPLICATION.
6. LINES LEAVING THE ECCS WATER SOURCE SHALL BE CAPABLE OF BEING ISOLATED BY A REMOTE MANUALLY ACTUATED SWITCH (RMS) SHUTOFF VALVE OR LOCKED CLOSED VALVE.
7. AIR SUPPLY LINES TO MSIV/ADS VALVES ARE SAFETY RELATED, GROUP B CLASSIFICATION.
8. A 3000# RESTRICTING ORIFICE COMPLYING WITH A BORE DIAMETER OF 1/4" MAY BE SOCKET-WELDED ONTO THE OUTER SIDE OF THE LINE FOR ISOLATION PURPOSE.
9. COMPONENTS WHICH FORM AN EXTENSION OF CONTAINMENT ARE DEFINED AS PRESSURE RETAINING COMPONENTS, PARTS, MATERIALS, OR APPURTENANCES IN THE POTION OF THE SYSTEM WHICH EXTENDS FROM THE CONTAINMENT VESSEL TO AND INCLUDING THE OUTERMOST ISOLATION VALVE LOCATED OUTSIDE OF THE CONTAINMENT VESSEL AND WHICH IS CAPABLE OF EXTERNAL ACTUATION. SEE P&ID'S FOR EXTENSION OF CONTAINMENT.
10. CLASSIFICATION OF SMALL DIAMETER LINES. SEE SUPP DOC 1 TABLE C.
11. BUTT WELDS OF ALL MAIN FEEDWATER PIPING INCLUDING BRANCH LINES LOCATED INSIDE THE TUNNEL UP TO THE THIRD ISOLATION VALVE ARE TO BE 100% RT. FEEDWATER PIPE IN TUNNEL IS SEISMICALLY ANALYZED FOR THE DESIGN BASIS EARTHQUAKE.
12. REACTOR VENT & DRAIN LINES SHALL COMPLY WITH DOUBLE VALVING REQUIREMENTS OF SUPP DOC 2.

FIG. 3.2.2-1
GROUP CLASSIFICATION & ISOLATION
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

EXHIBIT 3

Letter, dated April 8, 1982, from T.S. Ellis (Hunton & Williams)
to Larry Lanpher.

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707 EAST MAIN STREET P. O. BOX 1535
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TELEPHONE 804-788-8200

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919-828-9371

FIRST VIRGINIA BANK TOWER
P. O. BOX 3889
NORFOLK, VIRGINIA 23514
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1919 PENNSYLVANIA AVENUE, N. W.
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WASHINGTON, D. C. 20036
202-223-8650

FILE NO. 24566.3

DIRECT DIAL NO. 804 788- 8453

April 8, 1982

VIA TELECOPIER

Larry Lanpher, Esq.
Kirkpatrick, Lockhart, Hill,
Christopher & Phillips
1900 M Street, N.W.
Washington, D.C. 20036

Dear Larry:

Pursuant to our telephone conversation of today, I am writing to record the changes or clarifications to the equipment classification table which we discussed today. Our present intention is to reflect these changes or clarifications in an amended table to the FSAR.

Page 1 -- On page 1, for item 7(a) of the Reactor System, change the LILCO QA category from "I" to "II" and the seismic category to "NA." Also on page 1, for dryers, item 7(b), change the seismic category to "NA."

Page 2 -- On page 2 of 24, for item 5 of the Nuclear Boiler System, change the quality group classification from "A" to a dash.

Page 3 -- On page 3 of 24, delete items 1 and 5 under System IV. Also on page 3 of 24, for item 6, pump motor under System III, change the LILCO quality assurance category from "II" to "I."

Page 4 -- On page 4, for items 12 and 13 under System V, change the LILCO QA category from "NA" to "II."

Page 5 -- On page 5, change the location code for item 3 of the RHR System to "PC, RB." Also change the location code for item 4, System IX, to "RB, PC." The system component description for item 4 of the RHR System

should be changed from "Piping, beyond outermost isolation valves" to "Piping, other." Also on page 5, the location code for item 1 in System X should be changed from "PC" to "PC, RB." The description of that item should be changed from "Piping, within outermost isolation valves" to "Piping, connected to RCPB within outermost isolation valves."

Page 6 -- On page 6, item 2 for System X should be changed from "Piping, beyond outermost isolation valves" to "Piping, other." The location code for this same item should be changed from "RB" to "RB, PC." Also on page 6, with respect to item 5 of System X, the description "Valves, isolation and within" should have added to it the phrase "connected to RCPB." The next item, item number 6, should be changed from "Valves, beyond outermost isolation valves" to "Valves, other." Also on page 6, for System XI, the location code, quality group classification and principal code for item number 1, "Piping, within outermost isolation valves" should be changed from "PC," "A," "ASME III-1" to "RB, PC," "A/B," "ASME III-1, ASME III-2." Also, a new note 28 should be appended to this item which reads as follows: "Component is quality group A when connected to RCPB." Similarly, for item 6 of System XI, the quality group classification should be changed to "A/B" and the principal code should be changed to "ASME III-1, ASME III-2" and new note 28 is also applicable here.

Page 7 -- For item 1 of System XII, the location code should be changed from "PC" to "PC, RB" and the quality group classification and principal code should be changed to "A/B" and "ASME III-1, ASME III-2." New note 28 is also applicable to this item. Similarly, the quality group classification principal code and comment section for item 6 of System XII should be changed to "A/B" and "ASME III-1, ASME III-2" and new note 28, respectively.

Page 8 -- On page 8, the location code for item 3 of System XVIII should be changed from "PC" to "PC, RB." For item 5 in System XVIII, the typographical error "PB" in the location code should be changed to "RB." The remainder of the location code remains the same.

