NUREG-0420
Supplement No. 3

Safety Evaluation Report

related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 1983

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission's Safety Evaluation Report (SER) (NUREG-0420) on the application by Long Island Lighting Company (LILCO or applicant) to operate the Shoreham Nuclear Power Station was issued on April 10, 1981. Supplement No. 1 to the Shoreham SER was issued in September 1981, and Supplement No. 2 was issued in February 1982.

Each of the sections in this Supplement No. 3 is numbered the same as the section of the SER that is being updated. The discussions in this report are supplementary to and not in lieu of the discussions in the SER, except where specifically noted.

Copies of this report are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C. 20555 and at the Shoreham-Wading River Public Library, Route 25A, Shoreham, New York 11786. Copies of this report are also available for purchase from the sources indicated on the inside front cover.

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1.7 Outstanding Issues

In Section 1.7 of the SER, the staff identified 61 outstanding issues which were not resolved at the time of issuance of the SER. This report discusses the resolution of a number of these items previously identified as open. The items identified in Section 1.7 of the SER are listed below with status of each item. If the item is discussed in this supplement, the section where the item is discussed is identified. The resolution of the remaining outstanding issues will be discussed in future supplements to the SER.

Item		Status	Section
1.	Pool Dynamic Loads	Resolved	3.8.2
2.	Masonry Walls	Resolved	
3.	Piping Vibration Test Program - Small Bore Piping/Instrumentation Lines	Resolved	
4.	Piping Vibration Test Program - Safety-Related Snubbers	Resolved	
5.	LOCA Loadings on Reactor Vessel Supports and Internals	Resolved	
6.	Downcomer Fatique Analysis	Resolved	3.9.2.1
7.	Piping Functional Capability Criteria	Resolved	
8.	Dynamic Qualification	Partially resolved, awaiting further information	3.10
9.	Environmental Qualification	Partially resolved, awaiting further information	3.11
10.	Seismic and LOCA Loadings	Resolved pending confirmation	
11.	Supplemental ECCS Calculations with NUREG-0630 Model	Resolved with license condition	
12.	ODYN-Generic Letter 81-08	Resolved	

Item	Status	Section
13. NUREG-0619, Feedwater Nozzle and Control Rod Return Line Cracking Generic Letter 81-11	Resolved	
14. Jet pump Holddown Beam	Resolved	
15. Inservice Testing of Pumps and Valves	Resolved	3.9.6
16. Leak Testing of Pressure Isolation Valves	Resolved	
17. SRV Surveillance Program	Resolved	
18. NUREG-0313, Revision 1	Resolved	
19. Preservice Inspection	Resolved pending confirmation	
20. Appendix G - IV.A.2.a	Resolved	
21. Appendix G - IV.A.2.c	Resolved	
22. Appendix G - IV.A.3	Resolved	
23. Appendix G - IV.B	Resolved	
24. Appendix H - II.C.3b	Resolved	
25. RCIC	Resolved	
26. Suppression Pool Bypass	Resolved	6.2.1.7
27. Steam Condensation Downcomer Lateral Loads	Resolved	
28. Steam Condensation Oscillation and Chugging Loads	Resolved pending confirmation	
29. Quencher Air Clearing Load	Resolved	
30. Drywell Pressure History	Resolved	
31. Impact Loads on Grating	Resolved	
32. Steam Condensation Submerged Drag Loads	Resolved pending confirmation	
33. Pool Temperature Limit	Resolved	
34. Quencher Arm and Tie-Down Loads	Resolved	
35. Containment Isolation	Resolved	

Item		Status	Section
36.	Containment Purge System	Resolved	6.2.3.1
37.	Secondary Containment Bypass Leakage	Resolved	
38.	Fracture Prevention of Containment Pressure Boundary	Resolved	
39.	Emergency Procedures	Awaiting further information	
40.	LOCA Analyses	Resolved	
41.	LPCI Diversion	Resolved	
42.	Flow Meter	Resolved	
43.	Loss of Safety Function After Reset	Resolved	
44.	Level Measurement Errors	Resolved	
45.	Fire Protection	Resolved	
46.	IE Bulletin 79-27	Resolved pending confirmation	
47.	Control System Failures	Awaiting further information	
48.	High Energy Line Breaks	Awaiting further information	
49.	DC System Monitoring	Resolved	
50.	Low and/or Degraded Grid Voltage Condition	Resolved	
51.	Fracture Toughness of Steam and Feedwater Line Materials	Resolved	
52.	Management Organization	Resolved	13.1.3.1, 13.2
53.	Emergency Planning	Under review	13.3
54.	Security	Resolved	
55.	Q-List	Resolved	
56.	Financial Qualification	Resolved	

Item	Status	Section
57. TMI-2 Requirements:		
Shift Technical Advisor	Resolved with license condition	
Shift Supervisor Administrative Duties	Resolved	
Shift Manning	Resolved	
Upgrade Operator Training	Resolved	
Training Programs - Operators	Resolved pending confirmation	
Revise Licensing Examinations	Resolved	
Organization and Management	Resolved	
Procedures for Transients and Accidents	Resolved	
Shift Relief and Turnover Procedures	Resolved	
Control Room Access	Resolved	
Dissemination of Operating Experiences	Resolved	
Verify Correct Performance of Operating Activities	Resolved	
Vendor Review of Procedures	Resolved pending confirmation	
Emergency Procedures	Resolved	13.5.2
Control Room Design Review	Resolved pending confirmation	I.D.1
Training During Low-Power Testing	Resolved	
Reactor Coolant System Vents	Resolved	
Plant Shielding	Resolved	
Postaccident Sampling	Staff position	
Degraded Core Training	Resolved	
Hydrogen Control	Resolved	
Relief and Safety Valves	Resolved pending confirmation	

Item	Status	Section
Valve Position Indication	Resolved	
Dedicated Hydrogen Penetrations	Resolved	
Containment Isolation Dependability	Resolved with license condition	II.E.4.2
Accident-Monitoring Instrumentation		
Attachment 1	Resolved with post-implementation review	II.F.1
Attachment 2	Resolved	
Attachment 3	Resolved	
Attachment 4	Resolved	
Attachment 5	Resolved	II.F.1
Attachment 6	Resolved	
Inadequate Core Cooling	License condition	
IE Bulletins		
Item 5	Resolved pending confirmation	
Item 10	Resolved pending confirmation	
Item 22	Resolved	
Item 23	Resolved	
Bulletins and Order Task Force		
Item 3	Resolved	
Item 13	Resolved pending confirmation	
Item 16	Resolved pending confirmation	
Item 17	Resolved	
Item 18	Resolved	

Item		Status	Section
	Item 21	Resolved	
	Item 22	Resolved	
	Item 24	Resolved	
	Item 25	Resolved	
	Item 27	Resolved	
	Item 28	Resolved with license condition	II.K.3.28
	Item 30	Resolved	
	Item 31	Resolved	
	Item 44	Resolved	
	Item 45	Resolved	
	Item 46	Resolved	
	Emergency Preparedness - Short Term	Under review	13.3
	Upgrade Emergency Support Facilities	Under review	
	Emergency Preparedness - Long Term	Under review	13.3
	Primary Coolant Outside Containment	Resolved	
	Improved Iodine Monitoring	Resolved	
	Control Room Habitability	Resolved pending confirmation	
58.	Reactor Vessel Materials Toughness	Resolved	
59.	Control of Heavy Loads - Generic Letter 81-07	Resolved	
60.	Station Blackout - Generic Letter 81-04	Resolved pending confirmation	
61.	Scram System Piping	Under review	
62.	Remote Shutdown System	Resolved pending confirmation with license condition	7.4.3

Item	Status	Section
63. Design Verification	Under review	
64. Loose Parts Monitoring System	Awaiting further information	

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

Supplement No. 1 to the SER indicated that the applicant is committed to revise the Design Assessment Report (DAR) to address the generic Long-Term Program (LTP) condensation oscillation and chugging load methodologies and is also committed to the generic LTP as a whole for confirmatory checking.

In August 1981, NRC issued a report entitled "Mark II Containment Program and Acceptance Criteria" (NUREG-0808). This report provides the staff's evaluation of the Mark II Owners' Long-Term Program. It identifies and provides the bases for generic loss-of-coolant accident (LOCA)-related pool dynamic loads that the staff finds acceptable for the evaluation of plants with Mark II containment designs.

On the basis of NUREG-0808, the applicant has made a reevaluation of the design adequacy of the Shoreham containment structure, and the results of this reevaluation are reported in Revision 5 to the DAR (in Appendix L).

Even though in some cases the reevaluation indicates higher containment structure responses than previously computed, the applicant indicated that the basic conclusion is still valid; that is, that all hydrodynamic LOCA load effects are less severe than the long-term quasi-static pressure and temperature effects which remain the controlling LOCA loads for the containment structure. On the basis of its evaluation of the information provided by the applicant, the staff concludes that the design adequacy of the Shoreham containment structure for safety-relief valve (SRV)- and LOCA-related loads is ensured, and that the applicant has fulfilled the confirmatory check commitment.

3.8.2 Concrete and Structural Steel Internal Structures

Section 3.8.2 of the SER and Supplement No 1 to the SER indicated that the applicant had not completed the reevaluation of the effects of pool dynamic loads on the drywell floor and its support columns, platforms, ladders, and walkways, and on cable tray and conduit supports. In Revision 5 to DAR the applicant provided an assessment of the design adequacy of these structures for the SRV and LOCA loads. The assessment is made on the basis of the load combinations and acceptance criteria as established by the staff. The results of the assessment indicate that most of the structures have adequate design margin, and some of the structures which have not met the criteria have been modified or strengthened. The modification and strengthening range from the removal of grating within the pool-swell zone to installation of additional members to the provision of vertical restraint from the drywell floor slab adjacent to the reactor pedestal.

On the basis of the review of the information provided by the applicant, the staff concludes that containment internal structures are capable of resisting all the potential loads including SRV- and LOCA-related loads. The staff considers this item resolved.

3.9 Mechanical Systems and Components

3.9.2 ASME Code Class 2 and 3 Components

3.9.2.1 Design Load Combinations and Stress Limits

Section 3.9.2.1 of SER Supplement No. 1 discusses the staff evaluation of the preliminary results of ASME Code* Class 1 fatigue analyses of the Class 2 and 3 downcomers and SRV discharge piping in the wetwell air space. The staff considered the issue resolved pending receipt and documentation of the final results of these analyses.

In Revision 5 of the DAR, the applicant has provided the final results of the fatigue analyses of the downcomers and SRV discharge piping. The results show that the cumulative usage factors (CUF) for the downcomers and SRV discharge piping at the location of maximum fatigue are within the ASME Code allowable value of 1.0.

The maximum CUF calculated for the downcomers was 0.26 at the drywell floor anchor. The maximum CUF calculated for the SRV discharge piping was 0.76 at the location of the piping schedule (wall thickness) change.

Based on its review of the above fatigue results, the staff concludes that there is reasonable assurance that steam bypass will not occur as a result of a fatigue failure in the Shoreham downcomers and SRV discharge piping in the wetwell air space. The staff considers this issue to be resolved.

3.9.6 Inservice Testing of Pumps and Valves

Sections 3.9.2 and 5.2.1 of the SER discuss the design of safety-related pumps and valves in the Shoreham plant. The load combinations and stress limits used in the design of pumps and valves ensure that the component pressure boundary integrity is maintained. In addition, the applicant will periodically test and measure all safety-related pumps and valves. These tests and measurements are performed in general accordance with the rules of Section XI of the ASME Code. The tests verify that these pumps and valves operate successfully when called upon. Periodic measurements are made of various parameters, and these are compared to baseline measurements to detect long-term degradation of pump or valve performance. The staff review under Standard Review Plan (SRP) Section 3.9.6 covers the applicant's program for the inservice testng of pumps and valves. Particular attention is given to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

In Section 5.2.2 of SER Supplement No. 1, the staff identified an issue regarding the applicant's commitment to submit the inservice testing program for pumps and valves (item 15 in Section 1.7). In a letter from J. L. Smith (LILCo) to

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*The Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers

H. R. Denton (NRC) dated January 6, 1982, the applicant submitted a program entitled "Pump and Valve Inservice Testing Program Plan."

The staff has focused its review on those areas of the test program for which the applicant has requested relief from the requirements of Section XI of the ASME Code. The staff has identified one issue regarding the applicant's relief request for the testing of containment isolation valves.

The applicant proposes to perform a leak rate test according to Appendix J of 10 CFR 50 in lieu of the leak rate testing according to IWV-3420 of ASME Code Section XI for all containment isolation valves. The staff's position is that, for Category A and AC valves which form the interface with the reactor coolant system (RCS) and low pressure systems, the applicant must perform this leak rate test at system functional differential pressures. However, in many cases it is not practical for the applicant to test at functional differential pressure, and in those cases the Code allows testing at a lower pressure (such as those established for Appendix J) provided that the results are extrapolated in accordance with the Code. The basis for testing at functional differential pressure is that these pressure isolation valves are relied upon to isolate the reactor coolant system for a LOCA outside containment. In addition, Appendix J of 10 CFR 50 does not specify acceptance criteria for each valve but is concerned with the total leakage from the containment.

Therefore, the Technical Specifications will require that the reactor system coolant leakage be limited to 1 gpm leakage at reactor coolant system pressure for those pressure isolation valves listed in the Technical Specifications.

The staff has not completed its detailed review of the applicant's submittal. However, based on a preliminary review, the staff finds that it is impractical with the limitations of design, geometry, and accessibility for the applicant to meet certain ASME Code requirements. Imposition of those requirements would, in the view of the staff, result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. The relief requested will not endanger life or property and is in the public interest. Therefore, pursuant to 10 CFR 50.55a, the relief that the applicant has requested from the pump and valve testing requirements of 10 CFR 50.55(g)(2) and (g)(4)(i) is justified for that portion of the initial 120-month period during which the staff completes its review. On this basis, the staff considers this item resolved.

3.10 Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment

The staff evaluation of the adequacy of the applicant's program for qualification of safety-related electrical and mechanical equipment for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an onsite audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

As stated in the previous SER supplement, the Seismic Qualification Review Term (SQRT) conducted a site audit April 6 to 10, 1981 and found that motor-operated valves with Limitorque operators had not been fully qualified to seismic and hydrodynamic loads; as a result, only about 40% of the total safety-related equipment was actually qualified at the time of the audit. In addition, the SQRT found that auditable links did not exist for most of the equipment qualification documents that were audited. Based on the above general finding, the staff considered the extent of completion of the applicant's qualification program insufficient for the staff to draw any conclusions regarding the acceptability of all the safety-related equipment. The staff again reviewed the progress of the program and, based on the applicant's submittal of July 26, 1982, determined that the program was near completion. The SQRT then made a second site audit August 31 to September 3, 1982 to determine the extent to which the qualification of equipment as installed in Shoreham meets the intent of the current licensing criteria as described in RGs 1.61, 1.92, and 1.100, and SRP 3.10 (NUREG-0800) and would provide adequate assurance that such equipment will function properly under all imposed design and service loads including the loadings imposed by the safe shutdown earthquake (SSE), postulated accidents, and loss-of-coolant accidents (LOCAs). This program, if in conformance with the above criteria, will constitute an acceptable basis for satisfying the applicable requirements of GDC 2, 4, 14, and 30 and paragraphs XI of Appendix B to 10 CFR 50 and VI(a)(1) and (2) of Appendix A to 10 CFR 100 as they relate to qualification of equipment.

Twelve pieces of nuclear steam supply system (NSSS) and balance of plant (BOP) equipment were selected before the audit for detailed review. At the plant site, three additional pieces were selected for detailed review, and four pieces were selected for document review only. The review consisted of field observations of the actual equipment configuration and its installation, followed by a review of the corresponding qualification documents.

In this audit, the SQRT also reviewed the extent to which the Shoreham Mark II hydrodynamic loads confirmatory program was incorporated in the applicant's equipment seismic and dynamic qualification program. The objective of this confirmatory program is to evaluate the adequacy of the plant for final generic Long-Term Program (LTP) LOCA steam condensation and safety-relief valve (SRV) discharge load definitions, which had previously been designed to the Shoreham design-basis loads.

For the 15 pieces of the equipment selected for detailed document review and field examination, the staff found their qualification acceptable relative to the Shoreham design-basis loads, with the exception of certain details which need to be clarified by the applicant (letter from A. Schwencer to M. S. Pollock dated November 10, 1982). The information on confirmatory loads, however, was generally not available for review at the site. This same situation was found regarding the four pieces of equipment that were selected for review completeness of qualification documentation only.

In the SQRT onsite audit exit meeting and in the trip report, the staff concluded that, to complete its review, the applicant must provide additional information and to clarify the details of the qualification for some pieces of equipment. The trip report includes both generic and equipment-specific concerns. The generic concerns were considered more significant, because they apply to all safety-related equipment items.

In a letter dated November 23, 1982, the applicant provided responses to the above-identified concerns. This information--together with information provided by the applicant in a telephone conference on December 7, 1982 and in a meeting with the applicant on December 15, 1982 at Brookhaven National Laboratory--provided the basis for the staff conclusion on the resolution of certain items of concern. The following paragraphs discuss the actions on generic items the applicant is required to perform, as originally defined in the trip report, and the information provided by the applicant, as well as the staff review finding.

The staff indicated that the applicant should improve qualification documentation as follows:

(1) Provide a road map to define the qualification process for the balance of plant (BOP) equipment.

(2) Include complete test reports in BOP SQRT package.

(3) Clarify that the single spectrum included in SQRT package is the limiting (worst case) spectrum.

The applicant's response committed to the following:

(1) The equipment documentation will be augmented by a road map to define the scope of the dynamic qualification program for BOP equipment. It will be completed 30 days before fuel load.

(2) In a few cases where vendors have not supplied a complete test report as part of their qualification documentation (e.g., 480-v Emergency Switchgear Bus 112 and 480-V Motor Control Centers 1R24* MCC 1120) and in others that were not audited, complete test reports will be requested from the vendor. If the vendor holds them as proprietary, the applicant will review them again to document the substance of the test conducted. Any test anomalies that may have occurred will also be documented with their resolutions and included in the SQRT package on file at the site. This effort will be completed by June 1983, and NRC will be notified in writing.

(3) By fuel load, the single spectrum included in the SQRT packages will be identified as the limiting (worst case) spectrum.

This response has been reviewed by the staff, and the commitments are acceptable.

The staff also indicated that the latest confirmatory load spectra should be inlcuded in all SQRT packages by the end of March 1983.

The applicant's response committed that the latest confirmatory load spectra will be included in all SQRT packages for floor-mounted equipment by the end of March 1983. Clarification will be provided to the extent necessary to relate confirmatory load spectra to the qualification basis. All replacement equipment (not in kind) will be qualified to the confirmatory load spectra. The staff finds this commitment acceptable.

The staff also requested the applicant to consider the latest confirmatory loads for the qualification of pipe-mounted equipment (i.e., valves) as follows:

(1) Phase I - Before fuel load

- Provide a description of the results of 30 piping subsystems already analyzed.
- Provide a list of pipe-mounted equipment by Shoreham Valve Mark Number in these subsytems.
- Demonstrate qualification to confirmatory load values for the valves listed.

(2) Phase II - Before operation above 5% power

- Identify all associated pipe-mounted equipment for approximately 70 additional piping subsystems.
- Assess the existing margin of safety for accommodating the upper bound of any load increase that could result from the confirmatory loads.
- Where adequate margins of safety are not evident, perform analyses to demonstrate equipment qualification utilizing confirmatory loads.

The applicant has committed to the following program to evalute the effect of the hydrodynamic LOCA loads discussed in NUREG-0808 on pipe-mounted equipment:

(1) Phase I

In the Shoreham Design Assessment Report (DAR), Revision 5, Appendix L, the applicant stated that a representative cross-section of primary and secondary piping was evaluated to the NUREG-0808 confirmatory load definition. This cross-section consisted of 30 piping subsystems, 25 of which are attached to the primary containment at locations of high amplitude response spectra. In an attachment to a letter of November 23, 1982, the applicant provided a listing of all the pipe-mounted equipment on these 30 piping subsystems by Shoreham Mark Number. The applicant has agreed to provide the qualification level and calculated acceleration for each item, based on the NUREG-0808 confirmatory load definition. Should any equipment acceleration levels be found to be above the present qualification levels, computer reanalysis of these pieces of equipment will be performed, utilizing the NUREG-0808 confirmatory load definition but eliminating the simplifying assumptions that have been employed. This analysis will be completed before fuel load.

(2) Phase II

The applicant has committed to perform a 100% reevalution to the final Mark II LTP load definition (NUREG-0808) of the piping attached to the primary containment at the three additional locations of concern--21 ft, 83 ft, and 106 ft--where high stress occurred, as indicated in the applicant's letter of August 20,

1982. This reevaluation is considered confirmatory by the staff and is not required for fuel load and low-power testing. The applicant also committed to identify all pipe-mounted equipment on these additional piping systems (approximately 70 piping subsystems) and to determine if the existing margins of safety are sufficient to accommodate the upper bound of any load increase that could result from the confirmatory NUREG-0808 load definitions. For the set of equipment where adequate margins of safety are not evident, requalification will be performed utilizing the NUREG-0808 confirmatory load definition. This analysis will be completed before 5% power operation is exceeded. The staff finds the applicant's commitment acceptable.

The staff also required and the applicant committed to establish, before fuel load, a maintenance and surveillance program to maintain equipment in a qualified status throughout the plant life. The applicant has stated that the plant surveillance and maintenance program includes documented program plans, procedures, and results to ensure that the safety-related equipment identified in the dynamic qualification program is maintained in a state of readiness and operability so that it will perform its intended safety functions properly during and after the excitation imposed by the SSE, or hydrodynamic loads associated with suppression pool discharges, or a combination of the two. So the staff can be assured of the applicability of the above surveillance and maintenance program to the equipment seismic and dynamic qualification program, the applicant agreed to provide, before fuel load, a sample surveillance and maintenance prog am, including qualified life, for such age-sensitive equipment as electric motors and batteries. The staff finds this acceptable.

The staff also required and the applicant committed to provide a periodic status of equipment summary list and justification for the equipment that will be qualified after fuel load. In response, the applicant has indicated he will update equipment summary lists, including BOP and nuclear steam system supplier (NSSS) equipment, and provide these lists for the staff's information on a monthly basis. The applicant also agreed to provide, before fuel load, justification for interim operation for equipment that will not be qualified after fuel loading. The staff finds applicant's commitment acceptable. Before power operation beyond 5%, all safety-related equipment will have to be completely qualified.

The staff required that the applicant establish a qualification documentation file in the Shoreham plant file system by June 1, 1983. The SQRT audited the BOP qualification documentation file during the site visit and found it acceptably established in the plant file system. For the NSSS scope of supply, the applicant at present has on file at the Shoreham site NSSS equipment dynamic qualification summaries. These summaries provide the requirements, demonstrate equipment capability, and provide a rationale for qualification certification along with the qualification summary of equipment (SQRT) forms. The applicant is in the process of acquiring the backup qualification documentation. The detailed NSSS backup qualification documentation will be located in the SQRT documentation packages at the Shoreham site by June 1983. The staff finds this commitment for the NSSS file acceptable.

The staff required that the applicant provide a clarification that margin to cover uncertainty in manufacturing and test exists for equipment qualified by test, in accordance with IEEE 323-1974. IEE 323-1974, which specifies a margin requirement, is endorsed by the NRC in RG 1.89, "Qualification of Class 1E

Equipment for Nuclear Power Plants," dated November 1974. The Shoreham FSAR Appendix 3B indicates that RG 1.89 is not applicable to Shoreham because the Shoreham SER (dated February 20, 1970) preceded the implementation date given in Section D of RG 1.89; hence the applicant states that IEEE 323-1971 is the applicable standard for equipment qualification for Shoreham. The applicant feels that, with the Test Response Spectra (TRS) enveloping the Required Response Spectra (RRS) and the inherent conservatism used in developing the RRS, adequate margin is provided to cover uncertainty in manufacturing and errors for equipment qualified by test. Based on the above and the review finding of the audited equipment qualification documents, the staff finds the applicant's response acceptable.

The staff also required the applicant to discuss the cycling effects of hydrodynamic loads based on worst case consideration for the following:

(1) For equipment qualified by analysis, demonstrate that the cumulative fatigue usage factor is less than one.

(2) For equipment qualified by testing, define the number of equivalent safety relief valve (SRV) cycles.

The applicant's response indicates the following:

(1) For BOP equipment qualified by analysis, a survey will be conducted to identify the most highly stressed equipment in several categories, i.e., pumps, valves, heat exchangers, and tanks. Peak stress will be determined by applying a stress intensification factor applicable to the configuration. The applicant will provide cumulative fatigue usage factors for each equipment category.

    Vibration fatigue cycle effects for NSSS equipment designed to ASME Code requirements were reviewed at General Electric (GE) by NRC consultants from Battelle Pacific Northwest Laboratories on October 7, 1980. The consultants indicated they were satisfied with the GE approach, which encompasses operating basis earthquake, SRV, thermal, and pressure cycles. The applicant was requested and agreed to provide sample calculations of usage factors for both ASME Code and non-ASME Code components.

(2) For equipment qualified by testing, the number of equivalent SRV cycles has been defined in the Design Assessment Report, Revision 5, pages 9-11 and 9-12. The applicant will clarify how the fatigue testing was actually conducted in assuring that the TRS envelop the RRS and that the input loads are sufficient to cover the duration and number of SRV cycles that have been defined.

    The staff finds these commitments acceptable provided the required tasks are accomplished before power operation beyond 5%.

The staff also required and the applicant committed to provide information on any field modifications made to already qualified and installed equipment before fuel load. The applicant has provided the current status of field modifications to safety-related BOP equipment made since September 2, 1982, the site audit date. As of November 12, 1982, GE records do not indicate that there were any field changes made of NSSS equipment since September 2, 1982 that would affect seismic qualification as documented in GE SQRT reports. The applicant will provide a revised list of BOP and NSSS equipment before fuel load. The staff finds the above commitment acceptable.

In regard to the equipment-specific concerns as identified in the SQRT trip report, the information provided by the applicant--in a letter of November 23, 1982; the telephone conference of December 7, 1982; and the meeting of December 15, 1982--has resolved the staff concerns, with the exception of the following which are to be resolved as indicated:

(1) 480-v emergency switchgear bus 112

The applicant is to provide a test report for confirmatory review of the anomalies observed during the qualification testing by 30 days before fuel load.

(2) Adequacy of the single frequency/single axis testing method

The applicant is to provide detailed technical justification for the use of the single frequency/single axis testing method for the following representative equipment items, or their equivalent, by 30 days before fuel load.

- local panel devices B21-No55 (163C1292)
- transmitter, gage press-ship loose devices (163C1564)
- Limitorque actuator recirculation discharge valve, B31-F031

The applicant has defined a comprehensive program for seismic and dynamic qualification of safety-related electrical and mechanical equipment. This program generally meets the intent of current criteria indicated in SRP 3.10. The staff considers that the extent of the implementation of the program will be acceptable for interim operation at power levels not exceeding 5% of full power, pending resolution of the previously identified generic and specific items, in accordance with the schedules commited to by the applicant.

3.11 Environmental Qualification of Safety-Related Equipment

3.11.1 Introduction

Equipment which is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement--which is embodied in General Design Criterion (GDC) 1 of Appendix A to Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), "Quality Standards and Records," and GDC 4, "Environmental and Missile Design Bases," and in Sections III, XI, and XVII of Appendix B to 10 CFR 50--is applicable to equipment located inside as well as outside containment. More detailed guidance relating to the methods and procedures for demonstrating this capability is in NUREG-0588, "Interim Staff Position on

Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE Standard 323, and various NRC regulatory guides and industry standards.

3.11.2 Background

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions in this report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods which are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation.

Commission Memorandum and Order CLI-80-21, issued on May 23, 1980, states that NUREG-0588 forms the requirements that license applicants must meet to satisfy those aspects of GDC 4 that relate to environmental qualification of safety-related electrical equipment. IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," issued January 14, 1980, and its supplements dated February 29, September 30, and October 24, 1980 established environmental qualification requirements for operating reactors. This bulletin and its supplements were provided to operating license applicants for consideration in their review.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B of 10 CFR 50. The qualification methods defined in NUREG-0588 are general and can also be applied to mechanical equipment.

In response to the above, the applicant provided equipment qualification information by letters dated May 27, 1981 and January 25, May 17, July 8, July 23, September 9, October 29, and November 3, 1982 to supplement the information in FSAR Section 3.11.

3.11.2.1 Purpose

The purpose of the following discussion is to evaluate the adequacy of the Shoreham environmental qualification program for safety-related equipment.

3.11.2.2 Scope

The scope of this review includes an evaluation of the list of systems and equipment to be qualified, the criteria they must meet, and the environments in which they must function, and an assessment of the qualification documentation for equipment. It is limited to safety-related equipment that must function to prevent or mitigate the consequences of design-basis LOCAs or high- or moderate-energy line breaks, inside or outside of containment, while it is subjected to the harsh environments associated with these accidents.

3.11.3 Staff Evaluation

The staff evaluation of the applicant's response included an onsite examination of equipment, audits of qualification documentation, and a review of the applicant's submittals for the acceptability of systems and components, qualification methods, and accident environments. The criteria described in SRP 3.11 (NUREG-

0800); Sections III, XI, and XVII of Appendix B to 10 CFR 50; and NUREG-0588, Category II form the basis for the staff evaluation of the adequacy of the applicant's qualification program. Revision 1 of NUREG-0588 was utilized to clarify staff positions as required.

The staff performed audits of the applicant's qualification documentation and installed electrical equipment on April 26-30 and June 2-3, 1982. The audits consisted of a review of approximately 20% of the applicant's equipment. Qualification documentation for mechanical equipment was also reviewed by the staff.

3.11.3.1 Completeness of Safety-Related Equipment

The applicant was directed to (1) establish a list of systems and components that are required to prevent or mitigate the consequences of a LOCA or a high-energy line break HELB and (2) identify components needed to perform the function of safety-related display instrumentation, post-accident sampling and monitoring, and radiation monitoring.

The applicant's systems list in the environmental qualification program was compared to FSAR Table 3.2.1-1. The subset of systems from Table 3.2.1-1 that is required for emergency shutdown or accident mitigation was reviewed by the staff and found to be acceptable. The staff also reviewed and found acceptable, based on an audit review, the applicant's operability times, Class 1E safety functions, and required accidents for selected systems and components.

In addition, the staff reviewed the list of equipment in a harsh environment and determined which systems on the master list (harsh and mild) had been omitted. The omissions were adequately explained as systems located only in a mild environment.

Electrical equipment in a harsh environment specified by RG 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and NUREG-0737, "Clarification of TMI Action Plan Requirements," was identified by the applicant and included in the environmental qualification program. SECY 82-111, "Requirements for Emergency Response Capability," dated March 11, 1982, and approved by the Commission on July 16, 1982, defines the implementation of recommendations in RG 1.97, Revision 2. The staff review of RG 1.97, Revision 2 implementation may not be completed before final licensing of Shoreham. In the interim, the environmental qualification of equipment identified by the applicant has been reviewed in the same manner as other safety-related equipment.

The applicant identified 131 types of electrical equipment that were assessed by the staff. Of these, 50 are conditionally qualified, and of the remaining 81, most are either being retested or replaced. Documentation for eight items of mechanical equipment was also reviewed by the staff.

3.11.3.2 Qualification Methods

3.11.3.2.1 Electrical Equipment

Detailed procedures for qualifying safety-related electrical equipment are defined in NUREG-0588. The Category II requirements that are applicable to Shoreham are based on and supplement IEEE Standard 323-1971. The staff audits of qualification documentation verified that acceptable methods have been employed in the Shoreham program.

3.11.3.2.2 Mechanical Equipment

Although there are no detailed requirements for qualification of safety-related mechanical equipment, GDC 1 and 4 and Sections III and XVII of Appendix B to 10 CFR 50 contain the following requirements related to equipment qualification:

- Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.

- Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

- Design control measures shall be established for verifying the adequacy of designs.

- Records affecting quality shall be maintained and shall include the results of tests and materials analyses. The records shall also include data on the qualification of equipment.

The staff review of the environmental qualification of mechanical equipment has concentrated on materials that are sensitive to environmental effects, for example, seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms. Qualification documentation has been reviewed by the staff and conformance with the above criteria verified.

3.11.3.3 Service Conditions

NUREG-0588 defines the methods to be utilized for determining the environmental conditions associated with LOCAs or HELBs, inside or outside of containment. The review and evaluation of the adequacy of these environmental conditions are described below. The staff has reviewed the qualification documentation to ensure that the qualification conditions envelop the conditions established by the applicant.

3.11.3.3.1 Temperature, Pressure, and Humidity Conditions Inside the Primary Containment

The applicant provided the LOCA/MSLB profiles used for equipment qualification in the program submittal. The peak values in the drywell resulting from these profiles are as follows:

	Maximum Temperature, °F	Maximum Pressure, psig	Humidity, %
LOCA/MSLB	340°F	48	100

The staff has reviewed these profiles and finds them acceptable for use in equipment qualification; that is, there is reasonable assurance that the actual pressures and temperatures will not exceed these profiles anywhere within the specified environmental zone (except in the break zone).

In the wetwell, the staff calculated a temperature profile for steam bypass with a peak temperature of 270°F. Equipment in the wetwell was evaluated by the applicant using this profile.

3.11.3.3.2 Temperature, Pressure, and Humidity Conditions Outside the Primary Containment

The applicant has provided the temperature, pressure, and humidity conditions associated with high- and moderate-energy line breaks in the secondary containment. The staff has used a screening criterion of saturation temperature at the calculated pressure to verify that the parameters identified by the applicant are acceptable.

3.11.3.3.3 Submergence

The maximum submergence levels have been established by the applicant in the environmental qualification program and are 63 ft 4 in. in the drywell and 47 ft 0 in. (including froth) in the wetwell during suppression pool swell. Essential equipment below these levels has been qualified.

The effects of flooding on safety-related equipment in the reactor building are discussed in Appendix 3C of the FSAR. This analysis has been reviewed and approved by the staff in Section 3.6 of the SER. Included in the analysis of pipe failures outside containment in Appendix 3C were measures for protecting essential equipment subject to spray or dripping and assuring that essential equipment (other than cable) would not be submerged by flooding to a depth of 22 in. at elevation 8 ft.

3.11.3.3.4 Demineralized Spray

Demineralized water is available for primary containment heat removal following an accident. The applicant included this environmental parameter in the evaluation of equipment qualification.

3.11.3.3.5 Aging

NUREG-0588 Category II delineates two aging program requirements. Valve operators committed to IEEE Standard 382-1972 and motors committed to IEEE Standard 334-1971 must meet the Category I requirements of the NUREG. This requires the establishment of a qualified life, with maintenance and replacement schedules based on the findings. All other equipment must be subjected to an aging program that identifies aging-susceptible materials within the equipment. Additionally, the staff requires the applicant to:

(1) Establish an ongoing program to review surveillance and maintenance records to identify potential age-related degradation.

(2) Establish component maintenance and replacement schedules that include considerations of aging characteristics of installed components.

The applicant has established a qualified life for each qualified equipment type through test or analysis. In addition, the applicant has developed a plan for surveillance and maintenance to ensure that equipment will not degrade sooner than predicted. The program will utilize failure history data from licensee events reports, plant operating experience, manufacturers' recommendations, and pertinent information in the central files. The staff has reviewed this plan and finds it acceptable. Surveillance and maintenance procedures are to be implemented before low power operation is exceeded. The applicant will be required to notify the staff when the procedures are implemented.

The applicant has described a procedure for replacement of equipment and components in Revision 2 of the program. Replacement equipment is to be procured to the Category I requirements in NUREG-0588 unless there are "sound reasons" for not doing so. Replacement components for qualified equipment which are not "in kind" (same manufacturer and part number) shall have a documented evaluation to assure at least equivalent performance. American National Standard ANS-3.2/ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," which is endorsed by RG 1.33, Revision 2, describes additional guidelines for procurement of replacement parts. In a letter dated June 21, 1982, the applicant has committed to follow the guidelines in this regulatory guide and industry standard.

3.11.3.3.6 Radiation (Inside and Outside Primary Containment)

The applicant has provided values for the radiation levels postulated to exist following a LOCA. The application and methodology employed to determine these values were presented to the applicant in NUREG-0588 and NUREG-0737, "Clarification of TMI Action Plan Requirements." The staff review determined that the values to which equipment was qualified enveloped the requirements identified by the applicant.

The values specified in the drywell are integrated doses of 1×10^8 to 1.7×10^8 rads gamma and 1×10^9 rads beta. In the secondary containment, required values of 3.06×10^6 to 1.27×10^8 rads gamma were used in the evaluation of equipment in areas exposed to recirculating fluid lines.

The values used for qualification of equipment are identical to those in the applicant's response to TMI Action Plan Item II.B.2 in the FSAR and are acceptable.

3.11.3.4 Outstanding Equipment

For most items for which there is not complete qualification documentation, the applicant has provided commitments for corrective action and schedules for completion. Where complete qualification documentation will not be available by fuel load, the applicant has provided an analysis for each unqualified electrical equipment item to ensure that the plant can be operated safely pending completion of environmental qualifications. Mechanical equipment justifications must similarly be provided. These analyses must include, as appropriate, consideration of

(1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(2) The validity of partial test data in support of the original qualification.

(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(4) Completion of the safety function prior to exposure to the ensuing accident environment and the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(5) No significant degradation of any safety function or misleading of the operator as a result of failure of equipment under the accident environment.

Before fuel load the staff will evaluate this information to determine whether or not interim operation with this equipment will degrade safety functions or inhibit accident mitigation systems or equipment in the unlikely event of an accident. The staff findings will be published in a supplement to this report.

A justification for interim operation with the GE 200 Series penetrations has not been submitted since the applicant considers this item to be qualified.

The staff, however, has requested additional information to demonstrate qualification. The staff's evaluation of this information will be published in a supplement to this report before fuel load.

3.11.4 Qualification of Equipment

The following subsections present the staff assessment of equipment based on the applicant's submittal, audits of documentation at the plant site, information in the NRC Equipment Qualification Data Bank, and previous staff evaluations of equipment in other plants.

3.11.4.1 Safety-Related Electrical Equipment

The staff has separated the safety-related electrical equipment in a harsh environment into three categories: (1) equipment requiring replacement prior to plant startup, (2) equipment requiring additional qualification information or corrective action, and (3) equipment considered acceptable pending implementation of the maintenance and surveillance program. Tables listing equipment in each of these categories are included as Tables 3.1 through 3.3.

3.11.4.1.1 Equipment Requiring Replacement Prior to Plant Startup

Table 3.1 identifies equipment that the staff review has determined requires replacement prior to plant startup. There is currently no equipment in this category for Shoreham Unit 1.

3.11.4.1.2 Equipment Requiring Additional Information and/or Corrective Action

Table 3.2 identifies equipment in this category. Corrective action or deficiencies are noted by a letter relating to the legend identified below.

Legend

A - material-aging evaluation; replacement schedule; ongoing equipment surveillance

CS - chemical spray

EXN - exempted equipment justification inadequate

H - humidity

I - HELB evaluation outside containment not completed

M - margin

P - pressure

QI - qualification information being developed

QM - qualification method

QT - qualification time

R - radiation

RPN - equipment relocation or replacement; adequate schedule not provided

RPS - equipment relocation or replacement; schedule provided

RT - required time

S - submergence

SEN - separate effects qualification justification inadequate

T - temperature

These deficiencies do not necessarily mean that the equipment is unqualified. However, the deficiencies are cause for concern and require further case-by-case evaluation.

3.11.4.1.3 Equipment Considered Acceptable or Conditionally Acceptable

Based on the staff review, the items identified in Table 3.3 have been determined to be acceptable, pending implementation of the maintenance/surveillance program. Before exceeding low-power operation, the applicant is required to inform the staff of the implementation of the maintenance/surveillance program.

3.11.4.2 Safety-Related Mechanical Equipment

The staff review of the environmental qualification of safety-related mechanical equipment in a harsh environment concentrated on materials sensitive to environmental effects (principally organic materials). Eight equipment items were selected for review of documentation to determine if the applicable portions of Table 3.2 had been properly implemented. The documentation packages consisted of summary sheets, lists of equipment by plant tag number, engineering drawings and bills of materials, materials analyses, and references. This documentation is acceptable and meets the applicable requirements. The applicant has indicated that several items of mechanical equipment are deficient in qualification documentation. These items will either be qualified by fuel load or justified for interim operation.

3.11.5 Conclusions

The staff has reviewed and evaluated the Shoreham program for the environmental qualification of safety-related equipment required for safe shutdown and accident mitigation while exposed to accident conditions. This review has included the systems selected for qualification, the environmental conditions resulting from design basis accidents, and the methods used for qualification.

This review is complete and the applicant's program is acceptable, except for the following outstanding items:

- Qualification of the GE 200 Series electrical penetrations. Additional information has been requested by the staff.

- Review of the justifications for interim operation with documentation deficient equipment. The staff is currently reviewing the justifications provided for electrical equipment. Mechanical equipment justifications will be reviewed after they have been furnished by the applicant.

In addition, the following license conditions will be imposed:

(1) Before exceeding 5% power, complete the implementation of the maintenance and surveillance program as indicated in Section 3.11.3.3.5.

(2) All installed equipment that is required for safe shutdown or mitigation of the consequences of design basis accidents shall be qualified prior to startup from the first refueling outage. Upon completion of qualification, documentation shall be in an auditable file.

Based on these considerations, the staff concludes that satisfactory completion of the corrective actions identified herein will ensure conformance with the requirements in NUREG-0588 and relevant parts of GDC 1 and 4 and Sections III, XI, and XVII of Appendix B, 10 CFR 50 for safety-related equipment in an accident environment. In the interim, the applicant is in compliance with the Commission Memorandum and Order of May 23, 1980 (CLI-80-21) as modified by 10 CFR 50.49 with the exception of the two outstanding items identified above. The staff will provide supplemental evaluations as noted above in this section.

Table 3.1 Equipment requiring replacement prior to plant startup (Category 3.11.4.1.1)

No equipment in this category

Table 3.2 Equipment requiring additional information or corrective action (Category 3.11.4.1.2)

	Equipment	Manufacturer	Model	Deficiency/ corrective action
1.	Motor control centers	General Electric	DC MCC	Retest by 1984
2.	Motor control center	Square D	Model 4	Retest by 1983
3.	Breaker	GE	M26	Modify by 9/82
4.	Solenoid operated valve	ASCO	MV200-926-1F-EP	Replace by 9/82
5.	Temperature control valve	Beck	14-101-023645(ES)	Retest by 2/83
6.	Motor operated valve[1]	Limitorque	SMB Series	QI (84 items)
7.	Motor operated damper	ITT	NH91	Retest by 12/82
8.	Motor operated damper	Raymond	MASR-49	Retest by 12/82
9.	Motor operated damper	Raymond	MASR-9	Replace by first refueling outage
10.	Flow transmitter	Air Monitor Corp.	Veltron 800	Retest by 12/82
11.	Transmitters	Rosemount	1152	Circuit boards to be replaced by 9/82 (18 items)
12.	Pressure switch	ASCO	SB11AKR/TF10A32B	Retest by 9/82
13.	Pressure switch	ASCO	SB11AKR/TG10A32B	Retest by 9/82
14.	DP switch	Dwyer	1627	Retest by 12/82
15.	Level switch	Magnetrol	291-MPG-X-M14DC	Retest by 12/82
16.	Level element	GEMS	XM-54854	Retest by 10/82
17.	Radiation element	Kamen	---	Retest by 8/82
18.	Position switch	Namco	EA740	Replace by 9/82
19.	Position switch	Namco	EA750	Replace by 9/82
20.	Transfer switch	ASCO	307A66C	Retest by 6/82
21.	Selector switch	GE	CR2940	Retest by 8/82

Table 3.2 (continued)

	Equipment	Manufacturer	Model	Deficiencies/ corrective action
22.	Panel	Atomics Int.	---	Retest and replace subcomponents by 9/82
23.	Panel	Gould	5600 Series	QI
24.	Panel	Kamen	---	Retest by 8/82
25.	Pressure Indicator	Marsh Gage	H0212	QI
26.	Panel	Square D	Bkr. Dist.	Retest by 1983
27.	Panel	Square D	480V	Retest by 1983
28.	Flex conduit	Electro Flex	CEA Sealtite	Retest by 8/82
29.	Conduit couplings	Service-Air, Amer. Boa.	---	Retest by 8/82
30.	Penetration	Conax	Low Volt. Power	QI (2 items)
31.	Penetration	GE	Series 200	A, QT, R
32.	Tape	Okonite	T35, T95	Retest by 8/82
33.	Insulating material	Raychem	WCSF-N	Retest by 8/82
34.	Chico compounds	Crouse-Hinds	Chico (X), (A)	QI
35.	Terminal blocks	GE	EB25A04W, -12W	Retest by 8/82
36.	Terminal blocks	GE	EBI	Retest by 8/82
37.	Terminal blocks	GE	CR151	Retest by 8/82
38.	Recombiner	Atomics Int.	---	Retest and replace subcomponents by 9/82
39.	Hydrogen analyzer	Comsip	B	Retest sample pump by 6/82
40.	Oxygen analyzer	Comsip	J	Retest sample pump by 6/82
41.	Filter train	Farr	N-240	Retest by 9/82
42.	Solenoid operated valve	ASCO	HT-X-8320A20	Replace, first refueling outage
43.	Solenoid operated valve	Target Rock	1/2 SMS-A-1	QI
44.	Explosive valve	Conax	1832-159-01	QI
45.	Motor operated valve	Limitorque	SMB-2	QI (2 items)

Table 3.2 (continued)

	Equipment	Manufacturer	Model	Deficiencies/ corrective action
46.	Motor operated valve	Limitorque	SMB-3	Replace actuator by 9/82 (2 items)
47.	Pump motor	GE	CD259A7	Retest by 1/83
48.	Pump motor	GE	1.5HP, 120VDC	QI
49.	Pump motor	GE	3HP, 1900RPM	Retest by 1/83
50.	Pump motor	GE	3HP, 3500RPM	Retest by 1/83
51.	Pump motor	GE	5K324AK2084	QI
52.	E/P converter	Fisher Governor	546	QI
53.	Flow transmitter	Ametek	078-5004	Retest by 3/83
54.	Pressure transmitter	Bailey	KG556	Replace, first refueling outage
55.	Level transmitter	Barton	368	Replace, first refueling outage
56.	Level indicating transmitter	Barton	760	Replace, first refueling outage
57.	Transmitters	Rosemount	1151	QI
58.	Pressure switch	Barksdale	B1T	Retest (5)
59.	Pressure switch	Barksdale	D2H	QI
60.	DP switch	Barton	288A/289A	QI (16 items)
61.	Position switch	Namco	D1200	Retest by 1/83
62.	Position switch	Namco	EA740	Replace by 9/82
63.	Level switch	GE	---	QI
64.	Position switch	Not determined	---	QI
65.	Pressure switch	Square D	9012,ACW-12	Retest by 1/83
66.	Level switch	Square D	9036	Retest by 1/83
67.	Level switch	Square D	9038-AG154	Retest by 1/83
68.	Pressure switch	Static-O-Ring	5N,6N	QI
60.	Radiation element	GE	237X731G001	QI
70.	Flow element	Schutte & Koerting	---	Retest by 3/83
71.	Blower motor	GE	2CH6 041-U	QI
72.	Turbine	Terry	GS-1	Retest by 1/83
73.	Temperature element	Pyco/Calif. Alloy/NECI	145C3224/ 145C3234 Series	QI

Table 3.2 (continued)

	Equipment	Manufacturer	Model	Deficiencies/ corrective action
74.	Panel	GE	---	Retest by 2/83
75.	Panel	GE	M26	Retest by 2/83
76.	Level switch	Magnetrol	5.0-751-2X-MPG-M14HY	QI
77.	Temperature element	Pyco	102-9039-08	QI
78.	Switch	NMC	PMC-8000	QI
79.	Radiation element	NMC	SC-2-15, SC-2B	QI
80.	Temperature element (inside drywell)	Rosemount	89-86-4/88-14-1	QI
81.	Motor starter	Square D	Size 5, 460V	QI

[1]Limitorque valve operators were purchased to various specifications, contain various motors, and reference several test reports. Qualification evaluation is based on these factors and location in the plant.