Page 10 -- On page 10, the location code for item 2 of System XXV should be changed from "--" to "P,O,R,T." Also on page 10, "RB" should be added to the location code

for item 5 of System XXV and "R" should be added to the location code for item 6 of the same numbered system. Also with respect to System XXV, the descriptions of items 3, 4 and 5 should be amended to read "Pumps, safety related," "Pump motors, safety related," and "Valves, safety related," respectively. Two items should also be added to System XXV for clarification. First, add an item entitled "Pumps, other" with scope of supply "P," location "P," quality group classification "D," quality assurance category "II," and seismic category "NA." The second item to be added is "Motors, other" with scope of supply "P," location "P," quality group classification "--," quality assurance category "II," and seismic category "NA."

<u>Page 11</u> -- On page 11, the location code for item a.9 on System XXVII should be changed from "--" to "O." The description of item b. under System XXVII should read "A/C Power Systems, Safety Related."

<u>Page 12</u> -- On page 12, "RW" in the location code for item 5 of System XXVII should be stricken. For item d. in System XXVII, the seismic category should be "I." For item e.4 of the same system, the purchase order date should be "4/75" rather than "11/75." In System XXVIII, the location code for items 1 and 2 should have "PC" added to both.

<u>Page 13</u> -- On page 13, the quality group classification for item 1 of System XXX should be a dash.

<u>Page 14</u> -- On page 14, item 4 of System XXXIII should be deleted in its entirety as it is repeated elsewhere. In System XXXIV, the quality assurance category should be "I" for all 6 listed items. Also, for items 3, 4, 5 and 6 of System XXXIV, the quality group classification should be a dash rather than "NA." The seismic category "I" for item 5 should be changed to "NA" because it relates only to cables.

<u>Page 15</u> -- On page 15, the location code for item 2 of System XXXVI should be "PC, RB." The location code for item 4 of System XXXVI should read "R, RB." For item 7 of System XXXVI, the quality assurance category should be "II" and the seismic category should read "NA." For

System XXXVII on page 15, the location codes for items 3 and 4 should read "PC, RB." The description of item 1 for System XXXVIII should have the phrase "safety related" added to it.

Page 16 -- On page 16, the location code for item 2 of System XXXIX should be "PC." For item 9 of System XL, the quality group classification should read "D" and the principal code should read "x, B 31.1.0."

Page 17 -- On page 17, the quality assurance category for item 6 of System XLII should read "II." The quality assurance category for item 14 of System XLII should read "II" and the "X" in the principal code column for that item should be deleted. Similarly, the "NA" should be deleted from the principal code column for item 15 of System XLII. Finally, the dash in the principal code column for item 2 in System XLIII should be deleted.

Page 18 -- On page 18, the location code for item 3 of System XLIV should have "EOF, TSC" added to it. Note, however, that this addition should be written underneath the "R" that already exists as the quality group classification, quality assurance category, et al. for "EOF, TSC" is different from that pertaining to location "R." Those columns for the "EOF, TSC" location code are "--," "II," "NA," "Δ," and "x," respectively. Also on page 18, the seismic category for item 2, System XLVI, should be changed to "I." Similarly, the seismic category for items 4 and 6 of System XLVI should be changed to "NA." As I indicated to you, I believe the "NA" is meant to apply to both buildings. Finally, also on page 18, the term "QA category" should be substituted for the word "Class" in the note pertaining to □ at the bottom of the page.

Page 20 -- For note 10 on page 20, the language "and the control rod drive system water return line" should be deleted. For note 11 on the same page, the words "and including" in the second line should also be deleted.

Best wishes.

Sincerely,

T. S. Ellis, III

75/403

EXHIBIT 4

Table of Equipment Relied upon by LILCO for DBA Mitigation, Per FSAR Chapter 15 and Shoreham Emergency Operating Procedures.

TABLE OF EQUIPMENT RELIED UPON BY LILCO
FOR DBA MITIGATION, PER FSAR CHAPTER 15
AND SHOREHAM EMERGENCY OPERATING PROCEDURES

SYSTEMS/COMPONENTS	"Safety-Related"*	Feedwater Controller Failure -Max. Demand Ch. 15	Loss of Feedwater Ch. 15	Feedwater/Level Contr. System Failure EOP (1)	Loss of AC Power Ch. 15	Loss of Off-Site Power EOP (2)	Loss of Condenser Vacuum Ch. 15	Loss of Condenser Vacuum EOP (1)	ATWS Ch. 15	ATWS EOP (3)	LOCA Ch. 15	LOCA EOP (4)
Reactor Protection Sys and related trips												
--rods scram	*	x	x	x	x	x	x	x			x	x
--recirculation pump trip	*	x	x	x	x	x	x	x	x	x		x
--main turbine trip	*	x		x	x	x	x	x				x
--feedwater pump trip	*	x		x	x	x	x	x				x
--NSSS Isolations	*		x	x		x						
Manual Scram/Standby Liquid Control	(a)								x	x		x
Reactor Manual Control/Rod Sequence Control	(f)									x		
Main Steam Isolation Valves /System	*		x	x	x	x	x	x	x		x	x
--MSIV leakage collection	*											x
Safety/Relief Valves	*	x	x	x	x	x	x			x		
Scram Discharge Volume vent & drain valves	*			x			x			x		x
SRM's/IRM's	*			x			x	x				x
Diesel Generators	*				x	x					x	x
Emergency Buses	*					x						x
Control Rod Drive Hydraulic	(e)					x			x	x		
--Local Hydraulic Control Units	(b)								x	x		
Post Accident Monitoring	*			x								x