Table 3.3 Equipment considered acceptale pending implementation of aging program (Category 3.11.4.1.3)

	Equipment	Manufacturer	Model no.
1.	Solenoid operated valve	ASCO	NP8316E36E
2.	Pump motor	GE	5K6339XC157A
3.	Pump motor	GE	5K6339XC94A
4.	Temperature element	Pyco	102-3171
5.	Pressure switch	Barksdale	P1H
6.	Pressure switch (6 items)	Barksdale	B1T
7.	Pressure switch (8 items)	Barton	288A/289A
8.	Level switch	Magnetrol	3.5-751-1X-MPG-M14HY
9.	Solenoid operated valve	ASCO	WJHKX-8320-A89E
10.	Solenoid operated valve	ASCO	WJK-206-380-6F
11.	Motor operated valve (130 items)[1]	Limitorque	SMB Series
12.	Air operated damper	Centerline	32046-6
13.	Air operated damper	Powers	331-2792
14.	Pump motor	Reliance	100-HP-444T
15.	Pump motor	Reliance	30-HP-326T
16.	Instrument cable	Brand Rex	Low Capacitance Cable
17.	Cable	Kerite	5KV Power Cable
18.	Cable	Okonite	600V Power Cable
19.	Instrument cable	Raychem	---
20.	Instrument cable	Rockbestos	Coax/Triax
21.	TC wire	Rockbestos	---
22.	Control and instrument cable	Rockbestos	300/600V
23.	Switchboard wire	GE	Vulkene Supreme
24.	Transmitter	Rosemount	1152 Series
25.	Transmitter	Rosemount	1153GB Series
26.	Temperature element	Rosemount	88-149-1
27.	Temperature element	Rosemount	88-149-2
28.	Temperature element	Rosemount	89-138-2/88-14-3

Table 3.3 (continued)

	Equipment	Manufacturer	Model no.
29.	Temperature element (outside drywell)	Rosemount	89-86-4/88-14-1
30.	Temperature element	Rosemount	89-86-4/88-14-3
31	Position switch	Namco	EA180
32.	Fan motor	Westinghouse	143TCZ
33.	Fan motor	Westinghouse	286T
34.	Fan motor	Westinghouse	326T
35.	Fan motor	Westinghouse	364T
36.	Fan motor	Westinghouse	405TCZ
37.	Fan motor	Westinghouse	7.5HP/245T
38.	Blower motor	Reliance	324T
39.	Heater	GE	47D518673
40.	Panel	Comsip	K-IV
41.	Panel	Systems Control	120VAC Distr.
42.	Electrical penetration (2 items)	Conax	Low Voltage Power
43.	Lugs and splices	Amp	52900-53900
44.	Tape	Kerite	S-5MT-NUC
45.	Motor generator	Louis Allis	COGSF
46.	Transformer	Magnetics	L-12514
47.	Solenoid valve	Valcor	V105-205, 305; V526-5295-61, 62, 63; V526-5683-26, 27, 28, 29, 30, 32
48.	Fan motor	Westinghouse	143T
49.	Fan motor	Westinghouse	213T
50.	Electrical Penetration	GE	100 Series, MV

[1]Limitorque valve actuators are purchased to various specifications, contain various motors, and reference several test reports. Qualification evaluation considered these factors and the location in the plant.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.4 Component and Subsystem Design

5.4.2 Residual Heat Removal System

The residual heat removal (RHR) system was reviewed in accordance with SRP 5.4.7. The RHR system design has been compared with the functional, isolation, pressure relief, pump protection, and test requirements of Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," and found to comply with the implementation criteria for Shoreham.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

Subsequent to the issuance of SSER 2 Professor Bienkowski, a staff consultant, prepared a draft report on the results of his analyses of the containment load specifications for the chugging phenomenon in Mark II containments. His findings indicated that because of the random selection process for the individual vent chug initiation times the previously established load specifications for Mark II containments (Shoreham has a Mark II containment) may not be sufficiently conservative.

To show that the existing chugging load specifications are still adequately conservative, the Mark II owners used a two-step approach: First, they showed that containment response to the asymmetric chugging load specification was not significantly different from that for the symmetric specification. In fact, the comparison showed them to be remarkably similar. Having established the similarity, they applied the symmetric specification to the JAERI facility with 20 different sets of start times and showed that the calculated wall pressures were for the most part greater than the pressures recorded during some of the biggest chugs in the JAERI (Japanese Atomic Energy Research Institute) facility.

The staff and its consultant concluded that the Mark II Owners Group approach toward resolution of the chug start-time concern was a sound one. Also, there was general accord with the arguments presented, and the staff concluded that this is no longer an outstanding issue and that no modification to the load specifications (generic and plant unique) is required. Appendix A documents the work done by Professor Biewkowski on the effects of design desynchronization on the Mark II chugging load specifications and the work done by the Mark II owners to alleviate Professor Bienkowski's concern.

6.2.1 Containment Functional Design

Another set of concerns was raised after SSER 2 was issued. In a May 1982 meeting with the NRC and others relating primarily to the Grand Gulf Mark II containment, concerns were raised by a former General Electric (GE) employee who was involved in the detailed design of the standard Mark III containment design known as the STRIDE package.

Based on the results of this meeting, the staff has made a preliminary finding that no significant design deficiencies associated with these concerns have been uncovered. Therefore, the issuance of a low-power license for Grand Gulf was not withheld. The concerns, however, have raised certain questions which must be addressed prior to the issuance of a full-power license for the Grand Gulf and prior to final resolution of all the issues. The staff's review of these issues is continuing.

Although the concerns raised were specifically directed to the Mark III STRIDE design, the staff has evaluated the applicability of these concerns to the Mark I and Mark II containments. The preliminary review indicates that several concerns could be applicable to all boiling-water-reactor (BWR) pressure-suppression containments. Therefore, the staff will request all BWR Mark I and Mark II owners to address the issues that are applicable to their designs.

For the interim, however, it is the staff's judgment that these concerns need not delay the licensing schedule for the Shoreham plant. The basis for this judgment is as follows:

(1) Based on its review of the issues and the MP&L (Grand Gulf) response, the staff has concluded that the technical issues identified were for the most part considered in the design of the Grand Gulf containment; to date the staff has not uncovered any deficiency in the containment design.

(2) Design differences between Mark II and Mark III containment make many of the issues not pertinent to Mark II containments.

(3) The staff was informed by the Shoreham applicant, during a telephone conversation on June 24, 1982, that the applicant has completed a preliminary evaluation of the concerns. Based on this initial assessment of these concerns, the applicant has not identified any design deficiency. The applicant also stated that the many conservatisms employed in the design of the Shoreham containment have not been eroded.

In addition, the applicant has committed to submit plans for final resolution of these concerns to demonstrate the adequacy of the Shoreham containment design. Based on the above, the staff concludes that the applicant's approach to resolving these concerns is acceptable and that licensing of the Shoreham station may proceed as scheduled.

6.2.1.7 Steam Bypass of the Suppression Pool

SER Supplement No. 2 states that the resolution of the acceptance criterion for the low-pressure steam-bypass test must be completed before the fuel load date. In a letter dated April 23, 1982 (SNRC-693), the applicant committed to use an acceptance criterion for the low-pressure test of 3% of the calculated $A/\sqrt{K}$ for plant capability ($A/\sqrt{K}$ = 0.16 ft^2). Because this value (0.0048 ft^2) is more limiting than the staff's value (0.005 ft^2, which is equal to 10% of 0.05 ft^2, the generic plant capability calculated by the staff), the staff finds the applicant's commitment acceptable. The staff considers this item resolved.

6.2.3 Containment Isolation System

As part of the review of the containment isolation arrangement for the Shoreham station, the instrument lines were reviewed against the provisions of RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

Since the publication of the staff's review of the containment isolation arrangment in SER Supplement No. 1, the applicant has submitted Amendment 44 to its

License Application; the Amendment contained changes in the containment isolation arrangement design. Specifically, Amendment 44 stated that these are nonsafety-related instrument lines penetrating primary containment that rely on an orifice and a manual valve in each line to comply with the isolation provisions of RG 1.11. The backfitting considerations in RG 1.11 that apply to the Shoreham station include the following:

(1) Each instrument line connected to the reactor coolant pressure boundary and penetrating containment should be sized, or should include an orifice, such that if a postulated failure of the piping or of any component (including the postulated rupture of any valve body) in the line outside primary reactor containment occurs during normal reactor operation,

  (a) the leakage is reduced to the maximum extent practical consistent with other safety requirements;

  (b) the rate and extent of coolant loss are within the capability of the reactor coolant makeup system;

  (c) the integrity and functional performance of secondary containment, if provided, and associated safety systems (e.g., filters, standby gas treatment system) will be maintained; and

  (d) the potential offsite exposure will be substantially below the guidelines of 10 CFR 100.

(2) For each instrument line penetrating containment, including those connected to the containment atmosphere, some method of verifying during operation the status (open or closed) of each isolation valve should be provided.

After reviewing the information provided in the Shoreham FSAR on this issue, the staff concludes that the above considerations have not been adequately addressed for the Shoreham station.

The staff will report on the resolution of this new issue in a future supplement to the SER.

Subsequent to the issuance of SSER 2 the staff also noted that the control rod drive (CRD) insert and withdrawal lines depart from the explicit requirements of the General Design Criteria (GDC) and, as discussed below, are found to be acceptable on other defined bases. Both the CRD insert and withdrawal lines are provided with normally closed, fail-closed, solenoid-operated directional control valves, which open only during routine movement of their associated control rod. The normally closed, fail-open, air-operated scram inlet and exhaust valves open only when they are required to effect a rapid reactor shutdown (scram). In addition, manual shutoff valves are provided for positive isolation in the unlikely event of a pipe break within a hydraulic control unit. (These units and the valves described above are located outside containment to satisfy testing, inspection, and maintenance requirements.) In addition, each CRD insert line is provided with an automatically actuated ball check valve inside containment. The staff finds that the system design represents a departure from the explicit requirements of the GDC. However, in accordance with the provisions of GDC 55, which permits departure from its explicit requirement, the staff finds and notes for the record that the CRD containment isolation provision described above is acceptable on the basis stated in NUREG-0803, "Safety Evaluation Report Regarding Integrity of BWR Scram Systems," dated August 1981.

6.2.3.1 Containment Purge System

In SER Supplement No. 1, the staff reported that the applicant had committed to provide a debris screen in the vent line to ensure that isolation valve closure will not be prevented by debris. Since the issuance of this supplement, the staff has received, and found acceptable, the design details of the debris screen to be used in the vent line. The staff considers this item resolved.

7 INSTRUMENTATION AND CONTROLS

7.4 Systems Required for Safe Shutdown

7.4.3 Remote Shutdown System

The staff has now completed its review of the remote shutdown panel. GDC 19 requires in part that the ability be provided for the safe shutdown of the plant in case the main control room becomes uninhabitable. Plant design should provide for control stations in locations removed from the main control room. These stations are to be used for manual control and alignment operations needed to achieve and maintain a hot shutdown and subsequently to be able to achieve a cold shutdown. The applicant has provided a remote shutdown panel located within an enclosure in the reactor building. Except for reactor scram, which can be initiated from other remote locations, this panel allows the operator to bring the reactor to the cold shutdown condition in an orderly fashion and includes all instrumentation and controls required for operating the needed systems.

The following systems can be operated and monitored from this remote shutdown panel:

(1) Reactor core isolation cooling (RCIC)--turbine and valves

(2) SRVs--three solenoids

(3) RHR--pump, valves, and flow indication

(4) Reactor building service water (RBSW)--pump, valves, and discharge pressure

(5) Reactor building closed-loop cooling water (RBCLCW)--pump and inlet valves

(6) Fuel pool cooling system B--pump

(7) Miscellaneous recorders for reactor pressure vessel (RPV) level, RPV pressure, drywell pressure, drywell temperature, suppression pool level, and suppression pool temperature.

The remote shutdown capability is designed to control the required shutdown systems (one division of equipment) from outside the control room, irrespective of shorts, opens, or grounds in the control circuits in the control room that may have resulted from an event causing an evacuation. The functions needed for remote shutdown control (one division of equipment) are provided with manual transfer switches that override controls from the control room and transfer the controls to the remote shutdown panel.

When transferring control to the remote shutdown panel (RSP), controls for some functions are transferred to maintained contact switches. Assuming an orderly transfer of control to the RSP, the operator would have time to mimic the system's alignment of the main control panel before transferring control to the RSP. In a situation where realignment of the RSP prior to the transfer of control is not possible, Station Procedure 23.133.01 states the proper position for each switch during normal plant operation. These switch positions will be such that, when control is transferred to the RSP, system equipment will go to an alignment that will ensure no damage to either the system or the equipment. The system will then be at a predetermined alignment from which the operator can continue an orderly shutdown. The specific methods employed in these operations are described in Station Procedure 29.022.01, Shutdown from Outside Control Room Emergency Procedure, and in related station procedures.

The Shoreham design provides only for the transfer of one train of equipment to the remote shutdown panel. It appeared to the staff that, given a failure in this train of equipment, sufficient instrumentation and controls would not be available to attain a shutdown condition from outside the control room. It is the staff's position that to meet GDC 19, the remote shutdown system (RSS) design should provide redundant safety-grade capability to achieve and maintain hot shutdown and subsequently cold shutdown from a location or locations remote from the control room, assuming no fire damage to any required systems and equipment and assuming no accident has occurred. Credit may be taken for manual actuation (exclusive of continuous control) of systems from locations that are reasonably accessible from the remote shutdown panel. Credit may not be taken for manual actions involving jumpering, rewiring, or disconnecting circuits.

As a result of the staff's inquiries, the applicant has committed to provide and/or identify additional instrumentation and controls to meet the single-failure criterion prior to fuel load (except where noted below). The proposed additional instrumentation and controls are as follows:

(1) RCIC--Nothing additional; assume automatic operation of high pressure coolant injection to maintain RPV water level.

(2) SRVs--Provide controls for the Division II SRVs on a local panel in the relay room.

(3) RHR--Provide controls for the equivalent RHR train A pump and valves from the Division I emergency switchgear room and the reactor building secondary containment (RBSC) respectively. An RHR A flow indicator will be provided on a local panel. This flow indication will not be added until the first refueling.

(4) RBSW--Provide controls for the equivalent train A pump from the emergency switchgear room and controls for valves from the screenwell pump house and RBSC. RBSW train A flow indication will be provided in a local panel by first refueling.

(5) RBCLCW system--Nothing additional; single failure of this system would prevent the use of the normal RHR flow path whenever fluid temperatures exceed 212°F. However, a circulation path using suppression pool water could be established through the RHR heat exchanger using the RHR B pump on the RSP in the low-pressure coolant injection mode. Flow would return

to the suppression pool from the RPV via the RSP-controlled SRVs. The RHR pump can operate in this mode without RBCLCW cooling.

(6) Spent fuel pool cooling--Provide pump controls for the A spent fuel pool cooling pump on a local panel by the time of the first refueling.

(7) Miscellaneous local indicators--Provide a Division 2 indicator for RPV level, a Division 2 indicator for RPV pressure (by the time of the first refueling), Division 1 indicator for suppression pool level, and Division 1 and 2 indicators for suppression pool temperature (by first refueling).

The staff has concluded that the modifications to the RSS are an acceptable method for implementing the staff's position for redundant safety-grade capability. As noted above, several items will not be implemented until the first refueling outage. This is acceptable to the staff because there is an extremely low probability that an event will cause an evacuation of the control room to occur, concurrent with a single failure in the primary shutdown path at the RSP during the first cycle of plant operation. In addition, the redundant systems will still be operable from remote locations because only the indication for certain parameters will not be available until the first refueling outage.

The applicant has indicated that several of the readouts and associated sensors and power supplies on the remote shutdown panel are not safety grade. The applicant has reviewed the design to determine whether these nonsafety-grade readouts are required to achieve shutdown and has committed to upgrade them accordingly. The following primary path readouts are not presently classified as safety grade:

(1) RHR B flow

(2) RPV level

(3) RPV pressure

(4) service water B header pressure

(5) suppression pool temperatures

(6) suppression pool level

(7) RCIC flow

(8) RCIC turbine speed

(9) SRV N_2 pressure

The applicant has stated that the above readouts are to be upgraded to Quality Assurance Category I prior to the conclusion of the first refueling outage. This delay in implementation is the result of the long leadtime associated with the procurement for the above instrumentation. At this time, the RSP and all of the equipment located on the panel will be seismically qualified and environmentally qualified for its normal operating environment. This is acceptable to the staff because of the low probability of a seismic event occurring simultaneously with an event causing evacuation of the control room during the first cycle of plant operation.

In summary, the staff has reviewed the applicant's latest RSP design and has concluded that it will meet the regulatory requirements specified in GDC 19 and the guidance as detailed in SRP 7.4.II and III. However, as a confirmatory item, the applicant must provide acceptable final operating procedures and Technical Specifications for the RSP with the assumption of the most limiting single failure in the equipment train controlled from the RSP. The verification should include simulated system operability from remote stations away from the RSP with the assumption of the most limiting single failure in the equipment train controlled from the RSP. The verification should include a test of all communications required to accomplish the shutdown.

13 CONDUCT OF OPERATIONS

13.1 Organization and Structure of Applicant

13.1.3 Plant Staff Organization

13.1.3.1 Operating Division

TRAINING

Section 13.1.3.1 of SER Supplement No. 1 states

> We will condition the operating license to require that the training supervisor be a LILCO employee before the license is issued and that the other contract people be replaced with permanent LILCO personnel within six months following issuance of the license.

In a letter to the NRC dated December 11, 1981, LILCO stated

> The training Coordinator position in NOSD Nuclear Training section will serve the function of LILCo's training supervisor. That responsibility will include the overall training coordination of all contract and management personnel associated with Shoreham. The position will be filled prior to fuel load, thus placing LILCo in compliance with this part of the SSER license condition. The second aspect of this license condition calls for replacement of the contract training instructors within six months following issuance of the license. LILCo does not commit to replacing these individuals, but does agree to increase the current number of training instructors with qualified LILCo training personnel. LILCo believes this approach is prudent and conservative in that the training staff will be directly increased by the addition of the LILCo training instructors to the existing complement. This program will be implemented by the required date.

In response to an inquiry on the status of efforts to hire permanent LILCO training personnel, LILCO informed the staff in late May 1982 that the Training Coordinator position had been filled by a LILCO employee who was currently in training. LILCO also reported that there were on board three other instructors who are LILCO employees and six instructors who are not LILCO employees.

The staff concludes that (1) LILCO's December 11, 1981 commitment that "The Training Coordinator Position in NOSD Nuclear Training section will serve the function of LILCO's training supervisor" and (2) the information, as discussed above, that this position has been filled, meet the first part of the staff's position as stated above and in SER Supplement No. 1. The staff finds that LILCO's December 11, 1981 commitment "to increase the current number of training instructors with qualified LILCO training personnel" and to do this "by the required date," as well as the action that LILCO has already taken to accomplish this, indicate that LILCO will meet the intent of the second part of the staff's position as stated above. The staff concludes, therefore, that the applicant is committed to providing an inhouse training capability, and the staff will not require that conditions regarding the training as stated in Section 13.1.3.1 of SER Supplement No. 1 be included in the operating license.

13.2 Training

SER Supplement No. 1 stated that the staff had not completed its review of the training program for nonlicensed personnel. The applicant submitted an extensively revised version of FSAR Section 13.2 (Training) in a letter to NRC dated January 14, 1982, and submitted additional revisions to FSAR Section 13.2.1.2 in a letter to the NRC dated June 21, 1982. In both instances, the applicant stated that "the revised pages are not intended to represent a complete FSAR rewrite of these sections but merely to document the changes that have been made in response to Open Item No. 52 specifically" or to "clarify points arising from discussions of SER Item 52, Management Review." (See Section 1.7 of this report for the numbered list of open items.)

The licensee also stated: "The complete scope of the changes to the FSAR sections will be forwarded by a formal FSAR amendment in July." The evaluation reported here is based on the staff review of the revised description of FSAR Section 13.2.1.2 (Training Programs for Non-Licensed Personnel) as submitted in these two letters.

The staff reviewed this revised description of the training program for nonlicensed personnel against the acceptance criteria of Section 13.2 of NUREG-75/087 (SRP) Revision 1.

Training Responsibility

Each LILCO department that performs or provides support for safety-related activities in support of Shoreham plant operation is responsible to provide training for its personnel in accordance with the support function performed.

The overall coordination and evaluation of the LILCO corporate nuclear training program is the responsibility of the Manager, Nuclear Operations Support Department. Direct responsibility for coordination of the corporate nuclear training program and monitoring its effectiveness is delegated to the Nuclear Training Coordinator under the direction of the Nuclear Services Supervisor.

As discussed in Supplement No. 1 to the SER, the Shoreham Plant Manager has the overall responsibility for the conduct and administration of the plant training program while the day-to-day administration of the plant training program is carried out by the Station Training Supervisor.

General Employee Training

The General Employee Training Program includes the following topics:

(1) General description of the station and facilities

(2) Station security program and procedures

(3) Station fire protection program, including evacuation routes and signals, fire and fire hazard reporting, and basic fire fighting equipment

(4) Radiological health and safety including applicable portions of 10 CFR 19 and 20.

(5) Quality assurance

(6) Industrial health and safety

(7) Station emergency plan and implementing procedures

All LILCO and contractor personnel requiring unescorted access to all station areas participate in all topics of the General Employee Training Program. LILCO and contracted employees whose work assignments involve admittance only to station administration buildings participate in the program to the extent necessary to ensure safe execution of their duties. Personnel who receive training in topics 3, 4, and 7 above receive retraining instruction and are examined on these topics annually.

Training for Nonlicensed Managers, Engineers, and Technicians

All Shoreham managers, engineers, and technicians assigned to nonlicensed positions are trained in accordance with ANSI N.18.1-1971. Initially nonlicensed staff members participate in a basic nuclear course as an introduction to specific discipline programs. In addition, Reactor Engineering, Instrumentation and Control, Radiochemistry and Health Physics, and Maintenance supervisors, engineers, technicians, and mechanics receive training in their specialty areas and in plant systems. Examples of some of these specialty courses are:

(1) GE nuclear instrumentation course

(2) vendor-supplied specialized equipment training

(3) plant systems related to specialty area

(4) station procedures associated with specialty area

(5) SNP familiarization training

(6) GE BWR chemistry course

(7) GE health physics technology training course

(8) GE station nuclear engineering course

(9) Computer-user programming course

(10) Maintenance skills training course

Support Engineering personnel participate in plant systems familiarization training following their assignment to a particular support job. They are trained to a degree commensurate with their individual needs with respect to completing particular job assignments.

Requalification training for Instrumentation and Control, Radiochemistry and Health Physics, Reactor Engineering, Maintenance, and Support Engineering personnel consists of (1) self study of new and revised procedures, descriptions of plant modifications and event and operating experience reports related to each of their specialty areas with supervisory review and (2) the refresher training associated with the general employee training program. In addition, maintenance personnel requalification training includes periodic supervisory reviews and discussions of scheduled and completed maintenance courses, rehearsal of scheduled maintenance, and review of completed maintenance.

STA Training

Personnel assigned as Shift Technical Advisors (STAs) participate in an integrated program of theoretical and practical instruction. The components of this program are the following:

(1) basic theory

(2) plant-specific theoretical training

(3) plant systems/procedures training

(4) plant accident/transient analysis training

(5) mitigating core damage training

(6) management/supervisor skills training

An STA Requalification Training Program, 2 weeks long, is administered annually.

Mitigating Core Damage Training

A Mitigating Core Damage Training Program is provided for STAs and operating personnel, from the Plant Manager through the operations chain, to licensed operators. The program provides instruction related to degraded core recognition and methods for recovery from the degraded core condition. Managers and technicians in the Instrumentation and Control, Health Physics, and Radiochemistry sections will participate in the program to a degree commensurate with their responsibilities. The program includes a course which is 1-week long and includes the following components:

(1) Core cooling mechanics/accident recognition. Topics included are: adequate core cooling, heat sources, core cooling mechanisms, and inadequate core cooling recognition.

(2) Core damage mitigation. Topics included are: fixed/movable nuclear instrument use, degraded core effects on coolant chemistry, process instrument response, corrosion effects, gas generation sources, and accident environment dose determination.

(3) Core transient identification and damage mitigation through use of emergency procedure guidelines.

Fire Brigade Training

Personnel who make up the Station Fire Brigade will be trained via a program of classroom instruction, practice sessions, and drills prior to their official assignment to the Fire Brigade. The training will be provided by individuals who are knowledgeable and experienced in fighting the types of fires that could occur in the plant and in using the types of equipment available at Shoreham. The classroom phase of the training will include

- Fire hazard identification by location and fire type, including locations where breathing apparatus is required.
- Familiarization with plant layout, including routes for ingress and egress as well as locations of all fixed and portable fire fighting equipment.
- Methods and equipment appropriate to each type of fire.
- Indoctrination in the Plant Fire Protection Program. Each Fire Brigade member will be trained to handle any of the positions of responsibility under the Fire Brigade Chief.
- Proper use of respiratory protection, communication, lighting, and portable ventilation equipment.
- Review of the plant Pre-Fire Plans, which identify the preferred fire fighting equipment to be used in each specifically identified fire hazard area. This review also includes review of proper equipment and procedures to be used for the balance of the site.
- Review of pertinent modifications, additions, or changes to the Plant Fire Protection Plan or fire fighting equipment.
- Methods for fighting fires in buildings or tunnels.
- Toxic and corrosive characteristics of expected combustion products.
- Station evacuation signals and routes.

Candidates for the position of Fire Brigade Chief will receive Fire Brigade Training and Fire Brigade Leadership Training designed to teach direction and coordination of fire fighting activities.

All station Fire Brigade personnel will be provided with at least four sessions of refresher instruction each year. These will be scheduled such that the topics will be repeated at least every 2 years with appropriate updates in the detailed course material.

Practice sessions will be held at the Suffolk County Fire Training Center and will allow Fire Brigade members to train on actual fires. Each Fire Brigade member will attend at least one practice session per year and will don protective equipment (including respiratory protection) at least once a year.

Preplanned drills will be performed at least once every 3 months for each Fire Brigade, with each Fire Brigade member attending a minimum of two drills per year. At least one of these drills per year, per Fire Brigade, will be unannounced. At least one of these drills per year, per Fire Brigade, will be on a backshift. At least one drill per year will involve the participation of the Wading River Fire Department onsite

The drills will conform to the established plant fire fighting plans where possible and will include operating fire fighting equipment, where practical, and onsite operation of self-contained breathing apparatus, communication equipment, and portable and/or installed ventilation equipment. The drills will be critiqued to assess fire alarm effectiveness, response time and equipment selection, placement, and usage, as well as the leader's direction of the effort and each member's response. Unsatisfactory drills will be repeated within 30 days.

Fire Protection Staff Training

Personnel responsible for the implementation of the Station Fire Protection Program will recieve Fire Brigade Training and Fire Protection Technology Training. Fire Protection Technology Training will include the following topics:

(1) Station building layout and fire protection system design

(2) Design of and maintenance on fire detection, suppression, and extinguishing equipment

(3) Fire prevention techniques and procedures

Summary and Conclusions

The applicant has described his program for training nonlicensed personnel and a schedule for that training as related to the applicant's fuel load date. The applicant's program for training and retraining of nonlicensed personnel meets the requirements of ANSI N18.1 - 1971 as endorsed by Regulatory Guide 1.8; Fire Brigage personnel will undergo classroom instruction, fire-fighting practice and periodic fire drills; STAs will receive university-level training in the basic fundamentals of nuclear and reactor enginnering, radiation protection, thermodynamics, and fluid mechanics and will also receive training in plant systems, reactor operations, and transient and accident response; and STAs and other operating personnel will receive training in the mitigation of core damage.

The staff has reviewed these training plans and found that the training for nonlicensed plant staff personnel meets the acceptance criteria of SRP 13.2 (NUREG-75/087) Revision 1, and the acceptance criteria for STA training and core damage mitigation training of SRP 13.2.2 (NUREG-0800). On this basis, the staff concludes that the training for nonlicensed Shoreham station staff personnel is acceptable.

13.3 Emergency Preparedness Evaluation

SER Supplement No. 1 provides the staff's evaluation of the applicant's emergency plans for Shoreham Nuclear Power Station. In SSER 1, the staff specifically identified deficiencies requiring revisions or additional information. The applicant has provided the staff with the required additional or revised information. With a transmittal letter dated June 28, 1982, the applicant submitted a revised Emergency Plan and copies of the Training Manual and of the Emergency Plan Implementing Procedures. With a transmittal letter dated September 1, 1982, the applicant submitted additional revised pages of the Shoreham Emergency Plan; these were an evacuation time estimate document, copies of emergency preparedness information for dissemination to the public, and a dose calculation manual. The submitted material has been reviewed and evaluated by the NRC staff, which also made site visits to verify some information.

The applicant's responses--that, except for offsite planning, have resolved most of the deficiencies previously identified by the staff as requiring revision or additional information--are discussed below. The order of presentation corresponds to the listing of deficiencies that appears in Section 13.3 of the SER.

The staff's conclusions are provided in Section 13.3.17 of this supplement.

13.3.1 Assignment of Responsibility (Organizational Control)

13.3.1.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Provide agreements with the two state agencies (New York Department of Health and the Office of Disaster Preparedness) that have a primary response role in the event of a serious emergency.

(2) Update the agreement with Suffolk County to reflect the current guidance in NUREG-0654, particularly in the areas of emergency classification, means of notification, notification times, information requirements, and communication drills. Also because the staff has not yet received the county plans which presumably describe the concept of operations within the plume exposure pathway emergency planning zone (EPZ), provide agreements with any other agencies or support organizations that may have a primary response role within the EPZ.

(3) All agreements should be reviewed and certified current, and preferably dated within 1 year of anticipated license issuance.

Applicant's Response

(1) The New York State Site-Specific Emergency Plan for Shoreham is still under development and is not available for NRC review. This remains an open item.

(2) The Suffolk County Radiological Emergency Response Plan is still under development and is not available for NRC review. This remains an open item.

(3) Agreements by the applicant with Radiation Management Corporation, Central Suffolk Hospital, Wading River Fire District, and Old Mill Inn (for the News Center) have been verified to be current by NRC auditors. The applicant's revised plan includes letters of agreement with the Island Broadcasting Company (WALK); U.S. Department of Energy, at Brookhaven National Laboratory; and the Stone and Webster Engineering Corporation.

The applicant has committed to update or certify as correct all agreements approximately 3 months prior to fuel load.

Conclusion

The staff finds that the applicant has provided an acceptable response on agreements with offsite agencies, with the exception that New York State and Suffolk County Emergency Plans remain open items.

13.3.2 Onsite Emergency Organization

13.3.2.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) To assess the relationship between the emergency organization and the responsibilities and duties of the onsite staff complement, provide a brief description and diagram of the normal operating organization.

(2) Upon the arrival of the Recovery Manager, it is not clear as to whether the Recovery Manager or the Emergency Director has the overall responsibility for the direction and control of the integrated emergency response efforts. Also the transfer of the emergency response functions from the control room to the Technical Support Center (TSC) and the Emergency Operations Facility (EOF) should be in accordance with that specified in Table 1 of NUREG-0696.

(3) The minimum staffing requirements are deficient with respect to the guidance set forth in Table B-1 of NUREG-0654:

- The onshift complement should include individuals having the necessary expertise in the areas of mechanical and electrical maintenance.

- The onshift complement should include two individual capable of performing health physics technician duties for radiation protection activities.

- The augmentation of the shift team should include 11 individuals within 30 minutes as indicated in Table B-1 of NUREG-0654.

- The overall augmentation of the shift team within 60 minutes is deficient by one individual in each of the following: electrical/mechanical expert for technical support; rad waste operator for repair and corrective actions; and a health physics technician for radiation protection tasks.

(4) Figure 3-1 of the plan should separate the Operational Support Center (OSC), TSC, and EOF to more clearly indicate the interfaces between the onsite facilities and the offsite centers and organizations. It also appears that the figure is deficient with respect to the notification interaction that would involve the state and county warning points as well as the U.S. Coast Guard.

(5) The various support services to be provided by local agencies (police) and the private sector (General Electric, Stone & Webster, Institute of Nuclear Power Operations, and Nuclear Safety Analysis Center) should be supported by written agreements and appended to the plan. These agreements should contain the information set forth in criterion B.9 of NUREG-0654

Applicant's Response

(1) The applicant has described the normal operating shift of 10 individuals and security personnel, and has diagrammed the complete operating organization in Figure 5-1 of the plan. The normal onshift organization is diagrammed in Figure 5-2 and the onshift emergency response organization in Figure 5-3 of the plan.

(2) The revised plan states that when an Alert Emergency organization is functioning, the Plant Manager will report to the activated TSC and assume the Emergency Director position from the Watch Engineer. During either a Site or General Emergency, the Response Manager will report to the EOF and assume responsibility for overall direction and control of the response, and the Plant Manager reports to the Response Manager.

(3) The applicant will have a Maintenance Section consisting of a permanent staff of 7 utility workers, 22 mechanics, 3 foremen, 3 engineers, and 2 maintenance coordinators. In addition, the applicant has a floating maintenance force of about 250 individuals, 20 of whom will be assigned to Shoreham. Mechanical and electrical maintenance on the back shifts will be accomplished by shift personnel normally assigned other functions.

The onshift complement now includes a Health Physics technician and a Radiochemistry Technician who is capable of performing Health Physics Technician duties.

The applicant has conducted a survey of normal one-way travel time from home to work, and, on the basis of this survey, now states that shift staff augmentation can meet the 30-minute response described in Table B-1 of NUREG-0654.

On the basis of that travel time survey, the applicant now states that overall augmentation of the shift team within 60 minutes will include, as a minimum, all of the required personnel to perform the functions listed in Table B-1 of NUREG-0654.

(4) Figures 3-1.1 (Control Room), Figure 3-1.2 (Technical Support Center), and Figure 3-1.3 (Emergency Operations Facility) of the plan indicate the interfaces between licensee facilities and offsite centers and organizations, including the U.S. Coast Guard and state and county emergency facilities.

(5) Appendix B includes current letters from the Department of Energy's Brookhaven Area Office, GE Nuclear Power Systems Division, Radio Station WALK, and Stone and Webster Engineering Corporation describing available support services.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.3 Emergency Response Support and Resources

13.3.3.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Although the plan indicates that Section 5.3.1.1 discusses the provisions made for incorporating the Federal response capability into the Shoreham emergency plan, that section only identifies the individual authorized to request Federal assistance. Expand the discussion of emergency response support and resources to include the information required in criterion C.1.b and c of NUREG-0654.

(2) Identify the four radiological laboratories that could be utilized during emergencies as mentioned in the plan, including their location and general capabilities.

Applicant's Response

(1) The Federal Radiological Monitoring and Assessment Plan (FRMAP) for the Northeast region of the U.S. is located at Brookhaven National Laboratory, within 7 miles of Shoreham. NRC auditors have verified with the Director of the Brookhaven FRMAP team that the resources of the laboratory would be available in an emergency, if needed.

(2) Radiological laboratories that could be utilized during an emergency include Radiation Management Corporation in Philadelphia, Pa.; Public Service Electric and Gas in Maplewood, N.J.; NUS Corporation in Rockville, Md. and Pittsburgh, Pa., and Teledyne in Westwood, N.J. In addition, Brookhaven National Laboratory offers a complete research facility. These facilities have the capability to perform isotopic and radiochemical analyses.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.4 Emergency Classification System

13.3.4.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Establish Emergency Action Levels (EALs) for each initiating condition specified in Appendix 1 to NUREG-0654. The EALs should be observables (e.g., instrument readings, equipment status indications, alarm annunciators) that are both necessary and sufficient to explicitly and uniquely characterize each initiating condition. It is recommended that the format be in tabular form for each of the four emergency classes that list the initiating conditions and specifies the EALs for each condition.

(2) The initiating conditions should include the postulated accidents in the FSAR in addition to those referenced in item (1) above.

Applicant's Response

(1) The Emergency Plan now includes EALs for each initiating condition in Appendix 1 of NUREG-0654. The EALs are defined as observable and measureable indications such as instrument readings or equipment status indications. There are some blanks in the EALs relating to specific instrumentation readings. The applicant has committed to supply all missing information prior to fuel load. For the initiating condition that requires a combination of parameters, a logic diagram is provided to assist the operator in understanding the statements.

(2) The Emergency Plan now includes a list of the postulated accidents analyzed in FSAR Chapter 15 with a correlation for each accident to one of the four emergency classes.

Conclusion

Based on the applicant's commitment to supply all missing information in the EALs prior to fuel load, the staff finds that the applicant has provided an acceptable response to this item.

13.3.5 Notification Methods and Procedures

13.3.5.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Describe the mutually agreeable bases for notification of the response organizations consistent with the emergency classification action level scheme set forth in Appendix 1 to NUREG-0654. The use of predetermined EALs in establishing an emergency class should minimize the need for subjective judgment at the time of an actual emergency. This is not reflected in Section 6.2 and Appendix B of the plan.

(2) Describe the provisions for initial and followup emergency meassages to offsite authorities, and provide sample formats that include the information specified in criteria E.3 and 4 of NUREG-0654.

(3) Provide a complete description of the administrative and physical means for prompt alerting and notification of the public within the plume exposure pathway EPZ. Sufficient detail should be provided for evaluation against the criteria set forth in Appendix 3 to NUREG-0654. Include a schedule through operational readiness for the overall system.

(4) Provide copies of the written messages intended for release to the public in the event of a serious emergency. If these are still being developed, provide a schedule for their completion.

Applicant's Response

(1) The applicant will make a protective action recommendation to Suffolk County and New York State authorities based upon the EAS scheme and emergency classification system referenced in 13.3.4.3(1). The protective actions available will be detailed in New York State and Suffolk County emergency response plans. The applicant plans to coordinate planning in this area with local and state emergency planning personnel when the offsite plans are available for review.

(2) Examples of notification fact sheets for initial and followup messages are in Appendix F of the plan, and they include the information specified in criteria E.3 and 4 of NUREG-0654.

(3) The applicant has submitted a complete description of prompt alerting system in the document entitled "Final Design of Prompt Notification System for Shoreham Nuclear Power Station," April 1982. The system consists of outdoor sirens and tone-activated radios for special facilities such as schools and nursing homes. The siren system has been installed and is under test at the time of this report.

(4) Appendix F of the Emergency Plan now contains prepared sample news releases and messages to the general public for the four classes of emergencies.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.6 Emergency Communications

13.3.6.3 Prior Deficiencies

The following deficiencies were identified in SSER 1 and remain outstanding:

(1) Specify the organizational titles and alternates for both ends of the communication links that would be involved in initiating emergency response actions, and indicate that such stations will be staffed 24 hours per day.

(2) A diverse means of communication, such as a radio system, should be available between the site and the primary offsite response agency. It is not clear that such radio communications are available with the Suffolk County EOC.

(3) Identify the provisions for communications with DOE and the U.S. Coast Guard as set forth in criterion F.1.c of NUREG-0654.

(4) Communications from the control room and the TSC to hospitals, the Suffolk County Medical Communications Center, and the Wading River Fire Department

(for ambulance service) will be via commercial telephone. The Wading River Fire Department Ambulance is equipped with two-way radio communication to the Wading River Fire Department main station house via the normal, two-frequency County Fire Band Radio, and the Central Suffolk Hospital and the Suffolk County Medical Communications Center via the Medical Communications Radio Band.

The staff will document the resolution of these deficiencies in a future supplement.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.7 Public Education and Information

13.3.7.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Identify what advance arrangements have been or will be made for ensuring that a local motel will be available and have the capability for use as the Emergency News Center. Indicate the location of the motel(s) relative to the EOF.

(2) Provide a commitment to conduct the orientation program with the news media at least annually in accordance with the criterion G.5 of NUREG-0654.

(3) Provide a sample copy of the emergency preparedness information that will be provided for both the resident and transient population around the site.

Applicant's Response

(1) Appendix B of the Emergency Plan includes a letter of agreement with the Old Mill Inn, a hotel in Ronkonkama located about 18 miles from the reactor and about 4.5 miles from the EOF at Hauppage, for use as an Emergency News Center. It has a 3000-ft^2 press working area in addition to telephones, food service, and parking.

(2) The Emergency Plan now commits to an annual orientation program for the news media to acquaint them with the emergency plans, information concerning radiation, and points of contact for release of information in an emergency.

(3) The applicant has submitted draft copies of posters to be displayed at beaches, information for transients at motels, and a brochure for mailing to permanent residents in the Inhalation EPZ. This public information program will be coordinated with Suffolk County before it is finally implemented.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item. Final implementation will be coordinated with Suffolk County.

13.3.8 Emergency Facilities and Equipment

13.3.8.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Provide a commitment and schedule for the permanent Emergency Response Facilities in accordance with NUREG-0696.

(2) Upon declaring the appropriate emergency class to activate the Technical Support Center (TSC) and/or Emergency Operations Facility (EOF), specify the time required to achieve operational readiness at these response facilities.

(3) Provide a complete description of the meteorological measurement system for evaluation against the criteria set forth in Appendix 2 of NUREG-0696, and a schedule for meeting the milestones specified in Annex 1 to the appendix.

(4) Identify the onsite radiological and process monitoring system in accordance with criteria H.5.a and b of NUREG-0696. The monitors identified here should include those used for obtaining EALs for the appropriate initiating conditions listed in Appendix 1 to NUREG-0654. Include instrument identification, location, and range.

(5) Provide a description of the fire and combustion products detection system. Specific detector types and locations may be referenced if they are described in other documentation that is part of the fire protection program.

(6) Describe the provisions for obtaining offsite information regarding geophysical phenomena as specified in criterion H.6.a of NUREG-0654.

(7) Provide sufficient information to establish that the offsite dosimetry will meet the requirements of the NRC Radiological Assessment BTP for the Environmental Radiological Monitoring Program. It appears that sufficient dosimetry is not provided. Also, it is not clear from the discussions in Sections 6.1.1, 6.1.2, and 7.3.2 where the monitoring devices (i.e., thermoluminescent dosimeters (TLDs), air samples, radiation detectors) will be or if the figures indicate locations where field monitoring will be done using portable instruments.

(8) Indicate that the Operational Support Center (OSC) will have adequate capacity and supplies for the emergency personnel who potentially report to that area, and that the available equipment will include that specified in criterion H.9 of NUREG-0654.

Applicant's Response

(1) The onsite TSC is complete except for installation and testing of some instrumentation and communications equipment. It is located in the office and service building annex, about a 2-minute walk from the control room. The applicant has committed to install a shielded overpass to lessen the travel time from the control room to TSC, and to eliminate unshielded passage between buildings. The applicant has also committed that the TSC will be functional at fuel load.

The EOF is located in the LILCo Training Center near the intersection of the Long Island Expressway and Veterans Highway. It is complete except for installation and testing of some instrumentation and communications equipment. It will be functional before fuel load.

The OSC is located in the office and service building and will be functional before fuel load.

(2) On the basis of the survey of travel time from home or office to the Emergency Response Facilities, the applicant states that the TSC should achieve operational readiness within approximately 30 minutes and the EOF within approximately 60 minutes.

(3) The applicant has a 400-ft meterological tower about 1 mile west of the containment building that provides measurements of wind speed, direction, and stability class by temperature differential between two levels. There is a 33-ft backup tower on the site, northwest of the containment building, that provides wind speed, direction, and stability class by sigma theta measurement. Either system will be available to feed meterological information to the dose assessment computer. The system will be operational before fuel load.

(4) Table 6-1 of the Emergency Plan lists onsite radiological and process monitoring systems used in identifying the initiating conditions for EALs. With each monitor is listed the expected concentration of radioactivity in a vent system as the result of accident, the location, and the range of the monitor. Additional instrumentation, alarms, and annunciators used for accident assessment and classification are defined directly in the EALs.

(5) The NRC staff's evaluation of the applicant's fire detection and alarm system is in Section 9.5.2.4 of SER Supplement No. 1. The staff's conclusion is that the system meets the appropriate guidelines and is acceptable.

(6) The applicant states the shift operations personnel will obtain National Weather Service (NWS) information on severe weather warnings and watches in the site vicinity from either WSI or NWS Weather Radio. In addition, the applicant is negotiating with a consultant for offsite seismic information. Arrangements for such information will be made prior to fuel load.

(7) The Shoreham monitoring network now includes 36 locations equipped with TLDs as shown in Table 7-2 of the Emergency Plan.

(8) The Emergency Plan states that the OSC in the office and service building will have adequate capacity (for approximately 20 people) and supplies, including protective clothing, respiratory protection, portable lighting, portable radiation monitoring equipment, and a camera. The communications equipment shall include one dedicated telephone extension to the TSC, one dedicated telephone extension to the control room, and a dial telephone extension capable of reaching onsite and offsite locations.

Conclusion

Based on the applicant's commitments noted above to be fulfilled before fuel load, the staff finds that the applicant has provided an acceptable response to this item.

13.3.9 Accident Assessment

13.3.9.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) As part of the description regarding onsite capability and resources to provide initial and continuing assessment throughout the course of an accident, include post-accident sampling capability, radiation and effluent monitors, inplant iodine instrumentation, and containment monitoring. Provide sufficient detail to enable evaluation against clarification items II.B.3, II.F.1, and III.D.3.3 in NUREG-0737.

(2) Provide detailed information on the methods and techniques used for determining the source term of releases of radioactive material within plant systems. As an aid in assessing the extent of potential core damage, include plots that show the containment radiation monitor reading vs. time following an accident for incidents involving 100% release of coolant activity, 100% release of gap activity, 1% release of fuel inventory, and 10% release of fuel inventory.

(3) Provide the appropriate implementing procedures together with the Offsite Dose Calculation Manual to enable evaluation of the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions.

(4) Describe the methodology to be used for determining the release rate/projected dose if the instrumentation normally providing input to the radiation monitoring system (RMS) computer is off scale or inoperable.

(5) Provide estimates of the deployment time and the means of transportation to be used for personnel involved in the offsite assessment of radiological hazards through the liquid or gaseous release pathways.

(6) Describe the means for relating measured field contamination levels to dose rates for applicable isotopes listed in Table 3 of NUREG-0654.

Applicant's Response

(1) In Section 6.1 of the Emergency Plan, the applicant has described station instrumentation provided for accident assessment that includes post-accident sampling and analysis capability, radiation and effluent monitors, inplant iodine instrumentation, and containment monitors. The NRC staff review and acceptance of these systems is in Section 22 of SER Supplement No. 1.

(2) In the Emergency Plan and Implementing Procedures, the applicant has described the use of manual calculations and the use of the RMS computer for determining the impact of radioactive releases in the environs. The RMS computer evaluates inputs from radiation release monitors, computes airflow in vents, calculates χ/Q (relative concentration) values at downwind locations, and, finally, a dose rate at the location.

As an aid for operators to assess potential core damage, Figure 6.6 of the Emergency Plan shows drywell high-range monitor response as a function of time after reactor scram for the following releases into containment:

100% reactor coolant
100% gap activity
1% fuel damage
10% fuel damage
100% fuel damage

(3) The Offsite Dose Assessment Methodology for Emergency Applications was submitted as Attachment 3 of the applicant's latest submittal. The applicant has also submitted Emergency Plan Implementating Procedure (EPIP) SP.69.022.01, Determination of Offsite Dose, for evaluation of onsite and offsite doses.

(4) The manual calculation technique to be used for estimating downwind doses if the RMS computer is off scale or inoperative is described in EPIP SP.69.022.01, Determination of Offsite Doses.

(5) The Emergency Plan states that, at the Alert level, one survey team may be called in to better facilitate offsite surveys should the incident degrade to a more severe classification. For the Site and General Emergency, two additional survey teams will be capable of being dispatched within 60 minutes of such declaration. The plant Emergency Planning Coordinator has been assigned two dedicated vans for offsite surveys.

(6) Offsite teams will be equipped with postulated sampling equipment, airborne iodine sampling and measurement equipment, and portable beta/gamma dose rate meters. Airborne iodine samples will be analyzed in the field and the results of these analyses will be transmitted to the EOF/TSC by radio. Dose rate measurements will also be transmitted by radio. Particulate samples will be returned to the EOF/TSC for isotopic analyses.

Conclusion

Based on the applicant's commitment to complete installation of equipment for accident assessment before fuel loading, the staff finds that the applicant has provided an acceptable response to this item.

13.3.10 Protective Response

13.3.10.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Specify the time required to warn or advise persons who may be in the following owner-controlled areas: Wading River Creek marsh on the northeast portion of the site; shorefront and jetties along the north boundary of the site; and the summer camp on the Shoreham West property.

(2) Describe the provisions for evacuation, including routes and transportation for onsite individuals to an offsite location, in accordance with criterion J.2 of NUREG-0654.

(3) Describe the provisions for monitoring people evacuated from the site and the decontamination capabilities provided at or near the monitoring point.

(4) Indicate that the accountability for onsite personnel can be accomplished within 30 minutes from the start of an emergency.

(5) Describe the availability of respiratory protection equipment for onsite emergency personnel.

(6) Provide an explicit commitment for recommending protective action to the appropriate offsite authorities based on EALs corresponding to projected dose to the population-at-risk in accordance with Appendix 1 of NUREG-0654 and the EPA Manual of Protective Action Guides.

(7) Provide an evacuation time assessment study in accordance with the criteria set forth in Appendix 4 to NUREG-0654, as an appendix to the plan.

(8) Figure 6-2 in the plan is too small to permit identification of the primary and secondary evacuation routes. In addition, the evacuation areas, relocation centers, and shelter area are not identified. Provide an enlarged map or additional smaller maps to enable evaluation with respect to the information required by criterion J.10.a in NUREG-0654.

(9) Discuss the bases for the choice of recommended protective actions for the plume exposure pathway during emergency conditions including the expected local protection afforded by residential or other structures against direct and inhalation exposure. Also note that Figure 6-1 should not preclude the use of alternate protective measures such as sheltering.

Applicant's Response

(1) Upon assessment by the Emrgency Director that a situation exists that requires evacuation of areas of the plant, an evacuation signal will be activated simultaneously with an announcement of the emergency condition over the party page system indicating the areas to be evacuated.

Section 6.4.2 states that notification to members of the public who may be in a public access area within the site boundary would be accomplished via the Prompt Notification System (sirens). At the direction of the Emergency Director, notification to these areas shall be made by plant personnel by telephone, e.g., to the camp at Shoreham West, or by dispatch of a station employee with a power megaphone within 30 minutes of such a determination.

(2) Figure 6-1 of the Emergency Plan shows primary and alternate evacuation routes from the site. Transportation shall be by private vehicles.