SYSTEMS/COMPONENTS	"Safety-Related"▲	Feedwater Controller Failure -Max.Demand Ch. 15	Loss of Feedwater Ch. 15	Feedwater Level Cont System Failure EOP (1)	Loss of AC Power Ch. 15	Loss of Off-Site Power EOP (2)	Loss of Condenser Vacuum Ch. 15	Loss of Condenser Vacuum EOP (1)	ATWS Ch. 15	ATWS EOP (3)	LOCA Ch. 15	LOCA EOP (4)
High Pressure Coolant Injection	▲	x	x	x	x	x	x	x		x	x	x
Reactor Core Isolation Cooling	(c)	x	x	x	x	x	x	x		x		x
Residual Heat Removal												
--suppression pool cooling	▲			x	x	x		x		x	x	x
--steam condensing	▲		x		x					x		
--containment spray	▲										x	x
--LPCI	▲										x	x
Core Spray	▲											x
RBCLCWS	▲					x						x
TBCLCWS	—					x						
Service Water	▲					x						x
Motor Generators	(f)					x						
--lube oil pump	(f)					x						
--emergency lube oil pump	(f)					x						
RBSVS	▲			x		x					x	x
Control Room Air Cond.	▲					x					x	x
Reactor Water Cleanup	(d)									x		

1

SYSTEMS/COMPONENTS	"Safety-Related"*	Feedwater Controller Failure -Max.Demand Ch. 15	Loss of Feedwater Ch. 15	Feedwater/ Level Contr System Failure EOP (1)	Loss of AC Power Ch. 15	Loss of Off-Site Power EOP (2)	Loss of Condenser Vacuum Ch. 15	Loss of Condenser Vacuum EOP (1)	ATWS Ch. 15	ATWS EOP (3)	LOCA Ch. 15	LOCA EOP (4)
Feedwater												
--FW flow (pumps)	--									x		x
--FW controller	--	x		x								
--Reactor feed pump speed controller	--			x								
--Low flow control valve	--			x								
Main Turbine												
--stop valve	--	x		x		x	x	x				x
--stop valve position switch	--	x		x		x		x				x
--control valve	--			x	x	x		x				x
--bypass valve	--	x		x	x	x	x	x		x		x
--intermediate stop valve	--			x		x		x				x
--extraction line isolation valve	--			x		x		x				x
--motor suction pump	--			x		x		x				x
--bearing lift pump	--					x						
--DC oil pump	--				x	x						
--turning gear oil pump	--					x						
--emergency turning gear oil pump	--					x						
--moisture separator trip signal	--	x										
Automatic Depressurization	--											x
--timer	--											x

SYSTEMS/COMPONENTS	"Safety-Related"*	Feedwater Controller Failure -Max. Demand Ch. 15	Loss of Feedwater Ch. 15	Feedwater/ Level Contr. System Failure EOP (1)	Loss of AC Power Ch. 15	Loss of Off-Site Power EOP (2)	Loss of Condenser Vacuum Ch. 15	Loss of Condenser Vacuum EOP (1)	ATWS Ch. 15	ATWS EOP (3)	LOCA Ch. 15	LOCA EOP (4)
Condenser												
--condensate pump	--			x				x				x
--condenser vacuum	--			x	x	x		x				x
--cond. booster pump	--			x			x	x				x
--cond. circ. water & booter pump trip	--				x	x						
H_2 Recombiner	*											x
Post-LOCA H_2 Analyzer	*											x

NOTES:

* Safety-related, per Exhibits 2 and 3 of this testimony.
-- Non-safety related, per Exhibits 2 and 3 of this testimony.

(1) Review of this EOP (SP. 29.006.01) incorporates SP. 29.010.01, "Emergency Shutdown Emergency Procedure", as referenced.

(2) Review of this EOP (29.015.01) incorporates SP. 29.010.01, "Emergency Shutdown Emergency Procedure" and SP. 29.023.01, "Level Control", as referenced. SP. 29.015.02, "Station Blackout Emergency Procedure" was referenced in but not reviewed for this EOP scenario as this Station Blackout procedure was not available to Suffolk County.

(3) Review of this scenario incorporates SP. 29.024.01, "Anticipated Transients Without Scram", and SP. 29.004.01, "Emergency Use of S.L.C. Emergency Procedure".

(4) Review of this scenario incorporates SP. 29.014.01, "Loss of Coolant (Intermediate Break) Emergency Procedure", and SP.014.02, "Loss of Coolant (Large Break) Emergency Procedure". SP. 29.010.01, "Emergency Shutdown Emergency Procedure" was also incorporated into this review, as referenced in the two LOCA procedures.

(a) See Exhibit 2, Section V., Items 12 and 13.
(b) See Exhibit 2, Note (12).
(c) See Deposition, pg. 40, (Robare).
(d) See Exhibit 2, Section XIX., Items 1, 5 and 7.
(e) See Exhibit 2, Section IV., Items 6, 8 and 9.
(f) Equipment not readily identifiable in Exhibit 2.

EXHIBIT 5

Board Notification 82-08, "Errors in BWR Vessel Water Level Indication," dated February 9, 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos. 50-322
50-341
50-358
50-387/388
50-466
50-556/557

FEB 9 1982

MEMORANDUM FOR: The Atomic Safety & Licensing Boards for:

Shoreham Nuclear Power Station, Unit 1
Enrico Fermi Atomic Power Plant, Unit 2
William H. Zimmer Nuclear Power Station, Unit 1
Susquehanna Steam Electric Station, Units 1 and 2
Allens Creek Nuclear Generating Station, Unit 1
Black Fox Station, Units 1 and 2

FROM: Robert L. Tedesco
Assistant Director for Licensing
Division of Licensing

SUBJECT: BOARD NOTIFICATION - ERRORS IN BWR VESSEL WATER LEVEL INDICATION (Board Notification 82-08)

In accordance with present NRC procedures regarding Board notifications, the enclosed information is being provided for your information as constituting new information relevant and material to safety issues. This information is generic and has applicability to all dockets with boiling water reactors.

Robert L. Tedesco
Assistant Director for Licensing
Division of Licensing

Attachment:
DSI/NRR memo dated 1/15/82

cc: See next page

FEB 9 1982

DISTRIBUTION OF BOARD NOTIFICATION

Allens Creek, Docket No. 50-466

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Shoreham, Docket No. 50-322

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Enrico Fermi, Docket No. 50-341

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Atomic Safety and Licensing
Board Panel
Atomic Safety and Licensing
Appeal Board Panel
Docketing and Service Section

Black Fox, Docket Nos. 50-556, 50-557

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JAN 15 1982

MEMORANDUM FOR: Roger J. Mattson, Director
Division of Systems Integration

FROM: Themis P. Speis, Assistant Director for Reactor Safety
Division of Systems Integration

SUBJECT: ERRORS IN BWR VESSEL WATER LEVEL INDICATION

Attachment A provides a summary of the results of work done to date in the RSB and ICSB under Task Interface Agreement 81-21 "Pilgrim 1, Water Level Instrumentation Oscillation." It is emphasized that review of this issue is not complete, even though we have proposed some short and long-term recommendations. By copy of this memo, I am requesting that comments or other relevant feedback on the contents of this memo, and especially the proposed recommendations, be provided to C. Graves by 1/27/82.

Themis P. Speis, Assistant Director
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ATTACHMENT A

BWR WATER LEVEL INDICATION ERRORS

I. INTRODUCTION

On September 26, 1981, during a routine reactor shutdown and cooling operation at Pilgrim 1, there were several large oscillations of Yarway level detection indication (reference 1). The first oscillation caused high level isolation followed by low level scram. The oscillations were attributed to high containment temperatures, which caused flashing in the heated reference legs of the Yarway instruments. At the time, the reactor coolant temperature was about 220°F while the temperature in the upper part of the drywell was 240°F.

In a Task Interface Agreement of October 1981 (reference 2), NRR was assigned the following action plan items:

1. Review event to establish the generic licensing implications; (DSI/RSB & ICSB)
2. Review adequacy of Pilgrim Tech Spec on high containment temperature; (DSI/RSB)
3. Determine acceptability of oscillations in safety related instruments; (DSI/RSB & ICSB)

This memorandum summarizes the results of work in RSB and ICSB to date, provides preliminary responses to the Task Interface Agreement action items and lists some possible short and long-term solutions. It is emphasized that the information in this memorandum is preliminary since the review is not complete. A report dealing with the problem which was prepared for the BWR Owners was obtained from General Electric on 12/31/81 and has been given only a cursory review thus far. Detailed discussions with General Electric personnel will be held after staff review of the GE report.

II. BACKGROUND

As the result of the TMI-2 accident in March 1979, both the staff and industry have reviewed the adequacy of level detection instrumentation under accident conditions. In April, 1979, IE Bulletin 79-08 (reference 3) requested information from each licensee on vessel level indication. IE Bulletin 79-21, "Temperature Effects on Level Measurement" (reference 4) was issued in August, 1979. This bulletin addressed errors in steam generator water level resulting from high energy line breaks, including LOCA, inside containment and consequential high containment temperature which caused temperature increases and possible flashing of water in the reference leg of the level indicator. The problem was identified in a Westinghouse letter of June 1979. Although the bulletin required actions from PWR operators, it was also sent as information to all BWR operators. A staff letter (reference 5) addressing this problem was sent to all BWR licensees in July 1979. In July, 1979, General Electric notified its customers of false level indication caused by high temperatures and possible flashing of water in the reference legs of Yarway level instruments under post-LOCA conditions (reference 6). In September, 1980, General Electric again notified its customers of the importance of compensating for these false level indications in Yarway instruments and described false level indications in cold reference leg instruments caused by flashing in the sensing lines (reference 6). A staff review and evaluation of level instrumentation errors for BWRs, based on a review of GE information provided in August 1979 in NEDO-24708 (reference 7) is presented in NUREG-0626 (reference 8).

Additional information on the safety significance of errors in or total loss of level indication was provided during 1980 in NEDO-24708A (reference 9) and NEDO-25224 (reference 10). Some current information is available in the proposed emergency procedure guidelines for BWRs which are presently under staff review (see reference 11 and recent revisions) and in the Shoreham docket (reference 12).

III. WATER LEVEL INSTRUMENTATION

All level measurement systems in BWRs employ differential pressure transmitters, a reference leg connected to a condensing pot and in turn to the reactor vessel steam space, and a variable leg connected to the vessel at a lower elevation. Several differential pressure cells share common impulse legs. Temperature compensated and uncompensated reference legs are employed. Those level measurement systems which use a temperature compensated reference leg are called Yarways. Those level measurement systems which use an uncompensated reference leg are called cold leg instruments or, often, GEMAC.

BWR 1, 2, 3 and some 4's use two redundant Yarways to generate engineered safety feature actuation signals and cold reference leg instruments for indication and control. The remaining BWR 4's and all 5 and 6's use redundant cold reference leg systems exclusively.

A. Yarway (Heated Reference Leg) Instrument

A schematic of a Yarway level detector is presented in Figure 1. Steam condensed in the condensing chamber maintains the reference leg water level by overflow to the variable leg. The condensate heats the variable leg which, in turn, heats the reference leg. A thermal shield is provided to reduce heat loss to containment and to maintain relatively high reference leg temperatures. For short column Yarways metal clamps have also been used to improve heat transfer between the legs. Information in reference 9 indicates that the reference leg temperature is roughly equal to local containment temperature plus 40 percent of the difference between reactor steam temperature and local containment temperature. For example, a local containment temperature of 135°F and steam temperature of 546° (Tsat at 1000 psia) would result in a reference leg temperature of 300°F.