(3) Section 6.5.2 of the Emergency Plan describes the decontamination facility, which is located adjacent to the health physics office on the 15-ft elevation of the turbine building where monitoring would be performed. Monitoring will also be performed as personnel leave the site via the portable monitors in the guard house. Vehicle monitoring and decontamination would be performed at the 69-kV substation along the LILCo main access road.

(4) The Emergency Plan states that accountability for onsite personnel will be accomplished within 30 minutes. Drills have established that the objective of accountability within 30 minutes has been accomplished.

(5) Self-contained breathing apparatus for emergency use is stored in each of the onsite emergency response facilities, the control room, the TSC, and the OSC. A facility for testing and fitting respiration and a refilling system will be installed on site, and procedures for implementation will be developed prior to fuel loading.

(6) Section 6.4.1 of the applicant's Emergency Plan states the LILCo's protective action recommendation to Suffolk County and New York State authorities will be based on plant conditions in accordance with Appendix 1 of NUREG-0654 and and upon EPA Protective Action Guidelines.

(7) The applicant has submitted Radiological Emergency Evacuation Plan for the Shoreham Nuclear Power Station, prepared by KLD Associates, Huntington Station, N.Y. dated August 27, 1982. The document complies with the criteria in Appendix 4 of NUREG-0654.

(8) Figure 6-2 of the Emergency Plan shown relocation centers. Maps in the Radiological Emergency Evacuation Plan permit identification of the primary and secondary evacuation routes.

(9) The Emergency Plan discusses the bases for the choice of protective actions in general terms. Emergency Implementing Procedure 69.026.01 (Protective Action Recommendations) provides detailed guidelines that take into consideration radioactive releases, meteorological condition, evacuation time estimates, and shielding factors for structures.

Conclusion

Based on the applicant's commitment to complete installation of facilities for protective response before fuel loading, the staff finds that the applicant has provided an acceptable response to this item.

13.3.11 Radiological Exposure Control

13.3.11.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Include the emergency exposure guidelines for individuals involved in assessment actions, first aid, personnel decontamination, ambulance service, and medical treatment.

(2) Designate the individual(s) by position or title who can authorize exposures in excess of 10 CFR 20 limits and is (are) readily availalbe for such decisions on a 24-hour-a-day basis.

(3) Describe the capability for providing 24-hour-a-day determination of radiation dose to emergency workers, distribution and reading of dosimeters, and maintenance of dose records.

(4) Specify action levels of determining the need for decontamination and the means for decontamination of personnel wounds, supplies, instruments, and equipment.

(5) Describe the onsite contamination control measures for limiting areas access, drinking water and food supplies, and the quantitative criteria for permitting return of areas and items to normal use.

(6) Describe the capability for decontaminating personnel evacuated to offsite locations including provisions for extra protective clothing and decontaminants suitable for the contamination expected with particular attention to radioiodine contamination of the skin.

Applicant's Response

(1) Table 6-4 of the Emergency Plan shows Emergency Exposure Criteria for life saving and for the goal of general reduction of public exposure. The goal in emergency operations is that 10 CFR 20 limits will be observed. Emergency Plan Implementing Procedure 69.050.01 (Radiation Dose During an Emergency) provides guidance for authorizing the exceedence of exposure limits during an emergency that are consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

(2) Authorization to exceed 10 CFR 20 limits shall be made only by the Emergency Director and/or the Radiation Protection Manager. This capability, during an emergency, is readily available on 24-hour-a-day basis.

(3) Normal Health Physics procedures for distribution and reading of dosimeters and maintenance of dose records will be in force during an emergency. In addition, the Health Physics Department will provide 24-hour-a-day service to read TLDs for emergency workers on the site.

(4) In an emergency, normal station contamination limits of 100 cpm above background shall be adhered to. Decontamination procedures are detailed in the health physics procedures. The decontamination facility adjacent to the

health physics office contains showers with controlled drains and the necessary materials for personnel decontamination.

(5) Emergency Plan Implementing Procedures 69.030.03 (Contamination Control During Emergencies) provides guidance to emergency organization staff persons to supplement the guidance contained in the health physics procedures. Food and water supplies shall be provided from areas outside the access control boundaries. Return to normal use will be authorized by the Emergency Director with concurrence of the Radiation Protection Manager using normal contamination limits in health physics procedures.

(6) Section 6.5.2 states that personnel evacuating from the site and still found to be contaminated will be issued protective clothing and directed to the EOF decontamination facility for further monitoring and decontamination. The same material and equipment utilized in onsite decontamination will be utilized at the EOF. Provisions will be available for radionuclide analysis of the personnel contamination to determine the amount of radioiodine present. Personnel contamination that cannot be removed by normal health physics procedures will be referred to a medical specialist in personnel radiation accidents.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.12 Medical and Public Health Support

13.3.12.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Discuss the arrangements for backup hospital and medical service in addition to the Central Suffolk Hospital.

(2) Discuss the arrangements for transportation of contaminated victims to the offsite medical support facilities.

Applicant's Response

(1) Section 6.5.3 of the Emergency Plan states that arrangements for backup hospitals have been made with University Hospital in Philadelphia, Pa., though a contract with Radiation Management Corporation.

(2) Transportation for minor injuries will be accomplished by LILCO and/or privately owned vehicles. Major injuries shall be transported to Central Suffolk Hospital by ambulance service provided by the Wading River Fire Department. Transportation to the University Hospital in Philadelphia shall be by helicopter provided by Radiation Management Corporation.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.14 Exercises and Drills

13.3.14.3 Prior Deficiencies

The plan requires revision and/or additional information as follows:

(1) Communications with the NRC shall be tested monthly in accordance with paragraph E.O.d of Appendix E to 10 CFR 50.

(2) Communications with states within the ingestion pathway should be tested quarterly.

(3) Include the U.S. Coast Guard in the communications drill with the Federal response organizations.

(4) Discuss the mix of structures and less structured aspects of the program that will allow free play for decision-making during drills and exercises.

Applicant's Response

(1) Section 8.1.2 of the Emergency Plan states that communications with NRC shall be tested monthly.

(2) Section 8.1.2 of the Emergency Plan states that communications with the States of New York and Connecticut within the ingestion pathway will be tested quarterly.

(3) Section 8.1.2 of the Emergency Plan states that communications with the Federal emergency response organizations, including the U.S. Coast Guard, will be tested quarterly.

(4) Section 8.1.3 of the Emergency Plan states that exercise and drill scenarios will be structured so as to allow free play for decision making, as much as possible, providing that the basic objectives of the drill or exercise are satisfied.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item.

13.3.15 Radiological Emergency Response Training

13.3.15.3 Prior Deficiencies

The plan requires version and/or additional information as follows:

(1) Indicate that the training for offsite response organizations that may enter the site will include site access procedures and the identity of the onsite individual who will control their activities.

(2) Confirm that the training provided for first aid personnel is equivalent to the Red Cross Multi-Media Course.

(3) Describe the provisions in the training program for offsite police, the onsite security force, and local Civil Defense/Emergency Severice personnel.

(4) Provide radiological training and retraining on a periodic basis for all radiation protection personnel to ensure awareness of the latest techniques and their capability for fulfilling emergency responsibilities.

Applicant's Response

(1) Section 8.1.1 of the Emergency Plan states that members of local offsite response organizations who may enter the site (i.e., fire and ambulance companies or police) shall be given the opportunity to receive an initial familiarization training session at Shoreham to ensure that they are familiar with the plant layout and their assigned duties in the event of an incident. Orientation and retraining courses for such personnel shall include site access procedures and identification of the individual in the onsite emergency organization who will control their activities.

(2) Section 8.1.1 of the Emergency Plan states that selected station personnel will receive Red Cross Standard First Aid and Personnel Safety Course training to ensure that at least two members of each shift hold a valid certification. This training is equivalent to the Red Cross Multi-Media Coarse.

(3) The applicant has stated that training and drills have been conducted at the Central Suffolk Hospital and for fire and ambulance personnel, but that Suffolk County police have not responsed to the offer for training. The training for onsite security personnel included access control, patrol, evacuation procedures, and accountability.

(4) The applicant has provided a matrix of emergency response training that lists the courses for each member of the emergency organization with the applicable implementing procedures. Training records are being fed into a computerized records system which will alert the training coordinator to schedule required retraining.

Conclusion

The staff finds that the applicant has provided an acceptable response to this item, with the exception that training of Suffolk County police is an open item.

13.3.17 Conclusions

The staff has reviewed the applicant's responses to the plan deficiencies cited in the SER and concludes, based on its review and evaluation and the applicant's commitments to be fulfilled before fuel loading, that the applicant has provided acceptable responses to the deficiences.

Coordination with New York State and Suffolk County is required to achieve an acceptable overall state of emergency preparedness in areas such as the public information program and training of Suffolk County police. The staff was informed that the Suffolk County draft radiological emergency response plan has been submitted to the county legislature for approval. An onsite appraisal by

NRC auditors of the applicant's capability to implement the Emergency Plan has been accomplished, and the findings of this appraisal will be transmitted to the applicant. The appraisal included the review and evaluation of Emergency Plan Implementing Procedures (EPIPs).

The applicant has committed to review and upgrade the EPIPs. The applicant has committed to complete the installation, testing, and development of procedures for the following before fuel load:

- computerized dose assessment system
- radiation and effluent monitoring system
- emergency response facilities
- decontamination facility
- EAL instrumentation set-points

The final determination on the adequacy of the state of emergency preparedness for the Shoreham site will be made and published in another supplement to the SER, following the applicant's fulfillment of the above commitments, the resolution of the findings of the onsite NRC Appraisal, and the review of findings by FEMA on the adequacy of state and local plans.

13.5 Plant Procedures

Section 13.5 of the Shoreham SER states

> All safety-related operating, maintenance and repair, testing and modification activities are conducted in accordance with approved, written procedures meeting the requirements of Regulatory Guide 1.33, Quality Assurance Program Requirements, and ANSI N18.7-1972.

This earlier finding was based on a commitment to this 1972 version of ANSI N18.7 that was made by the applicant in FSAR Section 13.5 (Revision 4, dated February 1977). At the request of the staff, in a letter to the NRC dated June 2, 1982, LILCO committed to comply with Revision 2 to Regulatory Guide 1.33. This RG, dated February 1978, endorses ANSI N18.1-1976. LILCO made this commitment with the following clarifications:

(1) The requirements established by ANSI N18.7-1976, paragraphs 4.3, 4.4 and 4.5 shall be governed by commitments made in Section 6 of Shoreham's Technical Specifications.

(2) ANSI N18.7-1976, paragraph 5.2.2(2) is intended to apply to procedural steps such as immediate actions in Shoreham Emergency Procedures.

(3) In reference to ANSI N18.7-1976, paragraph 5.2.13.2, the LILCO QA Manual allows and controls the release of certain specific nonconforming items, which could be caused by documentation deficiencies, to be installed but not operated. The conditional release does not change the documentation requirement, but allows for installation before all required documentation arrives on the site.

The clarification under item (1) above is concerned with the independent review program, the review activities of the onsite operating organization, and the audit program. These activities are discussed in Section 13.4 of SER Supplement No. 1.

The clarification under item (2) above is concerned with identification of tasks that require the operator to have memorized the procedural steps. The staff finds these clarifications under items (1), (2), and (3) to be acceptable for the Shoreham application and concludes, therefore, that LILCO's commitment to RG 1.33, Revision 2, with these clarifications, is also acceptable.

13.5.2 Operating and Maintenance Procedures

General

The applicant's plan for development and implementation of operating and maintenance procedures has been reviewed to determine the adequacy of the applicant's program for ensuring that routine operating, offnormal, and emergency activities are conducted in a safe manner. The following description and evaluation are based on information in the FSAR and the applicant's response to NRC TMI Action Plan Items (NUREG-0660 and NUREG-0737).

In determining the acceptability of the applicant's program, the criteria of SRP 13.5.2 (NUREG-0800) were used. The review consisted of an evaluation of (1) the applicant's procedure classification system for procedures that are performed by licensed operators in the control room, and the classification for other operating and maintenance procedures; (2) the applicant's plan for completion of operating and maintenance procedures during the initial plant testing phase to allow for correction prior to fuel loading; (3) the applicant's program for compliance with the guidance in Regulatory Guide 1.33, Revision 2, "Quality Assurance Program Requirements (Operation)" (March 1978) regarding the minimum procedural requirements for safety-related operations; (4) compliance with the guidance contained in ANSI 18.7-1976/ANS 3.2; and (5) the applicant's program for compliance with Task Action Plan (NUREG-0660) Item I.C.1, "Guidance for the Evaluation and Development of Procedures for transients and Accidents," for the development of Emergency Operating Procedure Guidelines.

Operating and Maintenance Procedure Program

The applicant has committed in the FSAR to a program in which all activities are to be conducted in accordance with detailed written and approved procedures meeting the requirements of Regulatory Guide 1.33, Revision 2 and ANSI 18.7-1976/ANS 3.2. The applicant uses the following categories of procedures for those operations performed by licensed operators in the control room:

general operating procedures
system operating procedures
emergency operating procedures
alarm response
temporary procedures

Other procedures cover the following areas:

initial test
maintenance
instrument and control systems
surveillance
emergency plan
health physics
chemistry
reactor engineering
plant security
radioactive waste management

The staff review disclosed that the applicant's program for use of operating and maintenance procedures meets the relevant requirements of 10 CFR 34, and is consistent with the guidance provided in Regulatory Guide 1.33 and ANSI 18.7-1976/ANS 3.2. Therefore, the staff concludes that the applicant's program is acceptable.

Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, NRC required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, and conduct operator retraining. Emergency operating procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980, and implementation of procedures and retraining were to be completed 3 months after emergency procedures guidelines were established; however; some difficulty in completing these requirements has been experienced. Clarification of the scope of the tasks and appropriate schedule revisions were included in NUREG-0737, Item I.C.1.

Pending staff approval of the revised analysis and guidelines, the staff will con tinue the pilot monitoring of emergency operating procedures described in Task Action Plan Item I.C.8 (NUREG-0660). The adequacy of the (BWR) Owners Group Guidelines will be identified for each near-term operating license (NTOL) during the emergency operating procedure review.

In a submittal dated June 30, 1980, the BWR Owners Group provided a draft of the generic guidelines for BWRs. The guidelines were developed to comply with Task Action Plan Item I.C.1(3) as clarified by NUREG-0737 and incorporated the requirements for short-term reanalysis of small-break LOCAs and inadequate core cooling (Task Action Plan Items I.C.1(1) and I.C.1(2)). In a letter dated October 21, 1980, from D. G. Eisenhut (NRC) to S. T. Rogers, the staff indicated that the generic guidelines prepared by GE and the BWR Owners Group were acceptable for trial implementation at Shoreham Unit 1. Additional information was requested by the staff and was submitted by the Owners Group on January 21, 1981. The staff is still reviewing this additional information prior to making a final conclusion on the acceptability of the guidelines for implementation on all BWRs. The guidelines are still considered acceptable for trial implementation at Shoreham Unit 1.

Based on its review of the emergency operating procedures developed from the BWR Owners Group Guidelines and its observation of the procedures being implemented on a simulator and in a walk-through in the control room, the staff has concluded that the guidelines have been adequately incorporated into the procedures. This fulfills the requirements of Item I.C.1 of NUREG-0737.

In accordance with NUREG-0737, Item I.C.7 nuclear steam supply system (NSSS) vendor review of low-power testing, power ascension testing, and emergency operating procedures is necessary to further verify adequacy of the procedures.

This requirement must be met before operation above 5% power.

The NSSS vendor, GE, will review the startup tests and emergency operating procedures before these procedures are implemented. The startup tests encompass the low-power testing and the power ascension testing phases. The applicant has committed to ensuring these reviews are complete prior to fuel load. The staff must review the applicant's resolution of vendor comments to confirm vendor review and incorporation of vendor comments into the procedures. The staff will confirm that this review is completed prior to operation above 5% power.

The staff and personnel from Battelle Pacific Northwest Laboratories reviewed the procedures forwarded by the applicant to the NRC to ensure that the procedures were consistent with the plant design and the BWR Owners Group guidelines, and that they incorporated applicable human factors considerations. The review resulted in two pages of general comments and numerous specific detailed comments on the procedures. The general comments included human factors considerations on the use of standard logic format, procedure identification, interaction with nonemergency procedures, inconsistency between emergency procedures and control room displays, and the inadequacy of the graphs that were included in the procedures. The specific comments included clarification and the locations of caution statements, the inclusion of action steps in cautions, the need for the addition of specific information to reduce operator judgments such as the preferred sequence for starting various systems, the need to add decision points to aid operator actions, and numerous references to changing words and using standard logic format to clarify action steps. A meeting was held with the applicant on September 16, 1981, to discuss the results of the review. During the meeting many of the comments were resolved by incorporating the recommended changes.

On October 16, 1981, a simulator exercise was held at the Limerick Training Center. Operators used the revised emergency operating procedures to respond to simulated transients and accidents. Scenarios were designed to require the concurrent use of procedures and transition among procedures. The scenarios varied from minor transients to accidents involving multiple system failures. The simulated transients and accidents included

(1) Loss of feedwater from leaks or breaks in feed lines, faulty valve operation, and pump failure

(2) Various initiating events followed by failure of various injection systems (e.g., RCIC, HPCI, LPCI) when needed for level control, level restoration, and containment control

(3) Turbine trip followed by a reactor trip

(4) Failure of offsite power with subsequent failure of a diesel generator

(5) Stuck-open relief valves resulting in loss of reactor pressure vessel water inventory and emergency conditions in containment

All of the emergency operating procedures were tested in responding to the simulations. The review team observed the exercises and discussed them in detail with the operators. Special emphasis was placed on (1) the need to use written emergency procedures and (2) evaluating the clarity and usability of the procedures. Several changes were made to the procedures as a result of the exercises and subsequent discussions. The changes involved sequencing of steps, labeling to help locate specific steps, and clarifying priorities of actions.

On October 17, 1981, the team of reviewers that had participated in the simulator exercises conducted a walk-through of the emergency operating procedures in the control room. The operators were presented with the initiating event (an intermediate-size break), with the desired sequence of steps. The operators then walked through the scenario, while the team of reviewers evaluated the operators' use of the procedures, the interaction of the operators with the control panels, and the interaction between the operators. The entire sequence was discussed in detail with the control room operators and the plant operations staff at the conclusion of the simulated event. The effective manner in which the operators used the emergency operating procedures indicates that the procedures are clear, properly sequenced, and compatible with the control room and its equipment.

During the review, it was noted that: (1) some plant specific data were not available and noted by a "(Later)", (2) the graphs referenced in the procedures need revision to improve their usability, and (3) a few additional changes are required in the procedures as noted during the simulator exercises. The applicant has committed to incorporate the plant-specific data when they are available and to make the agreed-to changes to the procedures and graphs. The staff will verify that the missing data and changes have been included in the procedures before issuance of an operating license.

The applicant provided the confirmatory information in a September 16, 1982 letter from J. L. Smith to H. R. Denton. The letter forwarded Revision 1 of the Shoreham EOPs, which included all remaining plant-specific information, revised graphs, and the changes based on the simulator exercises. The staff reviewed the applicant's submittal and found it acceptable except for the graphs. None of the graphs clearly specified unacceptable operating areas, and some of the graphs were unusable after reproduction because (1) the distinction between major and minor grid lines was indistinguishable, making it difficult to determine the specific points of ordinates on the graphs, and (2) minor grid lines were in many cases not continuous.

The staff discussed the problems regarding the graphs with members of the Shoreham operating staff. In a letter dated November 26, 1982, LILCO provided examples of adequate graphs, and committed to ensure that all graphs used in EOPs are consistently usable. The staff finds these changes adequate to confirm the LILCO commitments, and considers Item I.C.8 complete for issuance of an operating license.

15 ACCIDENT ANALYSES

15.3 Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) are events in which the scram system (reactor trip system) is postulated to fail to operate as required. Although this was not carried as an open item in the SER (Section 1.7), this subject has been under generic review by the NRC staff for several years.

In December 1978, Volume 3 of NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," was issued describing the proposed type of plant modifications the staff believes necessary to reduce the risk from anticipated transients with failure to scram to an acceptable level. The staff issued requests for the industry to supply generic analyses to confirm the ATWS mitigation capability described in Volume 3 of NUREG-0460. Subsequently, the staff recommended to the Commission that rulemaking be used to determine any future modifications necessary to resolve ATWS concerns as well as the required schedule for implementation of such modifications. Shoreham is subject to the Commission's decision in this matter.

It is the expectation of the staff that the necessary plant modifications will be implemented in 1 to 4 years following a Commission decision on ATWS. As a prudent course, to further reduce the risk from ATWS scram events during the interim before plant modifications deemed necessary by the Commission are completed, the staff requires that the following steps be taken:

(1) An emergency operating procedure should be developed for an ATWS event including consideration of scram indicators, rod position indicators, average range flux monitors, reactor vessel level and pressure indicators, relief valve and isolation valve indicators, and containment temperature, pressure, and radiation indicators. The emergency operating procedures should be sufficiently simple and unambiguous to permit prompt operator recognition of an ATWS.

(2) The emergency operating procedure should describe actions to be taken in the event of an ATWS, including consideration of manually scramming the reactor by using the manual scram buttons, changing the operation mode switch to the shutdown position, tripping the feedwater breakers on the reactor protection system power distribution buses, scramming individual control rods from the back of the control room panel, tripping breakers from the back of the control room panel, tripping breakers from plant auxiliary power sources feeding the reactor protection system, and valving out and bleeding off instrument air to scram solenoid valves. These actions must be taken immediately after detection of an ATWS event. Actions should also include prompt initiation of the residual heat removal system in the suppression pool cooling mode to reduce the severity of the containment conditions and actuation of the stanbdy liquid control system is a scram cannot be made to occur.

The Shoreham ATWS procedure was reviewed by members of the NRC staff and contractor personnel from Battelle Pacific Northwest Laboratories, and comments were discussed with the operations personnel. Based on its evaluation, the staff concludes that the Shoreham Unit 1 ATWS procedure provides an acceptable basis for licensing and interim operation of Shoreham Unit 1 pending the outcome of the proposed rulemaking on ATWS in accordance with GDC 10, 15, 26, 27, and 29 of 10 CFR 50, Appendix A.

22 TMI-2 REQUIREMENTS

I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants

This item is discussed in Section 13.5.2 of this report.

I.D.1 Control Room Design Review

SER Supplement No. 1, dated September 1981, lists those human engineering discrepancies (HEDs) and the applicant's design improvements that are to be implemented before fuel loading. The SSER also indicates that all improvements are subject to NRC review and verification after they are implemented by the applicant. On December 6, 1982, the staff conducted an onsite review of the improvement implementations.

Most improvements were satisfactory to the staff. Some items were not yet accomplished or did not yet satisfactorily resolve the HED. The staff position on each of these resolved items is listed in the Appendix D to this report. Actions implemented to correct discrepancies will be audited for confirmation by the NRC before an operating license is issued.

II.E.4.2 Containment Isolation Dependability

In SER Supplement No. 1, the staff reported that the applicant had committed to follow the resolution developed between the staff and the BWR Owners Group of the requirement for a high radiation isolation signal on those purge/vent isolation valves that are opened during operation mode 1, 2, or 3. Since the issuance of the first supplement, the applicant has committed to provide the required radiation isolation signal to the purge/vent isolation valves but has stated that the necessary equipment could not be installed and made operable until December 1983 at the earliest because of equipment unavailability and personnel limitations.

The staff finds it acceptable to allow the applicant to use the purge/vent system in the interim without automatic radiation isolation signals if the following conditions are met:

(1) The operation of the purge/vent system shall be limited to safety-related cases; i.e., relieving containment pressure, verifying vacuum breaker position in accordance with the Technical Specifications, and inerting/deinerting the containment.

(2) A control room operator shall be dedicated to the purge/vent system whenever the operation of the system is required, which is expected to be very infrequent, to ensure timely closure of these line in event of high radiation containment alarm.

(3) The applicant shall commit to not use the purge/vent system after December 31, 1983 without specific approval from the NRC, if the radiation isolation signal to the purge/vent system is not operable by that time.

Containment purge valves that do not satisfy the operability criteria set forth in BTP CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item II.6.f (NUREG-0800) during operational conditions. Furthermore, these valves must be verified to be closed at least every 31 days. The applicant must be in compliance with this position before receiving an operating license.

The Shoreham primary containment purge system purge valves are 18-in. valves as follows:

1T46*AOV038A
1T46*AOV038B
1T46*AOV039B
1T46*AOV039A
1T46*AOV039C
1T46*AOV039D
1T46*AOV038D
1T46*AOV038C

These valves are to be operated in the cold shutdown and refueling modes only. The valves listed above are to be sealed closed per SRP 6.2.4 Item II.6.f (NUREG-0800) in modes other than cold shutdown or refueling and verified closed every 31 days.