The sensing lines leading from the Yarway to the differential pressure cell outside of the drywell are 1" schedule 80 stainless steel piping. Flow in these lines is blocked by the differential pressure cell. During normal operation, the stagnant water in these lines should be approximately at local containment temperature. If the lines are installed close to each other in containment, they should have about the same elevation change and local temperature, Hence, the effects of water density variations along the lines should be cancelled and have a minor effect on level measurement.

The Yarway level detector, which measures the collapsed water level in the outer annulus region of the reactor vessel, is subject to a number of uncertainites. Those resulting from differences between actual and assumed values of average coolant density in the annulus (affected by system pressure, subcooling and carryunder) were shown to be small in reference 9. However, in 1979 the General Electric Company identified rather large uncertainties associated with high reference leg temperatures that could occur under some accident conditions (steam line breaks) for which local containment temperatures up to 340°F are predicted.

The high reference leg temperatures would result in false high water level signals. In addition, a constant indicated lower water level could be reached even though the actual water level has dropped well below the low level tap at the reactor vessel. Hence, GE recommended that its customers review calibration of the Yarway instruments, increase certain trip points and take other corrective actions to compensate for this effect.

High containment temperature combined with reactor depressurization can also lead to false water level readings because of flashing or boiling in the reference leg or the sensing lines within containment leading to the differential pressure sensor. Flashing in the lines might occur during depressurization if the local containment temperature exceeds the saturation temperature corresponding to vessel pressure. Flashing in the reference leg might be expected earlier in the transient because of th higher initial temperatures in the reference leg. The GE communication of 1979 was concerned only with the effects of flashing in the reference leg of Yarway instruments Apparently, flashing in cold reference leg instruments was considered to be of minor importance at the time. In a later communication (September 1980), flashing in the sensing lines of cold reference leg instruments was also considered.

Flashing in the reference leg or lines could occur during normal system depressurization in preparing for initiation of RHR cooling or under accident conditions. During the cooldown event at Pilgrim on 9/26/81 (see reference 1), flashing of the reference legs in the Yarway instruments was indicated by several oscillations in the level readings. At the time, the reactor coolant temperature was 220°F and peak local containment temperatures were about 240°F. Under accident conditions such as a steamline break, local containment temperatures can reach 340°F. Hence, when vessel pressure drops below about 112 psig (p_{sat} at 340°F) flashing could occur in the lines. If it is assumed that the reference leg temperature rapidly increases to the steady state value for a containment temperature of 340°F and RCS temperature of 546°F, flashing in the reference leg might occur when vessel pressure drops below about 300 psig (P_{sat} at 422°F).

Another scenario involving flashing in the reference leg could occur for larger breaks and times such that the vessel pressure is about equal to containment pressure. In this case, as discussed in reference 13, the rapid reduction in containment pressure following initiation of the containment spray, combined with the delay in reduction of metal temperatures, could cause flashing in the reference leg. Tests were conducted to confirm that large errors in level indication could occur. The solution to this flashing problem involved installation of a cooling jacket around the reference leg which was supplied with water from the containment spray line.

Even without a break, loss of the non-safety grade containment coolers would cause the containment to heat up and could cause flashing upon depressurization.

With respect to the flashing problem it should be noted that there would be a time delay involved in the heating of the reference leg and lines under accident conditions. A delay in heat transfer would be expected because of the relatively large amount of metal in the walls of the reference leg and lines and the relatively low heat transfer coefficients expected for surfaces in contact with the containment atmosphere. In reference 9, the thermal time constant for the Yarway detector was estimated to be about 20 minutes. This value may have been calculated assuming only high temperature air. For steam-air mixtures, the condensation on cold surfaces results in appreciably larger heat transfer coefficients than those for air at the same temperature. It should also be noted that water expelled by flashing in the heated reference leg and corresponding line to the differential pressure sensor may not be replaced quickly. At the high containment temperatures and lower vessel pressure expected under accident conditions, the condensing chamber could cease to function. Hence, refill would be delayed until sometime after the vessel water level increases to a point above the tap leading to the condensing chamber. Even under these circumstances, boiling could occur for a while in the reference leg and lines as the result of continued high local containment temperatures. In the case of degraded core

cooling when water level remains well below the tap to the condensing chamber and noncondensible gases and superheated steam could be present, there could be extended time periods with large false indications of vessel water level. In fact, purging of the lines could be required to remove non-condensibles.

B. Cold Reference Leg Instruments

A schematic of a cold reference leg instrument is presented in Figure 2. In this case, the reference leg upper level is maintained by overflow of condensate in the condensing chamber back through the tap to the vessel. Water density effects and flashing in the lines within containment which lead to the differential pressure sensor could be of concern. Changes of elevation in the lines inside of containment range from 1 to 40 feet in operating plants. Hence, flashing in the lines under accident conditions could cause false water level indications and delay in refill problems such as those discussed in Section A. Flashing in cold reference leg level instrument lines was recognized in the guidelines developed by GE (reference 11). This situation (loss of reliable level indication for both heated and cold reference leg detectors) was treated by operator instructions to initiate ADS and ECCS actuation to fill the vessel and overflow to the suppression pool via the S/R valves.

IV. RESPONSE TO SPECIFIC ACTION ITEMS:

1. Review event to establish the generic licensing implications.

   All BWR vessel level instrumentation, to some degree, is susceptible to reference leg flashing and consequential loss of level indication following rapid vessel depressurization such as observed at Pilgrim. The generic BWR emergency procedure guidelines* include caution and action statements related to loss of level indication. The suscepta-bility of the level indication system to substantive non-conservative errors during event sequences which include depressurization, and the adequacy of emergency procedures is discussed below.