The primary containment purge system vent valves are 4-in. and 6-in Copes Vulcan Valves with D-100-100 operators as follows:

4 in. 1T24*AOV004 A
1T24*AOV004 B
1T24*AOV001 A
1T24*AOV001 B
6 in. 1T46*AOV078 A
1T46*AOV078 B
1T46*AOV079 A
1T46*AOV079 B

The applicant committed in a letter of November 23, 1981 to perform a test on a representative 6-in. valve as follows: During the containment structural integrity acceptance test, while the containment is pressurized to approximately 55 psig, one of the 6-in. vent valves will be opened, maintained open for a time sufficient to attain steady-state flow conditions, and then closed. Containment pressure will be monitored to ensure that pressure is maintained above 48 psig for the duration of this test. Flow direction during the test will be in the conservative direction, i.e., the direction that tends to open the valve. Closure time of the valve will be monitored to occur within 5 seconds. At the completion of the test, the valve is to be visually inspected.

In the applicant's letter of November 23, 1981, it was stated that the 4-in. air-operated valves are the same model as the 6-in. valve and use the same size 100 air operator. The tests performed on the 6-in. valve therefore will be a conservative demonstration of operability of the 4-in. valves.

Based on the applicant's commitment to perform this test, the staff found the applicant's program to meet this requirement satisfactory contingent on a confirmatory review of the results of this test before fuel load.

This test was performed during the structural acceptance test at a containment pressure of greater than 55 psig and demonstrated that the valve will close within 5 seconds as required. During the test, difficulty was experienced with the opening of one purge valve located inside the drywell because there was insufficient differential pressure for the valve operation. The test was performed satisfactorily using a valve physically located outside of the primary containment, and it was determined that, had the inside valve been opened, it would close also. This is because both sides of the valve diaphragm are vented during closure and the spring is the only closing force. Therefore, the results of this test are satisfactory and this commitment has been met.

II.F.1 (Attachment 1) Noble Gas Effluent Monitor

SER Supplement No. 1 stated that the applicant's procedures for monitoring noble gas effluent releases during an accident are acceptable subject to confirmatory documentation relating to (1) installation and calibration schedule, (2) calibration sources, (3) nature of display and recording, (4) assurance of the capability to obtain readings at least every 15 minutes during and following an accident, and (5) assurance a human factor analysis for changes that involve control room instrumentation will be provided.

Through submittals dated January 7, 1982 and February 17, 1982 (FSAR Revision 25), the applicant has provided confirmatory documentation relating to items (2) through (4) above. In addition, the applicant has confirmed that installation and calibration of the necessary instrumentation will be completed before startup. The applicant has, furthermore, provided the human factor analysis referred to above in response to TMI Action Plan I.D.1 (NUREG-0737), "Control Room Design Review."

Based on its review of the above mentioned submittals, the applicant's commitment to complete installation and calibration of the applicable instrumentation before startup and the applicant's performance of human factor analysis for changes that involve control room instrumentation, the staff concludes that the applicant's design provisions for monitoring noble gas effluent releases during an accident are now complete and that they meet the requirements of Attachment 1 to TMI Action Plan II.F.1 (NUREG-0737), "Noble Gas Effluent Monitor." Therefore, they are acceptable.

As mentioned in SER Supplement No. 1, a post-implementation review of the installed system, detailed drawings, and procedures for monitoring and calibration will be performed.

II.F.1 (Attachment 5) Containment Water Level Monitor

In SER Supplement No. 1, the staff reported that the lower limit of the containment water level monitor did not meet the requirements of NUREG-0737.

Since the issuance of this supplement, the applicant has provided the design details of a separate water level monitor that the staff agrees meets all of the requirements of NUREG-0737. However, this system may not be completely functional by commercial operation because of equipment procurement delays. The existing water level monitoring system, which deviates from the NUREG-0737 functional requirements only by being 3 ft short on the lower range requirement, will be operational. Because of the relatively short period of time the improved water level monitoring system will not be completely operational and because there is a water level monitoring system that meets almost all of the functional requirements of NUREG-0737, the staff finds the associated schedule delay acceptable.

II.K.3.28 Verify Qualification of Accumulators on Automatic Depressurization System Valves

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensees and applicants must demonstrate that the ADS valves, accumulators, and associated equipment and instrumentation meet these requirements and are capable of performing their functions during and following exposure to hostile environments, taking no credit for nonsafety-related equipment or instrumentation. Additionally, air (or nitrogen) leakage through valves must be accounted for in order to ensure that enough inventory of compressed air is available to cycle the ADS valves. If this cannot be demonstrated, it must be shown that the accumulator design is still acceptable.

Shoreham is designed for four modes of supplying air to the ADS valves (normal operation, short-term supply, intermediate supply, and long-term supply). During normal operation air is supplied to the intermediate-term accumulators by a compressor. The short-term supply systems consist of individual accumulators on each of the ADS valves. The individual accumulators are isolated from the supply header by means of check valves.

The intermediate supply system consists of two accumulators sized for at least 55 actuations of a safety relief valve (SRV) for a minimum of 48 hours. One intermediate supply system accumulator supplies four of the seven ADS valves; the other serves the remaining three ADS valves. The intermediate accumulators and connected piping are designed to ASME Code Section III, Class 2 requirements.

The long-term supply system consists of an air connection outside of the reactor building for replenishing the air supply. The air connection and connecting piping are designed to ASME Code Section III, Class 2 requirements. A portable compressor or air/nitrogen bottle may be connected to the system to sustain SRV operability beyond the 48 hours.

To verify the leaktight integrity of the ADS accumulator system, the applicant will leak test each redundant train during each station refueling outage. The ADS SRV pilot valves will be opened. The A header will be isolated by closing valves 1P50*MOV-113A and 105A and verifying that 1P50*MOV114A opens. The A header pressure will be established at 90 psig $\pm$ 0.5 as measured on a header pressure test gauge. The train B header will be similarly tested. This test will be performed over a time period of 45 minutes, which corresponds to a pressure decay rate of approximately 1.3 psi/hr. At this decay rate, the pneumatic supply to the ADS SRV pilot valves will be operable for a minimum of 48 hours prior to establishment of long-term pneumatic supply.

If at the end of the accumulator header leak test, it is found that pressure drops by greater than 1 psig, the cause shall be identified/corrected and the system will be retested to verify leaktight integrity.

The staff concludes that the applicant's proposed testing of the ADS supply system for Shoreham is sufficient to ensure the availability of pneumatic supply to the ADS SRV valves until long-term supply can be made available by means of a portable compressor or air nitrogen bottles.

APPENDIX A

MARK II CHUGGING LOAD SPECIFICATION
EFFECTS OF DESYNCHRONIZATION

1 INTRODUCTION

This appendix documents the work performed by Professor George Bienkowski, a staff consultant, on the effects of desynchronization on the Mark II chugging load specifications. This work was performed as part of the NRC technical assistance program at Brookhaven National Laboratory to support the staff in reviewing the chugging load specifications. This appendix also documents the results of the additional studies performed by the Mark II Owners Group and the staff's evaluations and conclusions.

1.1 Background

Following receipt of Professor Bienkowski's report, the staff conducted a preliminary evaluation of its contents and concluded that a deficiency exists in the chugging methodology proposed by the Mark II Owners Group. As a result, the staff recommended that additional studies of this issue be conducted with input from the owners of Mark II plants. The staff also concluded that other conservatisms in the chugging loads are such that adequate safety margins are maintained to allow licensing activity efforts to continue.

This appendix is the product of the review conducted by the Mark II Owners Group, the NRC staff, and Professor Bienkowski. The remainder of this section and Section 2 contain Professor Bienkowski's report in its entirety. Section 1.2 presents an executive summary of Professor Bienkowski's report; Section 2 presents the mathematical evaluation of the effect of desynchronization on chugging loads. Section 3 contains the Mark II Owners Group comments on Professor Bienkowski's report and the results of additional analyses performed to confirm the conservatisms of the existing chugging load specifications. Section 4 presents the staff's evaluation of the Mark II Owners Group analyses and conclusion regarding the chugging load specifications.

1.2 Executive Summary

While the data bases, source strengths, or calculational procedures may differ between the generic chugging load specifications and the chugging load specifications for the Susquehanna Steam Electric Station (SSES) and for the Washington Public Power Supply System (WPPSS), the procedure for desynchronization of chug start times is identical. Both the symmetric and asymmetric specifications are based on the application of the minimum variance set of start times (at the N vents of the plant) from 1000 such sets, based on uniform probability distribution within a 50 msec time window. The same set of start times is used for all of the sources in the specifications of both the symmetric and asymmetric loading.

The NRC staff review of the specifications concluded that the data bases, deduced design sources, and application are conservative for the symmetric load, and, while difficult to quantify for the asymmetric case, provide a reasonable measure of asymmetry. The justification of the selection of the minimum variance set of start times was based on an examination of the root mean square (rms) values of vertical force and overturning moment. The decrease of rms value of vertical force with start time variance and the relative insensitivity of overturning moment rms aplitude convinced the staff that the specification was "reasonable." No information was presented, by either GE or the owners of individual plants, on the sensitivity of the frequency content of the loads to the specific selection of start times. Clearly the underestimation of even small amounts of energy at major natural frequencies of the overall plant configuration could lead to potential nonconservatism in individual loads or accelerations at various structural components.

Section 2 (The Effects of Desynchronization on Chugging Loads) examines the potential impact of the specific selection of a single set of start times on the frequency content in the vertical force and overturning moment for three plant configurations and specifications (generic, SSES, and WPPSS). Figures A.2.11 to A.2.14 summarize the results. All four figures show that both the vertical force and overturning moment can have a reasonable chance of 1 in 1000 of exceeding the specification by as much as a factor of 10 at frequencies with significant energy in the source. Alternatively, one can interpret these results to conclude that there is a high exceedance probability (approaching 1) that at some frequency in the 20- to 50-Hz range the true load on the structure will substantially exceed the specified load.

The consequences of the potential nonconservatism on the response spectrum level at specific nodes of the structure are difficult to assess without access to the full computer codes for the individual plants. The analysis and calculations of Section 2 suggest, however, that a specific set of start times will always produce substantial cancellation of any measure of structural response at some frequencies above 20 Hz. Because these frequency "holes" are dependent on the specific selection and assignment of start times to individual vents, it is virtually impossible to guarantee a low exceedance probability at any frequency above 20 Hz on the basis of the specified desynchronization. While the choice of the minimum variance set optimizes the synchronization to maximize the symmetric load at low frequencies, no generalization of this hypothesis can be justified either at higher frequencies or for other measures of structural response.

The possible high probability of exceedance of the specified chugging loads, of course, does not necessarily imply lack of safety margin of any individual component in the plant. Other loads could be bounding in the relevant frequency range or other design constraints may have resulted in safety margins well above those imposed by chugging. Individual assessment, component by component, is clearly a difficult procedure at best. If one retains the "physical" intuition that the symmetric and asymmetric loadings provide two "extreme" conditions that adequately describe the "major" structural excitations, one has a clear and attainable objective. The specification must provide loading conditions with low exceedance probability of both the vertical force and overturning moment, or frequency regions where the exceedance probability is high have to be bounded by other specifications. For instance, the generic condensation oscillation load provides adequate margin for the vertical force power spectral density (PSD) in the range of 20 to 50 Hz because of the synchronous application of the loading. Unfortunately, the lack of any appreciable energy in that frequency range in the KWU chugging/oscillation specification fails to provide the same conservatism for the SSES plant. No other asymmmetric loading appears as an obvious candidate to bound the chugging induced overturning moment.

Relatively simple "fixes" to the present specification can define loading conditions that provide an exceedance probability of less than 1 in 1000 for the vertical force and overturning moment. For instance, the addition of a loading specification which applies the sources at about 20% amplitude but synchronized in time, will provide adequate bounds over the 20- to 50-Hz range for the generic, SSES, and WPPSS symmetric loads. The application of the asymmetric loading with an asymmetric factor increased slightly above the specification and full synchroniztion in time can ensure a low exceedance probability of the overturning moment over the entire frequency range. Whether these are the best, or the easiest procedures, to provide adequate conservatism is not obvious without a more detailed examination of the actual application of these loading conditions.

2 THE EFFECTS OF DESYNCHRONIZATION ON CHUGGING LOADS

2.1 Introduction

A substantial body of experimental evidence exists to indicate that chugging has a random character. While mean values and standard deviations exhibit dependence on both the properties of the fluid and the nature of the steam being condensed, any individual chug amplitude can be defined only on a probabilistic basis. Both subscale and fullscale multi-vent tests[1-4] also indicate that although on a gross time scale associated with the repetition rate, events at different vents are synchronized, on the time scale of the chug itself, start times have a highly random character.

The proper assessment of a chugging design load (or response spectrum) on a Mark II containment must take full cognizance of the stochastic nature of the phenomena. An evaluation of the conservatism associated with any loading configuration or any local response can only be performed on the basis of an exceedance probability. This is true whether or not the probabilistic nature of the data base is used directly or indirectly in defining the loading condition. It is also true that different measures of a loading may yield different levels of exceedance probabilty for a given loading configuration. Conversely, a given exceedance probability will require different loading configurations if different global or local measures of the load are used.

For a combination of practical and historical reasons, the load specification for Mark II plants consists of two loading configurations, the symmetric and the asymmetric cases. The measures chosen for evaluation of conservatism in the loads are total vertical force for the symmetric case and total overturning moment for the asymmetric loading configuration. Although the data base and detailed application are different, the fundamental definitions of the loading configurations are essentially the same in the generic and the plant-unique methodologies.

The symmetric loading configuration consists of the application of chugs of equal strength A at all vents (WPPSS applies an increased amplitude at three vents). The start times, however, are chosen from that sequence of random numbers that produced a minimum variance in 1000 Monte Carlo trials from a uniform distribution within a 50 msec time window. In the WPPSS methodology, each group of three vents at given angle ϕ is taken to chug synchronously. The source strength and time history are different in the generic and the SSES and WPPSS methodologies. All procedures, however, are a source strength that is greater than the mean of the data on which it is based to account for the probability of an event significantly different from the "average" or "expectation value" event.

The asymmetric configuration is obtained by distributing the source strengths asymmetrically: $(A+B\cos\phi)$ distribution in SSES, $(1+\alpha)A$ and $(1-\alpha)A$ on opposite sides of a containment diameter in the generic methodology, and $A(1+CR\cos\phi)$ in WPPSS. The values B, C, and α, in each case, are chosen from some evaluation of the variance in amplitudes of the respective data bases for the methodologies. The start times, however, are chosen in exactly the same way as in the symmetric case.

Because each of the design loading configurations consists of a "single" distribution of source strengths and start times at the vent exits in the containment, the quantitative value of exceedance probability for any given load associated with that specific configuration is difficult to assess. The use of a "minimum variance" event in assigning start times appears intuitively conservative for the net force as a measure of symmetric load. The use of the minimum variance event is much more difficult to justify for the asymmetric case.

In order to provide a formalism within which the exceedance probabilities of the design load specifications can be assessed, a formal fully probabilistic analysis is presented in Section 2.2. These theoretical results are compared to theoretically predicted results using the SSES specification in Section 2.3. Some results of Monte Carlo computations are presented in Section 2.4, and a discussion of the implications on the symmetric and asymmetric load specifications is presented in Section 2.5.

2.2 Stochastic Formulation

Because of the linear nature of both the fluid description (IWEGS/MARS)[5] and the structural analysis (ANSYS),[5] any measure of either global or local load can be represented as a sum over the responses due to each source applied independently at each vent exit. The specific measure of response due to any individual source can be represented in terms of linear operator L_υ (Green's function or influence coefficient) acting upon that source $S_\upsilon(t)$. A generalized response R_υ due to a source of amplitude A_υ with a start time t_υ can be symbolically written as:

$$R_\upsilon = L_\upsilon(A_\upsilon S(t-t_\upsilon)) = A_\upsilon L_\upsilon(S(t-t_\upsilon)) \qquad (2.1)$$

The total response to all of the sources in a Mark II containment can then be obtained by a straightforward summation,

$$R = \sum_{\upsilon=1}^{N} A_\upsilon L_\upsilon(S(t-t_\upsilon)) \qquad (2.2)$$

where N is the number of vents.

To facilitate the stochastic analysis and to provide better measures of the loading it is convenient to replace the time variable by the frequency variable through the Fourier transform $\bar{f}(\omega) = \int_{\infty}^{\infty} f(t)e^{i\omega t}dt$. The response measure R_υ can now be written as

$$\bar{R}_\upsilon(\omega) = A_\upsilon e^{i\omega t_\upsilon} H_\upsilon(\omega)\bar{S}(\omega) \qquad (2.3)$$

where $H_\upsilon(\omega)$ is now the operator in Fourier space and $\bar{S}(\omega)$ is the Fourier transform of the normalized source with a start time at $t_\upsilon=0$. Note that $H_\upsilon(\omega)\bar{S}(\omega)$ can be considered the unit response or just the contribution at frequency $f = \frac{\omega}{2\pi}$ to the response $\bar{R}(\omega)$ from a normalized source with a zero start time. The total response $\bar{R}$ at frequency f is then just the sum over the amplitude factors $A_\upsilon e^{i\omega t_\upsilon}$ times the unit responses. Because $A_\upsilon e^{i\omega t_\upsilon}$ is a complex number, there will clearly be both an inphase contribution $Re(A_\upsilon e^{i\omega t_\upsilon}) = A_\upsilon \cos \omega t_\upsilon$ and an out-of-phase contribution $Im(A_\upsilon e^{i\omega t_\upsilon}) = A_\upsilon \sin \omega t_\upsilon$.

Because both A_υ and t_υ are random variables, each with an associated probability distribution, the specific response $R(\bar{\omega})$ will clearly be random in character with some resultant probability distribution $P(\bar{R})$.

For N sufficiently large, the central limit theorem[6] states that $P(\bar{R})$ will approach the normal distribution with a mean $\mu = \sum_{\upsilon=1}^{N} \mu_\upsilon$ and a variance $\sigma^2 = \sum_{\upsilon=1}^{N} \sigma_\upsilon^2$ under some relatively weak condition on boundedness of the random variables $\bar{R}_\upsilon$. Experience shows that, unless the probability distribution of $\bar{R}_\upsilon$ is very peculiar, the number N need not be very large for the normal distribution to become a very good approximation for $\bar{R} = \sum_{\upsilon=1}^{N} \bar{R}_\upsilon$. We will therefore examine probabilities of any measure of loading $\bar{R}(\omega)$ exceeding some pre-selected value on the assumption that N of the order of 100 in a Mark II containment is sufficient for the central limit to hold.

The mean value μ_υ and the variance σ_υ^2 can be obtained on the basis of the prescribed probability density $f_a(A)$ for the amplitudes A_υ and the probability density $f_t(t_\upsilon)$ for the start times t_υ. If we assume these probability densities to be independent of each other and further take

$$f_t(t_\upsilon) = 1 \qquad \frac{\tau}{2} \leq t_\upsilon \leq \frac{\tau}{2}$$

$$= 0 \qquad t_\upsilon > \frac{\tau}{2} \tag{2.4}$$

the resultant mean values become

$$\mu_\upsilon = \mu_a \frac{\sin(\omega\tau/2)}{(\omega\tau/2)} H_\upsilon(\omega)\bar{S}(\omega) \qquad \text{in phase}$$

$$= 0 \qquad \text{out of phase} \tag{2.5}$$

where μ_a is the mean value of the chug amplitudes.

$$\mu_a = \int_0^\infty f_a(x)\, x\, dx \tag{2.6}$$

The associated variances become

$$\sigma_{\upsilon_1}^2 = \left(\frac{1 + (\frac{\sigma a}{\mu_a})^2}{2} \left(1 + \frac{\sin n\omega\tau}{\omega\tau} \right) - \left(\frac{\sin(\omega\tau/2)}{\omega\tau/2} \right)^2 \right) \mu_a^2\, H_\upsilon^2(\omega) \bar{S}^2(\omega)$$

in phase

and (2.7)

$$\sigma_{\upsilon_2}^2 = \left(\frac{1 + (\frac{\sigma a}{\mu_a})^2}{2} \left(1 - \frac{\sin n\omega\tau}{\tau\omega} \right) \right) \mu_a^2\, H_\upsilon^2(\omega) \bar{S}^2(\omega)$$

out of phase

where σ_a^2 is the variance of the amplitude probability distribution

$$\sigma_a^2 = \int_0^\infty fa(x)(x - \mu_a)^2 dx \tag{2.8}$$

The normalized mean $\bar{\mu}_\upsilon$ and the associated normalized standard devisions $\bar{\sigma}_{\upsilon_1}$ and σ_{υ_2} (normalization is performed by dividing by the response due to the average chug $\mu_a H_\upsilon(\omega)\bar{S}(\omega)$ is shown as a function of $\omega\tau$ in Figures A.2.1a, b, and c. The corresponding frequencies $f = \omega/2\pi$ are also indicated on the abscissa for τ = 50 msec. The standard deviations are plotted for several normalized variances of the amplitude distribution $\bar{\sigma}_a^2 = (\frac{\sigma_a}{\mu_a})^2 = 0.1$, 0.3, 0.5 and 1.0. Note that for low values of $\omega\tau$ (near synchronization) the inphase standard deviation from the mean is primarily determined by the variance of the chug amplitudes but at higher values of $\omega\tau$ both the inphase and out-of-phase standard deviations arise primarily from the dephasing of start times and are only weakly affected by the variance of amplitudes.

The specific response amplitude (a global load, local deflection, or response spectrum) for any given exceedance probability p_e can be simply determined from the normal probability distribution as

$$\bar{R}_1(\omega,p_e) = \sum_{\upsilon=1}^{N} \mu_\upsilon(\omega) + z(p_e) \sqrt{\sum_{\upsilon=1}^{N} \sigma_{\upsilon 1}^{2}} \quad \text{in phase}$$

(2.9)

$$\bar{R}_2(\omega,p_e) = z(p_e) \sqrt{\sum_{\upsilon=1}^{N} \sigma_{\upsilon 2}^{2}} \quad \text{out of phase}$$

where $z(p_e)$ is a factor obtained from the normal distribution. For $p_e = 10^{-3}$, $z_e \approx 3.09$ and for $pe = 10^{-5}$, $z_e \approx 4.28$. If the total amplitude $R(\omega) = \sqrt{\bar{R}_1^2(\omega)+\bar{R}_2^2(\omega)}$ is to be determined, or some combination such as $\sqrt{\bar{R}_x^2+\bar{R}_y^2}$ where $\bar{R}_x$ and $\bar{R}_y$ are two loads along mutually orthogenal axes, the results can be determined from different integrals of the multi-dimensional normal distribution. While in general the results may be very complicated, for the low levels of $p_e \leq 10^{-3}$ the effect is primarily to change the function $z(p_e)$ to some new function $z(p_e;\ \mu,\ \sigma_1,\ \sigma_2)$. For instance if R_x and R_y represent moments about two perpendicular axes the results for a symmetric containment show that if one picks an axis and asks for the exceedance of a fixed moment about that axis for $p_e = 10^{-5}$, $z_e = 4.28$ while if one asks for the exceedance of the magnitude of the load in any direction at the same p_e, $z(p_e;\ o,\ \sigma,\ \sigma) = 4.80$ implying only a 12% higher amplitude. Alternatively, the magnitude of the moment about a fixed axis for an exceedance probability of 10^{-5} corresponds to the magnitude independent of direction at an exceedance level of about 10^{-4}. Therefore, rather than getting involved with the complexities associated with any loads or Fourier coefficients that must be summed as the square root of the sum of the squares, we shall examine the net inphase vertical force and the net inphase overturning moment about a fixed but arbitrary axis as measures of the symmetric and asymmetric loads. In the following section these loads are computed based on the analysis above and compared to the Pennsylvania Power and Light Co. (SSES) specification.

2.3 Stochastic Analysis of Vertical Force and Overturning Moment

2.3.1 Symmetric Load

If one uses the total vertical force as a measure of the symmetric load as done in the SSES Design Assessment Report (DAR) each source's contribution to the force $H_\upsilon(\omega)\bar{S}(\omega)$ corresponds to the Fourier transform of the integral of the pressure from the vent υ over the entire basemat. Because the major contribution comes from near the vent, except for fringe effects near the pedestal and outer wall, each of the contributions can be considered identical and interpreted as the basemat pressure times some effective area ($\bar{P}(\omega)A$). Using this interpretation we can deduce the value of $(\sigma_a/\mu_a)^2 \sim 0.11$ as being consistent with the DAR evaluation of the low frequency filtered amplitudes and with the rms values in both GKM and JAERI. The results of Figure A.2.1 can be applied together with equation 2.10 to plot the effective symmetric amplitude factor $\bar{A}_s(\omega)$ versus frequency for any desired exceedance probability. The inphase components of the vertical force are shown as the solid lines in Figure A.2.2a for $p_e = 10^{-3}$ and 10^{-5}. The DAR load specification is represented by dashed lines, with both the expectation value μ for totally random selection of start times and the 3σ deviation from that value shown. Because the specification uses the most synchronized set out of 1000 sets of starting times and the symmetric load increases with increasing synchronization, the $\mu+3\sigma$ is considered to be more representative of the specification. The inphase vertical force, therefore, is expected to be represented generally conservatively over the entire relevant frequency range, with the greatest conservatism near the lower frequencies where most of the energy is concentrated.

The out-of-phase component can also be analyzed by the present technique and compared to the specification. As can be seen from Figure A.2.1 the major contribution will come at higher frequencies. If the start time set is assumed to be the most conservative out of 1000 trials for the out-of-phase component, the load resulting from the specification corresponds to an exceedance probability of 10^{-3}. Because the contribution of the out-of-phase component to the total amplitude of the vertical force at low exceedance probability is small, the proper matching of that component is not very important. For the present analysis at $p_e = 10^{-5}$ the total amplitude is never more than 12% higher than the inphase component; thus even if the specification start times were to produce no out-of-phase component, the comparison would not significantly change from that shown in Figure A.2.2a.

2.3.2 Asymmetric Load

If one uses total overturning moment as a measure of asymmetric loading as done in the SSES DAR, each source's contribution to the moment $(H_\upsilon(\omega)\bar{S}(\omega))$ corresponds to the Fourier transform of the integral over the basemat of the pressure multiplied by a moment arm from the selected axis. As in the symmetric case, the fact that the major contribution comes from beneath the vent allows one to approximate $H_\upsilon(\omega)\bar{S}(\omega)$ by $L_\upsilon\bar{P}(\omega)A$, where L_υ is the perpendicular distance from the selected axis to the vent location. Using this interpretation plus the value of $(\sigma a/\mu a)^2 = 0.11$ deduced from the amplitude variance, we can generate from Figure A.2.1b and the data presented in the DAR the effective asymmetric amplitude factor $\bar{A}_a(\omega)$ for any exceedance probability p_e based on the present fully stochastic analysis.

Figure A.2.2b shows a comparison of the in-phase component of $\bar{A}_a(\omega)$ from the present analysis for exceedance probabilities of 10^{-3} and 10^{-5} (shown as solid lines) to the possible results from the application of the DAR load specifications. The fact that the asymmetric load depends not only on the specific selection of start times but also on the distribution of those start times around the containment makes it difficult to precisely define the loading arising from the specification. For the asymmetric specification, both the expectation value and the $\pm 3\sigma$ values are shown, roughly covering the range of possibilities within 1000 trials. Because the minimum variance in start times does not necessarily lead to highest loads, as in the symmetric case, we cannot assume that the specification will produce the $(\mu+3\sigma)$ values. For frequencies below about 10 Hz for τ = 50 msec, the specification is clearly conservative, with even the worst result $(\mu-3\sigma)$ always bounding the 10^{-3} exceedance level.

For higher frequencies the asymmetric specification is clearly not conservative. However, the symmetric specification with nonsynchronized events leads also to a moment and can thus in principle cover the high frequency asymmetric amplitude factor. Shown on Figure A.2.2b are the plots for the resulting amplitude for σ, 2σ, and 3σ values. Although a very fortuitous choice of distribution of start times around the containment could approach the 3σ values and thus correspond to the 10^{-5} exceedance probability even at high frequencies, this is clearly unlikely. The more probable result around 1σ leads to an exceedance probability of about 10^{-1} for frequencies above about 15 Hz (see Section 2.4).

The results of Figure A.2.2 show clearly that the use of amplitude factors in the SSES DAR specification coupled with random selection of start times leads to loads with statistical properties that are generally more conservative than the random selection of both amplitudes and start times that could be considered the more "realistic" representation of multi-vent chugging. The more disturbing feature is the behavior of the actual application of the specification (a single application of minimum variance start times) at frequencies above 15 Hz. Because of the possible cancellation of contributions from different vents, a single selection of start times can and indeed does lead to "holes" in frequency at which, regardless of the source, no net effect on vertical force or moment may be transmitted. This appears particularly pronounced for the asymmetric load. In order to investigate this effect of desynchronization more fully, many Monte Carlo calculations have been performed. The results are presented in Section 2.4.

2.4 Monte Carlo Computations Compared to Symmetric and Asymmetric Load Specifications

In order to more fully evaluate the potential lack of conservatism resulting from a single application of a specific set of minimum variance start times, a number of Monte Carlo calculations were performed for the SSES, generic, and WPPS configurations and specifications. For each of these, the net vertical force and overturning moment were computed on the same basis as the theoretical evaluations in Section 2.3, i.e., equal contribution from each vent to the force and a moment contribution proportional to the moment arm of each vent about a preselected axis.

For each of the plant configurations considered, 1000 Monte Carlo trials were performed. Start times were selected randomly from a uniform distribution within a 50-msec time window. For the variable amplitude cases source amplitudes were selected from a normalized distribution using a JAERI established variance of $\bar{\sigma}_\alpha = 0.11$. The symmetric and asymmetric amplitude factors and spatial distributions were selected for each configuration on the basis of the relevant specification. A number of statistical measures were calculated and compared to the theoretical results from Section 2.3 where appropriate. The inphase and out-of-phase expectation values and standard deviations, determined "experimentally" from the 1000 trials, agree so well with the "theoretical" values that on a figure such as A.2.1 or A.2.2 they are indistinguishable.

A summary of the results is presented in Figures A.2.3 through A.2.8. For each plant configuration and corresponding specification the vertical force results are presented as the square of the force amplitude normalized by N times the contribution from a single vent versus the frequency. (N is the number of vents in the configuration.) The overturning moment is presented as the amplitude squared normalized by the results from synchronized sources distributed geometrically as shown on the figure label. Both results can be interpreted as the PSD one would obtain with random phasing, normalized by the PSD for synchronized sources and specifed spatial distribution. These results are therefore independent of the frequency content of the source. The figures show: (1) the effect desynchronization has on the transmission of the frequency content in the source to overall measures of structural response such as force and moment and (2) the comparison of true bounds in 1000 trials to the results of the direct application of the appropriate specification.

Figure A.2.3 shows the PSD for the vertical force for the SSES plant normalized by the PSD one would obtain for synchronized application of average chugs. The "true" bound of 1000 trials of variable amplitude chugs applied at random start times to the SSES plant configuration of 87 vents is shown as a solid line. The use of the symmetric specification amplitude factor, defined in the DAR with random start times, leads to a bound in 1000 trials that is conservative over the entire frequency range (designated as -·-). However, the use of the amplitude factor together with the application of the specific set of start times with minimum variance is only conservative at frequencies below 20 Hz. Because minimum variance does not uniquely determine the start times, two results from two different sets of 100 trials are shown (dashed lines ---). Note that the specific frequency "hole," where the PSD will be virtually zero regardless of the energy content within the source, does depend on the particular minimum variance set. Regardless of the specific set chosen, the DAR specification can lead to high exceedance probability over some significant (5 to 10 Hz) frequency range at some frequency above 20 Hz.

Figure A.2.4 shows similar results for the PSD of the overturning moment for the SSES configuration normalized by the PSD one would obtain from a fully synchronized application of the chugs with a $(1 + \cos \phi)$ distribution of amplitudes. Note that again the use of the asymmetric load factor of 0.4 combined with desynchronized start times leads to a generally conservative bound within a 1000 trials. Four possible applications of the specification using a minimum variance set of start times lead to a very pronounced lack of conservatism above about 10 Hz. Clearly if the overturning moment is a reasonable measure of a loading configuration significant to the structure, the DAR specification may totally miss energy input at quite moderate frequencies of 10 to 50 Hz.

The generic specification does not explicitly use any statistical information on the distribution of amplitudes. To compare the results of the "more realistic" variable amplitude chugging to the generic specification, the effective amplitude factor for each of the generic sources has to be estimated. Table A.2.1 gives the results computed on the basis of the rms pressure in the

Table A.2.1 Generic chugging amplitude factor

Source	Run	No. of Chugs*	RMS Statistics Peak	Mean	GE Avg.	Spec	A = Spec/Mean
801	26	4	5.46	4.30	4.96	5.54	1.29
802	19	4	4.21	2.16	3.05	3.22	1.49
803	1	5	4.42	2.33	3.28	3.45	1.48
804	25	4	5.16	3.51	4.53	5.13	1.46
805	15	4	5.19	3.05	4.31	4.38	1.43
806	15	4	3.42	2.41	3.13	3.34	1.39
807	20	5	4.26	3.14	3.63	3.55	1.13
808	1	5	4.42	2.33	4.42	5.06	2.17
809	25	4	5.16	3.51	5.16	6.90	1.97
810	15	4	5.19	3.05	5.19	5.37	1.76

* (± 20% mass flow) used as criterion

Generic Chugging Report.[7] The amplitude factor is based on the ratio of the specified source rms pressure to the "local" mean rms pressure of the chugs within a ± 20% mass flow variation around the "key" chug used for that particular source specification. For all of the sources except No. 807 the amplitude factor is $>$ 1.29, which is quite comparable to the SSES specification. Source 807 comes from run 20 in 4TCO near a region of nearly constant chug amplitude resulting in an effective amplitude factor of 1.13. Because the PSD of Source 807 is bounded by other sources at frequencies above about 10 Hz, an amplitude factor of 1.29 was used in the comparisons of the Monte Carlo trials to the generic specification.

Figures A.2.5 and A.2.6 show analogous information to that shown in Figures A.2.3 and A.2.4 but using the generic specifications[7] for comparison, and the same 87-vent configuration. The conclusions are not very different. The vertical force spcification can be appreciably below the bound of 1000 trials above 20 Hz, and the overturning moment specification can be orders of magnitude below the "true" bound for virtually any frequency above about 5 Hz.

The WPPSS specification, while using very different calculational procedures,[8] relies on the minimum variance set of start times as done in the generic and SSES specifications. The start tmes, however, are selected for groups of 3 vents going synchronously rather than being selected for all 102 vents independently. Figures A.2.7 and A.2.8 show the results of the specification compared to the "true" bound basd on 1000 trials of randomly selected amplitudes and start times for all 102 vents. Note that the greater synchronization produced by grouping of three vent sets is a conservative procedure. The vertical force specification, therefore, is generally near the "true" bound over almost the entire relevant frequency range. The overturning moment, while showing the characteristic sensitivity to the specific "minimum variance" set chosen, does come closer to the "true" bound than either the generic or SSES specification. Note, however, that an "unlucky" choice of the minimum variance set could still lead to a PSD "hole" at virtually any frequency above 5 Hz.

Two general conclusions from Figures A.2.3 to A.2.8 can be drawn:

(1) The amplitude factors for the symmetric load and the spatial distributions for the asymmetric load lead to representations of the loading conditions with statistical properties that produce a higher load at the same exceedance probability than that resulting from statistically distributed chug amplitudes.

(2) The specification of a single set of start times (no matter how determined) does not give a result which corresponds to, even approximately, the same exceedance probability at all frequencies. Indeed frequency "holes," where virtually no energy is transmitted from the source to the resultant measure such as force or moment, will in general rise for any single set of start times. This conclusion is relevant to any other response of the structure whether local or global, although the importance of this effect may be significantly reduced for local measures of structural response.

The conservatism of the loading on a Mark II containment depends both on the conservatism in the source strengths and on the methodology of application. Reference 7 shows an application of the generic sources to the JAERI facility compared to the JAERI data. In order to match statistics of the application to the quantity of data available, the theoretical computation used the bounds of 8 "Monte Carlo" trials averaged over 20 such sets of 8 trials each. The information presented in Figure 6.3 of Reference 7 suggests a conservatism in the source strength of the order of three or higher over most frequencies up to 50 Hz. In order to test whether this conservatism could be consumed by the demonstrated nonconservatism in the desynchronization specification, Monte Carlo trials analogous to those presented in Figures A.2.3 to A.2.8 were performed for the JAERI configuration.

Figure A.2.9 shows a comparison for the normalized PSD of the vertical force (the moment is not meaningful for this configuration) as computed for Figure 6.3 in Reference 7 to the results based on variable amplitudes and synchronization based on the specification. The same potential nonconservatism exists for this facility as for the full-scale plant configurations, although the specific minimum variance results may actually be more conservative than the (bound of 8 average over 20) GE result at frequencies below about 25 Hz. If one applies the ratio of the "minimum variance" result to the GE result to Figure 6.3 of Reference 7, one can compare the actual application of the generic specification to the measurements in the JAERI facility. Figure A.2.10 shows such a comparison. Note that, above 25 Hz, the specification does not provide any conservatism over the data, and may indeed miss a small, although significant, amount of energy above 40 Hz. While the source strengths in the JAERI facility may indeed be conservatively bounded by the specified sources based on 4TCO data, the application of the specified desynchronization could lead to either no margin or even some nonconservatism for the seven-vent configuration in JAERI. While no information on asymmetric loading can be deduced from JAERI, comparison of Figures A.2.2 to A.2.5 shows that a lack of margin in the symmetric load suggests a very high potential for exceedance in the asymmetric load because of the greater sensitivity to the specific selection of start times. The comparison to JAERI results cannot, therefore, be used to show overall conservatism in the specification of chugging loads.

2.5 Discussion and Conclusions

To examine the effect of desynchronization on some specific sources the PSDs of the vertical force and overturning moment were computed for both SSES and the generic specifications. The results of Section 2.4 were applied directly to the bottom center pressures computed on the basis of the appropriate design sources.

For the SSES comparison, PTH No. 6 based on Source 306 was used as an example. This source was selected because it exhibits the highest energy content in the 25-50 Hz range. Figures A.2.11 and A.2.12 show the symmetric and asymmetric results respectively. Note that the PSD of the vertical force shows a potential nonconservatism at a significant peak around 29 Hz. While the energy content potentially missed by the specification is a small fraction of the total energy in the vertical force, it may have important consequences if a natural frequency of the structure exists in the underestimated frequency range. The potential nonconservatism of the specification of the overturning moment is even more evident in Figure A.2.12. The energy content may clearly be underestimated at virtually all frequencies above 10 Hz.

Similar results for the generic specification are presented in Figures A.2.13 and A.2.14 based on the bottom center pressure PSD bound of all the generic chugging sources (Figure 4.27 of Reference 7). The potential underestimation of energy content in the vertical force above 20 Hz and in the overturning moment above 10 Hz is clearly evident. For the specific choice of start times used, it is quite clear that any possible excitation of an asymmetric mode of the structure with a natural frequency above 10 Hz could be totally missed by the specification.

The consequences of the potential nonconservatism on the response spectrum level at specific nodes of the structure are difficult to assess without access to the full computer codes for the individual plants. The theoretical results of Section 2.3 together with the Monte Carlo trials of Section 2.4 suggest, however, that a specific set of start times will always produce almost total cancellation of any measure of structural response at some frequencies above the frequency ($f_o = 1/\tau$) associated with the time window τ. Because these frequency "holes" are dependent on the specific selection and assignment of start times to individaul vents, it is virtually impossible to guarantee a low exceedance probability at any frequency above f_o on the basis of the specified desynchronization. While the choice of the minimum variance set optimizes the synchronization to maximize the symmetric load at frequencies below f_o, no generalization of this hypothesis can be justified either at higher frequencies or for other measures of structural response.

The possible high probability of exceedance of the specified chugging loads, of course, does not necessarily imply lack of safety margin on any individual component in the plant. Other loads could be bounding in the relevant frequency range or other design constraints may have resulted in safety margins well above those imposed by chugging. Individual assessment, component by component, is clearly a difficult procedure at best. If one retains the "physical" intuition that the symmetric and asymmetric loadings provide two "extreme" conditions, that adequately describe the "major" structural excitations, one has a clear and attainable objective. The specification must provide loading conditions with low exceedance probability of both the vertical force and overturning moment, or frequency regions where the exceedance probability is high have to be bounded by other specifications. For instance, the generic condensation oscillation load (Reference 9, Figure 2.1) provides adequate margin for the vertical force PSD in the range of 20 to 50 Hz because of the synchronous application of the loading. Unfortunately, the lack of any appreciable energy in that frequency range in the KWU CO specification fails to provide the same conservatism for the SSES plant. No other asymmetric loading appears as an obvious candidate to bound the chugging induced overturning moment.

Relatively simple "fixes" to the present specification can define loading conditions that will provide an exceedance probability of less than 1 in 1000 for the vertical force and overturning moment. For instance, the addition of a loading specification which applies the sources at about 20% amplitude but synchronized in time, will provide adequate bounds over the 20 to 50 Hz range for the generic, SSES, and WPPSS symmetric loads. The application of the asymmetric loading with an asymmetric factor increased slightly above the specification and full synchronization in time can ensure a low exceedance probability of the overturning moment over the entire frequency range. Whether these are the best--or the easiest--procedures to provide adequate conservatism is not obvious without a more detailed examination of the actual application of these loading conditions.

3 MARK II OWNERS' RESPONSE TO VENT PHASING CONCERNS

The Mark II owners and GE responded to the concerns raised in Section 2 of this report at a meeting on April 8, 1982 in San Jose, California. The arguments presented at this meeting, documented in a subsequent report[10] dated April 28, 1982, demonstrate the adequacy of the chugging load specification for Mark II containments even with the phasing concerns taken into account. The three principal points discussed in this report are

(1) The effect of a selected set of start times on the pool response to the asymmetric specification is no more severe than the effect on the symmetric specification.

(2) The structural response of the containment to the asymmetric specification is similar to the response to the symmetric specification.

(3) Comparison of the symmetric chugging load specification with the JAERI data shows the specification to be conservative. The adequacy of the asymmetric chugging load specification is proven based on item (2) above.

The following paragraphs present the Mark II owners' arguments and the associated staff evaluation for each of these points.

3.1 Effect of Start Time Set on Asymmetric Specification

With the simplified analysis described in Section 2, the effect of a particular vent start time set (STS) on the asymmetric pool response is of greater severity than the effect on the symmetric response. The Mark II owners contend that in reality the effect of a selected STS on the asymmetric response is actually much less than that predicted by the simplified analysis and no greater than the effect on the symmetric response. They argue that analysis of the complete pool transfer function shows that asymmetric loads are generated primarily by the lowest asymmetric mode. Because this mode shape for the pressure generated by a single vent is a cosine function, and is constant with radius, the effect on the overturning moment is not proportional to the moment arm as assumed in the simplified analysis. Finally, because the asymmetric mode is not stationary in space but rotating in the Mark II geometry, there is no fixed axis in time about which a full amplitude overturning moment occurs with all vents participating. For these reasons, the Mark II owners concluded that the effect of a unique STS is deemed no more severe for the asymmetric than for the symmetric pool response.

The staff and its consultant agree with the qualitative arguments presented by the Mark II Owners Group. The approach of Section 2 was of necessity simplified because Professor Bienkowski did not have access to the pool acoustic models developed over the past few years by the Mark II owners. There is no doubt that some effects that influence the pool response are not included in the analysis of Section 2. The factors presented by the Mark II owners as mitigating the influence of start times on asymmetric response appear to be reasonable and acceptable.

3.2 Structural Response Similarities

The Mark II owners provided an analysis to show the similarity of the containment structural response to the asymmetric and symmetric part of the chugging load specification. Based on these analyses, the Mark II Owners Group concluded

(1) Hydrodynamic loads do not appear to induce any rocking motion in Mark II containments.

(2) Plant response to the chugging loads decreases as one moves away from the wetwell and is greatly reduced outside the primary containment.

(3) The analysis shows very little difference in responses to the symmetric and asymmetric chugging load.

As an example for these three conclusions, the Mark II owners provided[10] acceleration responses from various locations in the Shoreham plant that were the result of both symmetric and asymmetric loading. These results confirm for a concrete containment what the WNP-2 analysis showed for a steel one[11,12]--i.e., response to asymmetric and symmetric loading was within 20% at worst, and much closer over most of the frequency range. At the April 8, 1982 meeting, representatives of the Mark II owners stated that this similarity held at all of the many containment locations analyzed and that no exceptions to this similarity in response had been found.

The views of the staff and its consultant in regard to these new results are similar to that expressed for the WNP-2 data: If one looks at the details of the symmetric and asymmetric parts of the actual load specification, the response similarity is not surprising. The terms "symmetric" and "asymmetric" date back to the time of the lead plant chugging load specifications when all vents were assumed to chug synchronously. For the present chugging load specification, they are actually a misnomer because they refer to the spatial distribution of amplitudes only. The current desynchronized chugging load specification, which is much more realistic, assumes the vents chug independently within a 50 msec time window. Therefore, at all but the lowest frequencies, the asymmetry as a result of time desynchronization is greater than the asymmetry as a result of the amplitude variation, which is only from +14% average of the amplitude on one side of the containment to -14% on the other side. In other words, in reality, both parts of the specification contain similar major asymmetric elements in time, as the real chugging load would; hence, the similarity in response. Therefore, the staff and its consultant regard the conclusions of the Mark II owners regarding the containment structural response similarity to both parts of the chugging load specification to be valid.

3.3 Comparison with JAERI Data

Since chugging data became available from the prototypical multivent JAERI facility, the Mark II owners have used such data to assess the conservatism of their specifications. This comparison with JAERI became especially crucial once the specification included phasing among the vents. However, because the JAERI facility only represents a 20° sector of a typical Mark II containment, asymmetric effects as defined in the asymmetric specification (i.e., amplitude variation from one side of a Mark II containment to the other) cannot be assessed from the data. Therefore, only comparisons of the data with the symmetric part of the chugging load specification can be made. The points discussed in Sections 3.1 and 3.2 are provided to show that such a comparison of the symmetric part of the specification is sufficient to prove the whole chugging load specification (i.e. the symmetric and asymmetric) to be conservative. The actual comparison will be discussed below.

The original comparison made by GE[7] and the Mark II owners between the specification and JAERI data did not apply the specification to JAERI in the same way it would be applied to an actual Mark II plant. Although the particular procedure employed may have been more conservative for some aspects of the specification, it was clearly insufficient to resolve the concerns raised in Section 2 regarding vent phasing because average PSD envelopes were used.

To respond to the phasing concerns of Section 2, the following method was utilized by the Mark II owners to check the adequacy of the chugging load specification: The symmetric part of the chugging load specification was applied to the JAERI facility as it would be to a Mark II plant. One thousand STSs were drawn randomly from a 50 msec uniform window, and a set having the minimum variance was used as the start times for applying each of the 10 generic chug sources, one at a time, in an acoustic model of the JAERI facility. Spatially averaged pressure time histories for each source were then generated at the 1.8-m and 3.6-m elevations. The power spectral density (PSD) and pressure response spectra (PRS) were then calculated from the pressure time histories and envelopes over the response from the 10 sources were obtained. This whole procedure was repeated 20 times. Each time a new minimum variance STS was drawn from a sample of 1000. In other words, the specification was applied 20 times to the JAERI facility. Finally, minimum PSD and PRS envelopes of these 20 applications were constructed. These minimum envelopes from the specification were then compared with the corresponding maximum envelopes of the eight largest chugs from JAERI Test 0002, which contained some of the largest chugs observed in any of the tests. For the specification calculation a sonic speed based on the 4TCO facility was used for the JAERI facility. Because the JAERI sonic speeds appear to be higher than the 4TCO speeds and would result in higher calculated pressures, use of the 4TCO sonic speed produces a conservative comparison. The PRS comparison between the pressures obtained from the 20 minimum variance trials and the JAERI data are based on a damping value of 4% in accordance with Regulatory Guide 1.61 for reinforced concrete containments.

The calculations show that indeed each unique set of start times will result in a unique pressure time history on the pool boundaries. The PSD comparison between the minimum specification envelope and the maximum data envelope showed that the specification bounded the data up to about 45 Hz at both the 1.8-m and 3.6-m elevations with good margin. Above 45 Hz the JAERI data envelope exceeded the specification envelope in spots but the signal level for both the calculated and measured responses was extremely low.

Although a PSD comparison is valuable, the Mark II owners felt that a PRS comparison was more appropriate for a multimodal structure such as a containment. This is because at a given modal frequency a multimodal structure responds with a "resonant response" to the forcing function amplitude at a modal resonant frequency plus a static or "forced response" as a result of forcing function amplitudes at other frequencies. Whereas a PSD shows only the forcing function component at a given frequency (i.e., the "resonant response"), the PRS takes both the "resonant" as well as the "static or forced" response into account. Therefore, it is felt that a PRS is more indicative of the response energy felt by structures, piping and equipment due to chugging loads.

The PRS comparisons showed even more conservatism than the PSD ones. With no damping at all there was a slight "poke through" of the specification by the JAERI data between 45 and 50 Hz. However, with 4% or 7% damping, the JAERI data PRS was bounded by the specification minimum envelope PRS with substantial margins over the entire frequency range considered, i.e., zero to one hundred Hz. The more conservative bound at the higher frequencies shown in the PRS comparisons to the PSD were felt to be the result of the conservatively high power in the specification at the low frequencies which carries over (as it should) as a static response at the high frequencies in the PRS.