2. Review adequacy of Pilgrim Technical Specification on high containment temperature.

   The Pilgrim Technical Specifications do not include drywell temperature as a limiting condition for operation. We believe such a specification would be prudent to prevent undue equipment aging. However, a LCO on the pre-accident drywell temperature will not preclude post accident loss of vessel level indication.

3. Determine acceptability of oscillation in safety related instruments.

   Engineered safety feature actuation signals are generated using the following process variables:

   High pressure core spray (HPCS) - vessel level or drywell pressure

   Low pressure core spray (LPCS) - vessel level or drywell pressure

*These guidelines are presently under review by the staff and are not, to date, employed at Operating Reactors.

Low pressure coolant injection (LPCI) - vessel level or drywell pressure

Automatic depressurization system (ADS) - vessel level and drywell pressure

Containment Spray (CS) - vessel level and drywell pressure

Reactor Core Isolation Cooling (RCIC) - vessel level only.

Delays in initiation of engineered safety features due to reference leg heatup and boiloff have been considered in response to IE Bulletins 79-08 and 79-21. The staff concluded in NUREG-0626 that for all break sizes, the reactor either depressurizes fast enough to allow timely initiation of the low pressure system on high drywell pressure, or the breaks are small enough that (at worst) ECC functions occurred before the potential boiling of the reference leg fluid.

Furthermore, ESFAS systems employ latching circuitry except on the ADS level permissive to ensure that safety actions, once initiated, go to completion (IEEE 279).

Hence, concerns related to initiation accuracy for automatic safety systems due to reference leg heatup and/or flashing and concerns related to potential reference leg fluid oscillation have been previously and adequately addressed for design basis events; however, there are event sequences involving multiple equipment failure which will require manual initiation of engineered safety features.

For some accident scenarios involving a break inside containment, adequate indication of actual vessel water level could be lost for all pertinent level instruments as the result of flashing and boiling in the reference legs. The emergency guidelines (reference 11 and revisions) consider the case

where the operator has recognized that vessel level cannot be determined. For this case, the guidelines involve actions to depressurize the reactor and to refill the system until it overflows to the suppression pool via the S/R valves. However, if the operator fails to recognize that he has lost level indication and has a <u>false</u> high reading of water level, he might take action to throttle or stop ECCS systems in order to avoid filling steam lines or to reduce load on emergency power systems. In this case, the flashing or boiling in the reference legs could lead to operator actions prejudicial to plant safety.

V. RECOMMENDATIONS

These are preliminary. Once we have received feedback from people on the distribution list and met with the BWR Owners Group, they will be finalized.

A. Short-Term Recommendations

(1) Operators should be warned that all level indication is susceptible to large inaccuracies. We are concerned that operators may have been trained to unduly depend upon cold leg instrumentation should they recognize errors in Yarway reference leg instrumentation.

A cursory examination of plant procedures at Pilgrim 1 and Browns Ferry show that concerns related to cold leg instrumentation inaccuracies have not been incorporated in their procedures. The operators may have been warned of these concerns by other mechanisms such as training sessions. We believe that utilities are aware of potential water level inaccuracies in Yarway and cold leg instrumentation based on staff review of GE documents prepared for the staff and documents prepared for GE owners. Early documents recommended reliance on cold leg instrumentation. Later documents warned that these instruments, depending on the plant specific installation, might also exhibit substantive indicated level errors. We do not know whether or not these concerns and corresponding warnings and actions have been communicated to the control room operators.

(2) Plant specific emergency procedures should be confirmed and/or modified to:

(a) Clearly identify which level indicators in the control room employ Yarway reference legs and which employ cold reference legs, and direct the operator to the appropriate indicators.

(b) Include procedures to help the operator decide when level instrumentation is to be mistrusted. Relate specific drywell temperature indication, readily and reliably available to the operator in the control room, to reference leg temperature.

(c) Include procedures to help the operator recognize those plant conditions and observed instrument responses which indicate successful refilling of reference legs following flashing.

(3) Operability limits of the temperature sensors used in (2)(b) above should be included in the plant Technical Specifications.

B. Long-Term Recommendations

We believe that it is prudent to provide the operator with continuous reliable level information. Event sequences have been identified during which reliable indication will be temporarily lost. This potential is addressed in the emergency procedure guidelines now under review by the staff. Hardware modifications should be sought to address this problem.

We believe that operator recognition of loss of accurate level information as addressed in the emergency procedure guidelines is cumbersome at best. The operator is to relate indicated water level and drywell temperature using a table contained in a caution statement of the emergency procedures. Indicated water level values beyond the ranges shown in the table are to be mistrusted. Automation of these actions and decisions seems in order.

Should the operator decide that the water level indicators are to be mistrusted, the operator is to fill the vessel. Supposedly reference legs would ultimately refill. At some point in the event sequence, the operator should be provided with positive means to confirm that reliable water level indication has been restored. This problem may not be adequately addressed in the emergency guideline procedures which are presently under staff review.

Several potential plant modifications are being considered by the staff. It is not our intent to dictate hardware fixes. Rather, we give the below recommendations as illustrations that reference leg flashing is a tractable problem.

(1) Perform plant specific analysis of susceptibility of cold leg level instrumentation to reference leg flashing and/or local heatup and corresponding water expulsion. Those plants which are designed with small vertical drops of reference legs inside the drywell should be satisfactory as designed.

(2) Consider rerouting of reference legs to meet condition (1) above.

(3) Install temperature measurement of the reference leg. Such measurements could be used to confirm operability following a drywell temperature excursion and subsequent cooldown. The measurement would be of little use should high drywell temperatures be sustained.

(4) Develop means to cool the reference leg by establishing flow within the leg. Two techniques have been suggested: (1) the temporary opening of equalization valves and/or drain valves, and (2) pumping water with a positive displacement pump from outside the drywell, up reference lines and into the vessel. Equalization and drain valves are local manual valves They are hypothetically accessible following an accident. The drain lines are routed to the waste treatment system. Following vessel depressurization, reference leg flashing and subsequent vessel filling in accordance with emergency procedures, temporary opening of the valves could be used to ensure reference leg filling. No hardware modifications would be required. Should a sufficiently large LOCA occur, or should an event sequence involving multiple equipment failure occur, such that the

vessel cannot be filled above the reference leg taps, this technique would be of little use. Pumping water up reference legs would obviously require hardware modifications. The flowrate need only be high enough to overcome the heat load on the reference legs inside the drywell under accident conditions. This technique would permit reference leg filling even if high drywell temperatures existed and the vessel could not be filled to the reference leg tap.

(5) Develop means to cool the reference leg by using a coolant jacket and diverted ESF flow.

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4. IE Bulletin 79-21, "Temperature Effects on Level Measurements", August 9, 1979.

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12. Attachment to letter from B. McCaffery of Shoreham Nuclear Power Station to H. Denton, NRC, August 18, 1981.

13. Memorandum to Carl Berlinger, CPB, NRR, and Faust Rosa, ICSB/NRR, from N. Kondic, ICB, DFO, "Two Phase Fluid Water Level in Nuclear Vessels (Reactor SG, PZR), November 23, 1981.

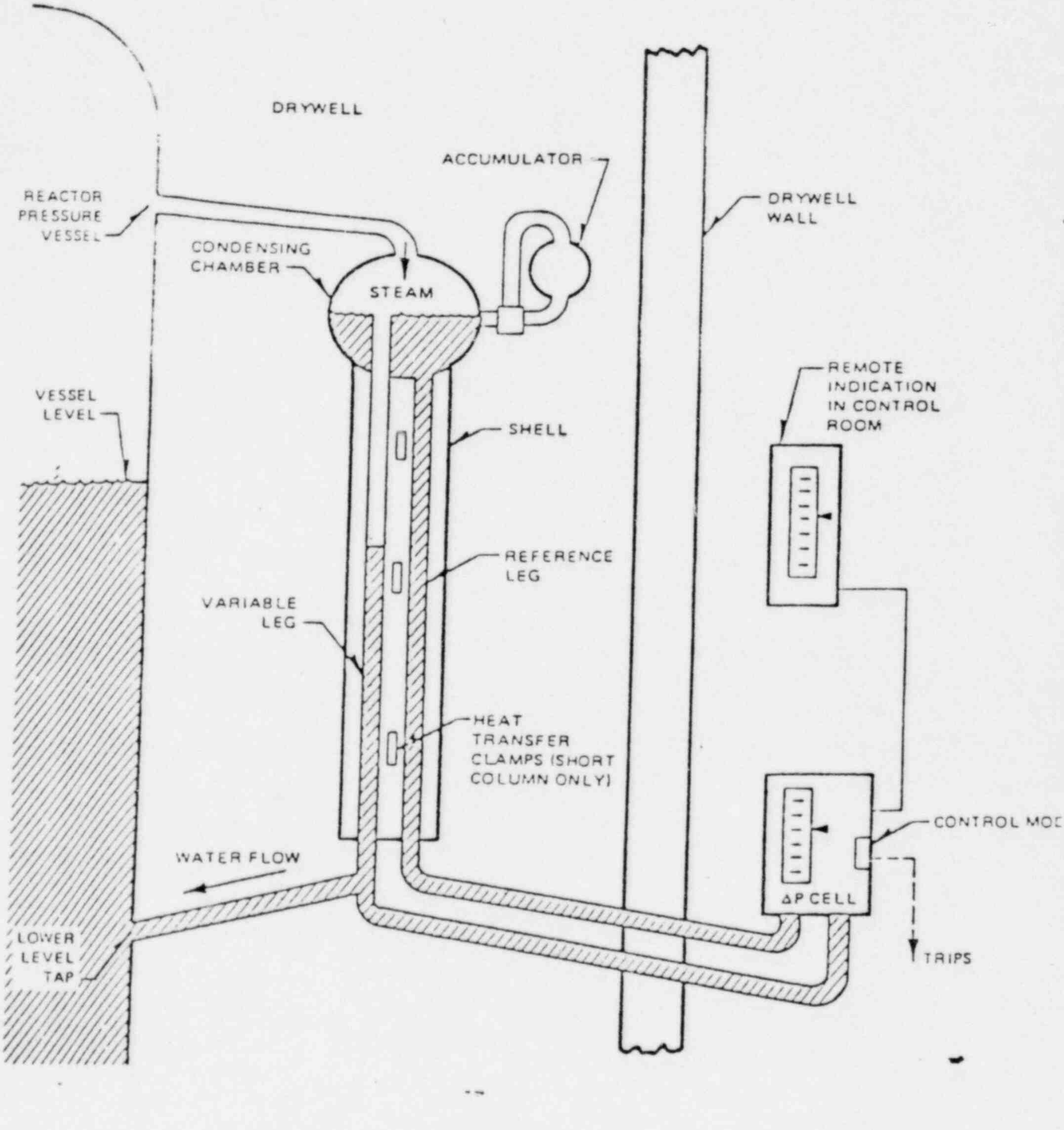

Figure 1. Yarway (Heated Reference Leg) Level Detection Instrucment (From NEDO-24708A)

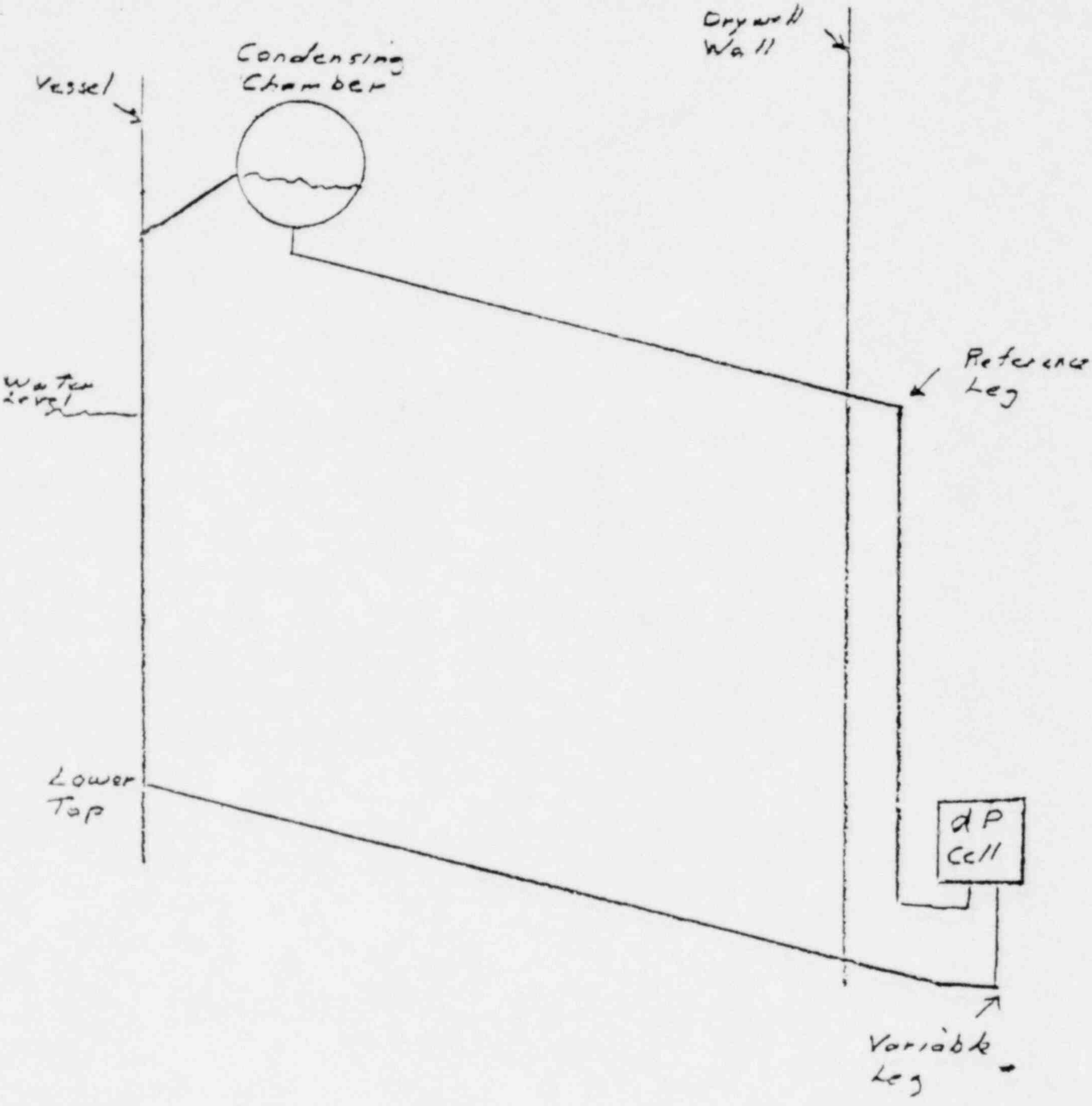

Figure 2. Cold Reference Leg Level Detection Instrument

EXHIBIT 6

Weekly Information Report, Week Ending January 22, 1982, dated January 27, 1982 -- Cover Sheet and Enclosure K.

January 27, 1982

For: The Commissioners

From: T. A. Rehm, Assistant for Operations, Office of the EDO

Subject: WEEKLY INFORMATION REPORT - WEEK ENDING JANUARY 22, 1982

A summary of key events is included as a convenience to those Commissioners who may prefer a condensed version of this report.

T. A. Rehm, Assistant for Operations
Office of the Executive Director
for Operations

*No input this week.

Contact:
T. A. Rehm, EDO
49-27781

FOR SUBSCRIBERS ONLY

OFFICE FOR ANALYSIS AND EVALUATION
OF OPERATIONAL DATA

ITEM OF INTEREST

WEEK ENDING JANUARY 22, 1982

Case Study on BWR Vessel Level Instrumentation

Following completion of the peer review, AEOD has completed a case study on vessel level instrumentation in boiling water reactors (BWRs). The study was initiated following events at Brunswick 1 on January 20, 1981 and Browns Ferry 2 on March 13, 1981.

The study included the review of a number of operating reactor events involving BWR vessel level instrumentation. The review has shown several cases where interaction between plant control systems and protection systems are evident. Our evaluation of these cases has raised the safety concern of a single random failure in the vessel level instrumentation system causing a control system action that could (1) result in a station condition requiring protective action and, at the same time, (2) prevent proper action of some of the protection system channels designed to protect against such a condition, leaving the remaining protection system channels to provide the protective function. A further single active failure in the remaining channels could then prevent the required protective actions.

The study addresses the interaction between feedwater control, reactor protection, containment isolation and emergency core cooling systems and includes findings and recommendations regarding these systems and the safety concern.

Although the postulated control system or protection system interaction was not considered an immediate concern, AEOD believes that the safety concern and associated problems needs to be addressed. Thus, the report was forwarded to NRR for appropriate action.

ENCLOSURE K