The staff and its consultant feel that the method chosen by the Mark II owners for comparing the symmetric part of the specification with the JAERI data is an appropriate one. Comparing the minimum envelope of 20 different applications of the specification with the maximum envelope of the JAERI data is certainly conservative. The PSD relationships are unquestionably conservative below 45 Hz where most of the energy of the chugging loads lie. The "poke throughs," which occur only at higher frequencies where amplitudes are low, are not troublesome in light of the PRS data. The staff and its consultant feel that the PRS comparison is an appropriate measure of response and that it confirms that the structure and equipment will be conservatively assessed by application of the chugging load specification as it now stands.

4 CONCLUSION

The staff and its consultant feel that the Mark II owners' approach toward resolving the concerns of Section 2 is a sound one. After reviewing the arguments presented at the April 8, 1982 meeting and documented in Reference 10, the staff and its consultant are in general accord with the conclusions drawn by the Mark II owners. The qualitative arguments, showing that the effect of start time selection on the asymmetric specification is not as severe as that predicted by the analysis in Section 2, is reasonable. The quantitative data provided by the acceleration response at various containment points[10,11] show that the asymmetric and symmetric portions of the chugging load specification elicit very similar responses from the containment, i.e., no significant response as a result of one was found that was not also produced by the other. These arguments, in the staff's view, provide sufficient rationale for accepting the comparison of the symmetric portion of the chugging load specification with the JAERI data as proof for the adequacy of the whole chugging load specification. The method chosen by the Mark II owners for this comparison is indeed a rigorous one and shows that the inherent conservatism in the source strength, as presently specified, is sufficient to compensate for any "frequency holes" resulting from a specific set of start times. Any minor "poke throughs" in the PSD comparisons are adequately resolved by the PRS data.

The staff and its consultant conclude that the present generic chugging loads specification is still adequately conservative.

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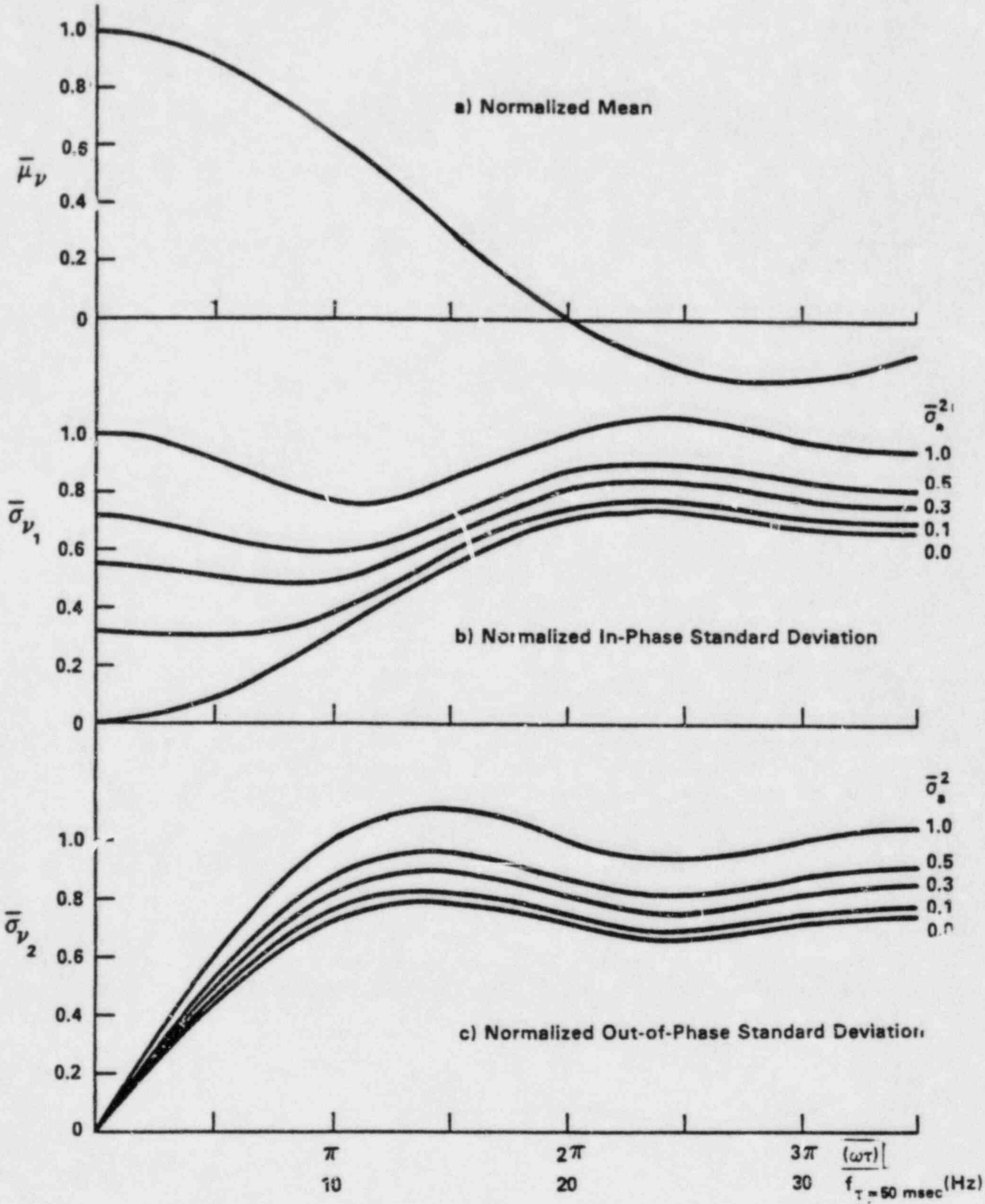

Figure A.2.1 Probability parameters

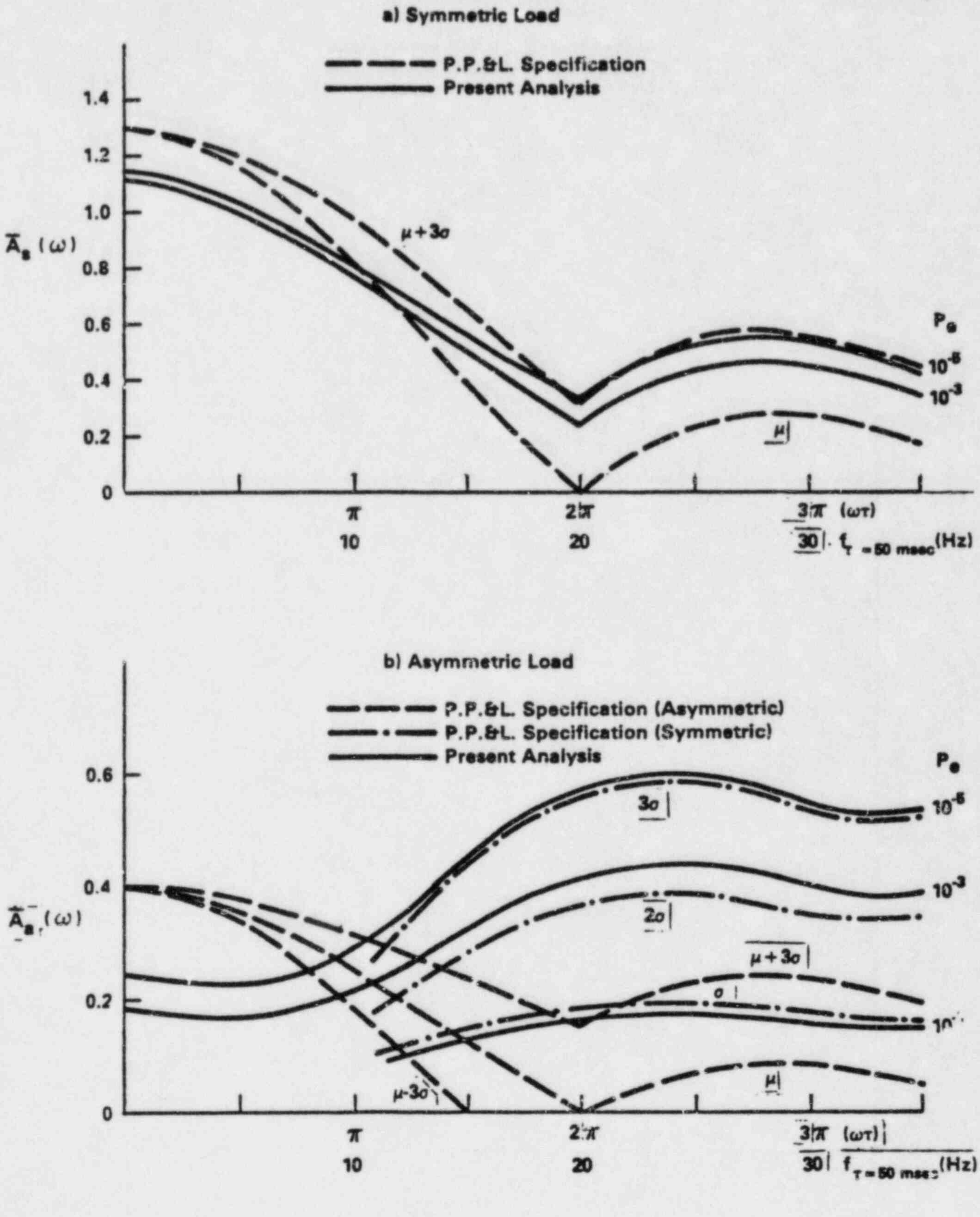

Figure A.2.2 Comparison of present analysis to PP&L specification

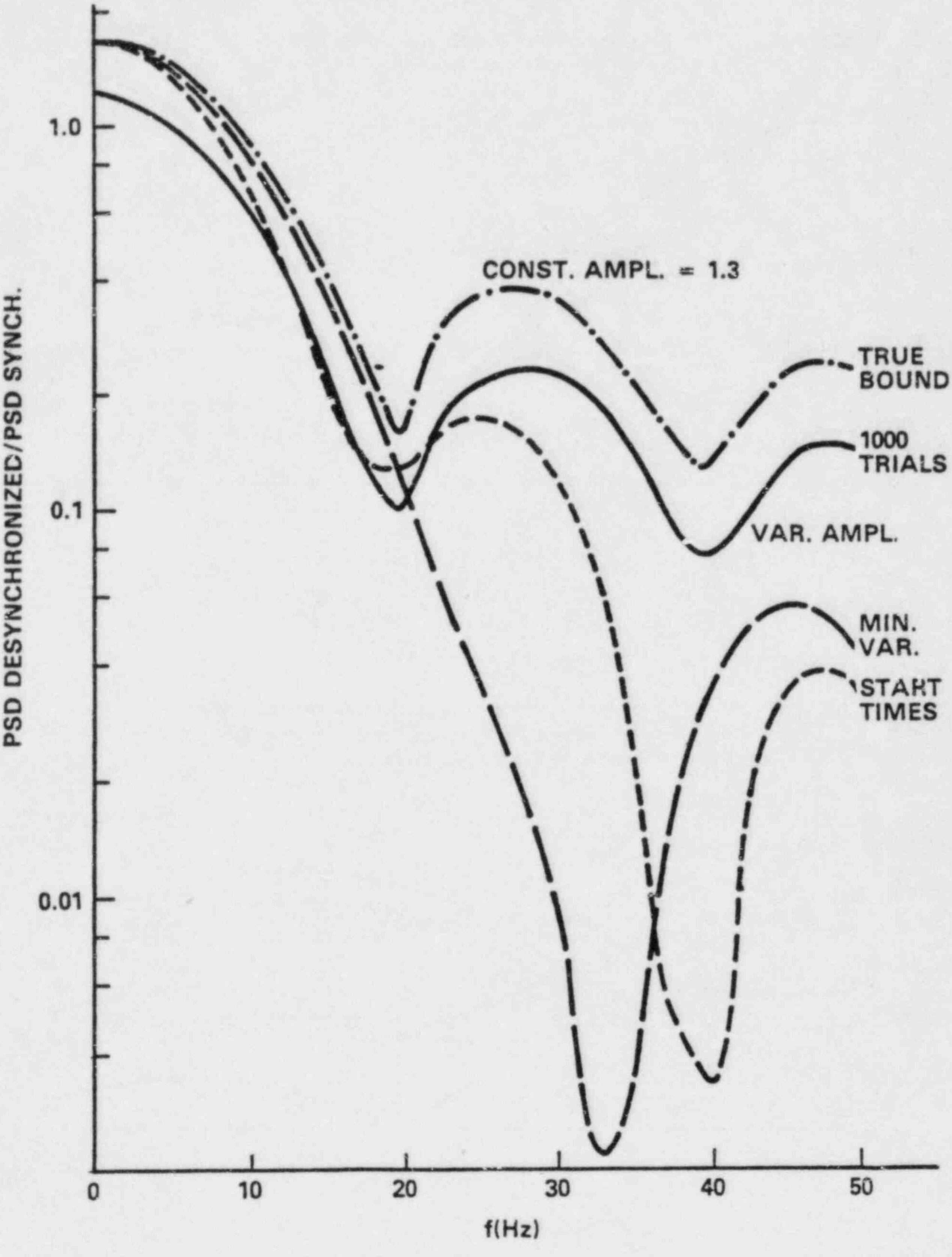

Figure A.2.3 PP&L symmetric load vertical force

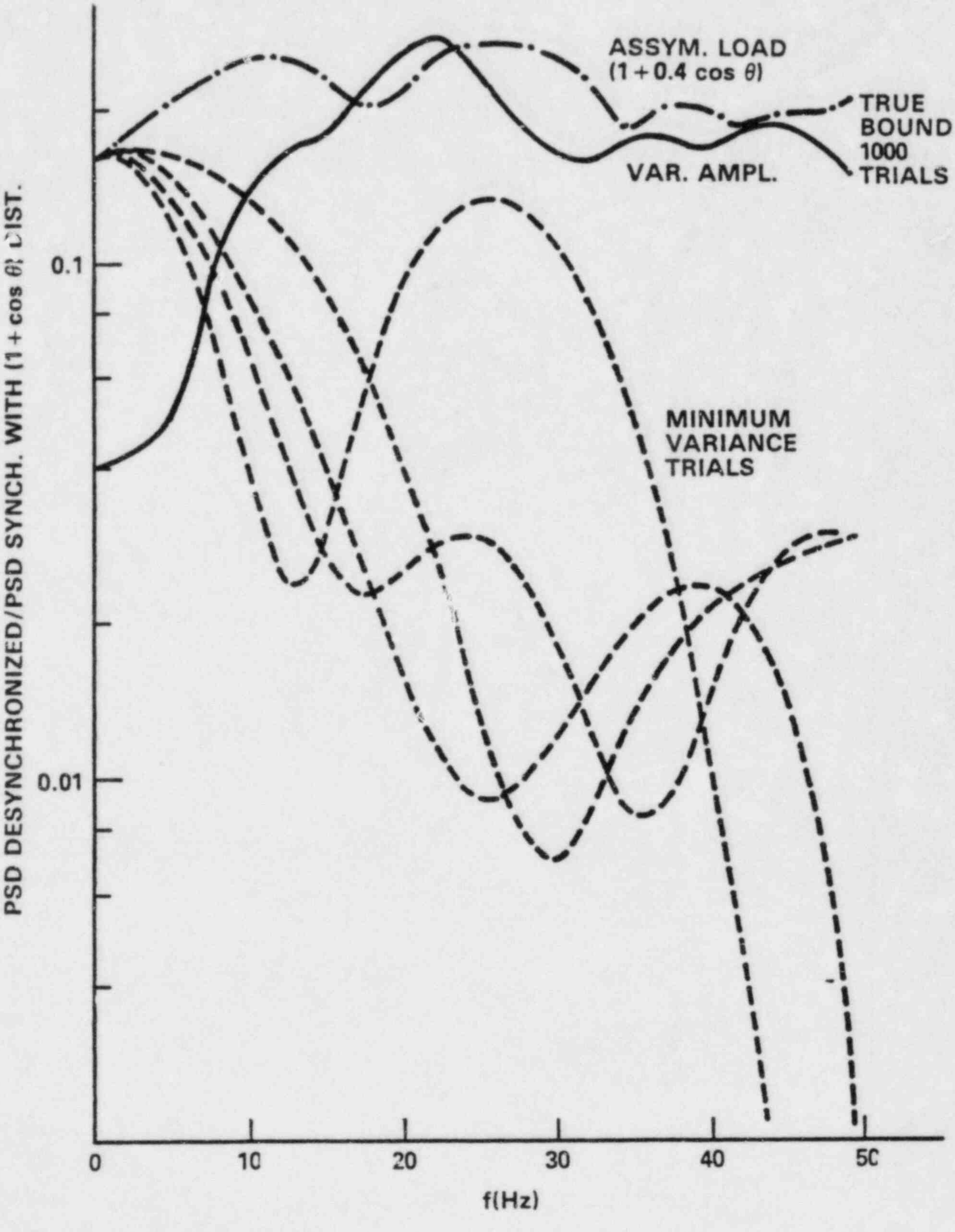

Figure A.2.4 PP&L asymmetric load overturning moment $\alpha = 0.4$

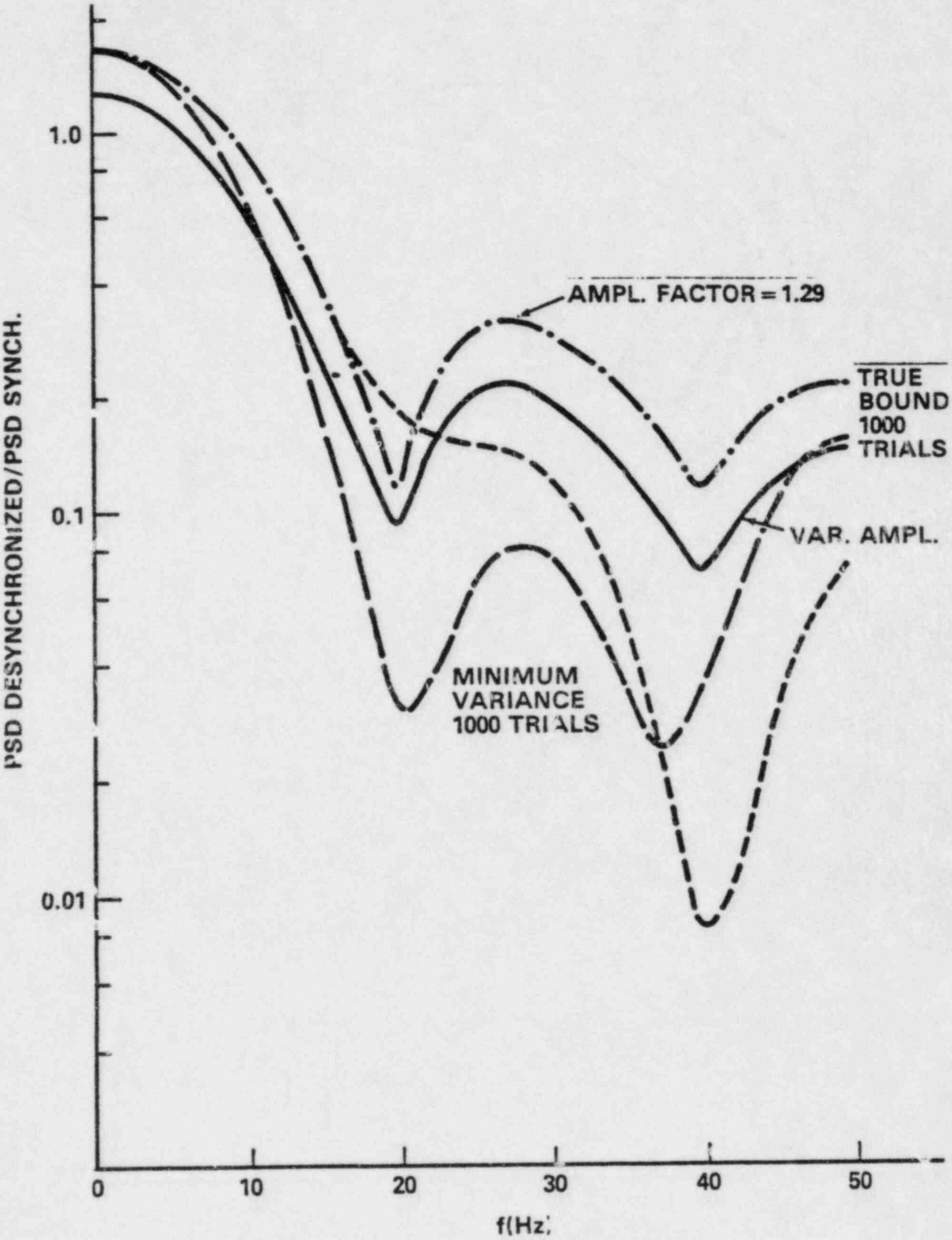

Figure A.2.5 Generic 87-vents symmetric load vertical force

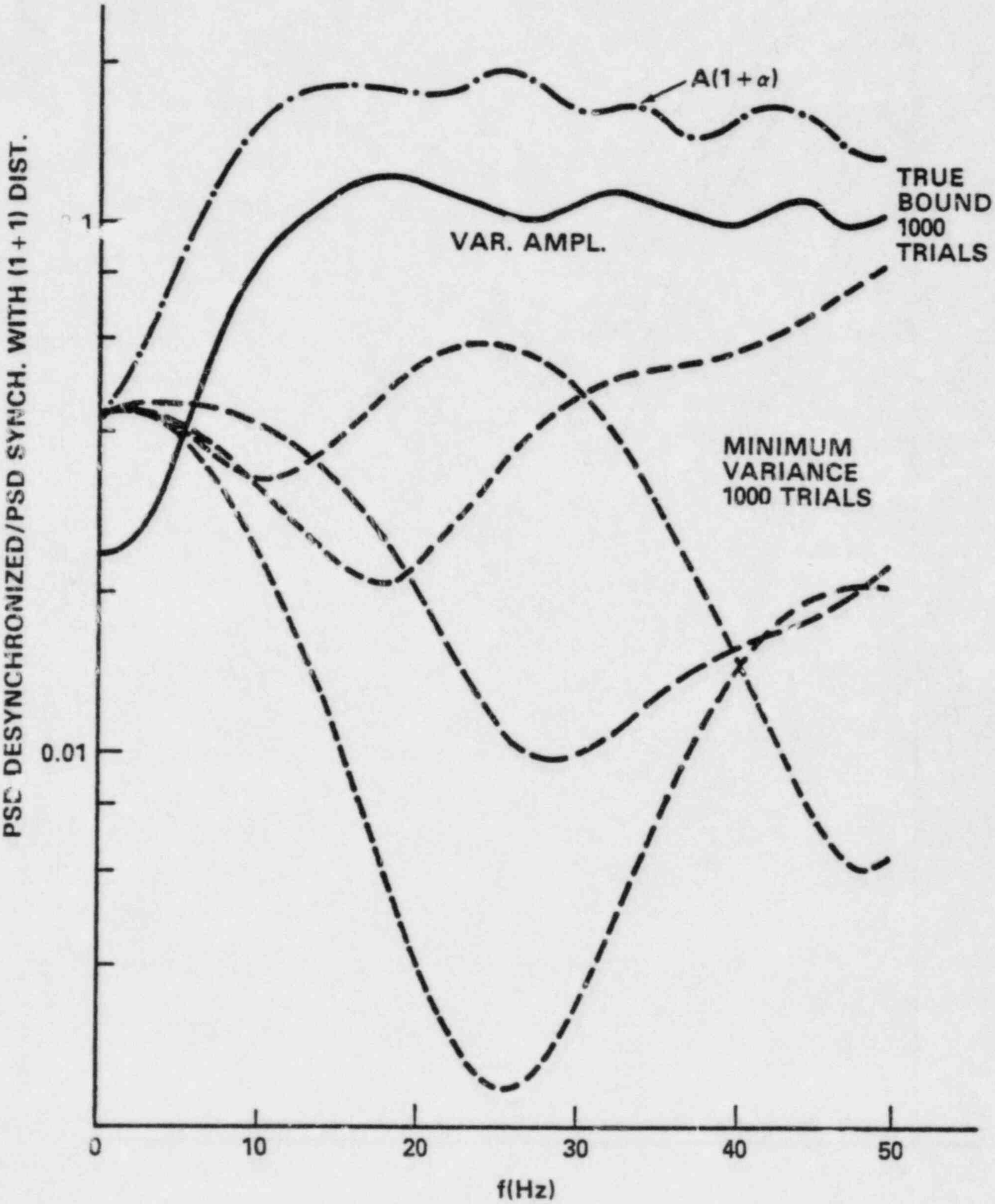

Figure A.2.6 Generic 87-vents asymmetric load overturning moment $A(1 + \alpha)$ with $A = 1.29$ $\alpha = 0.155$

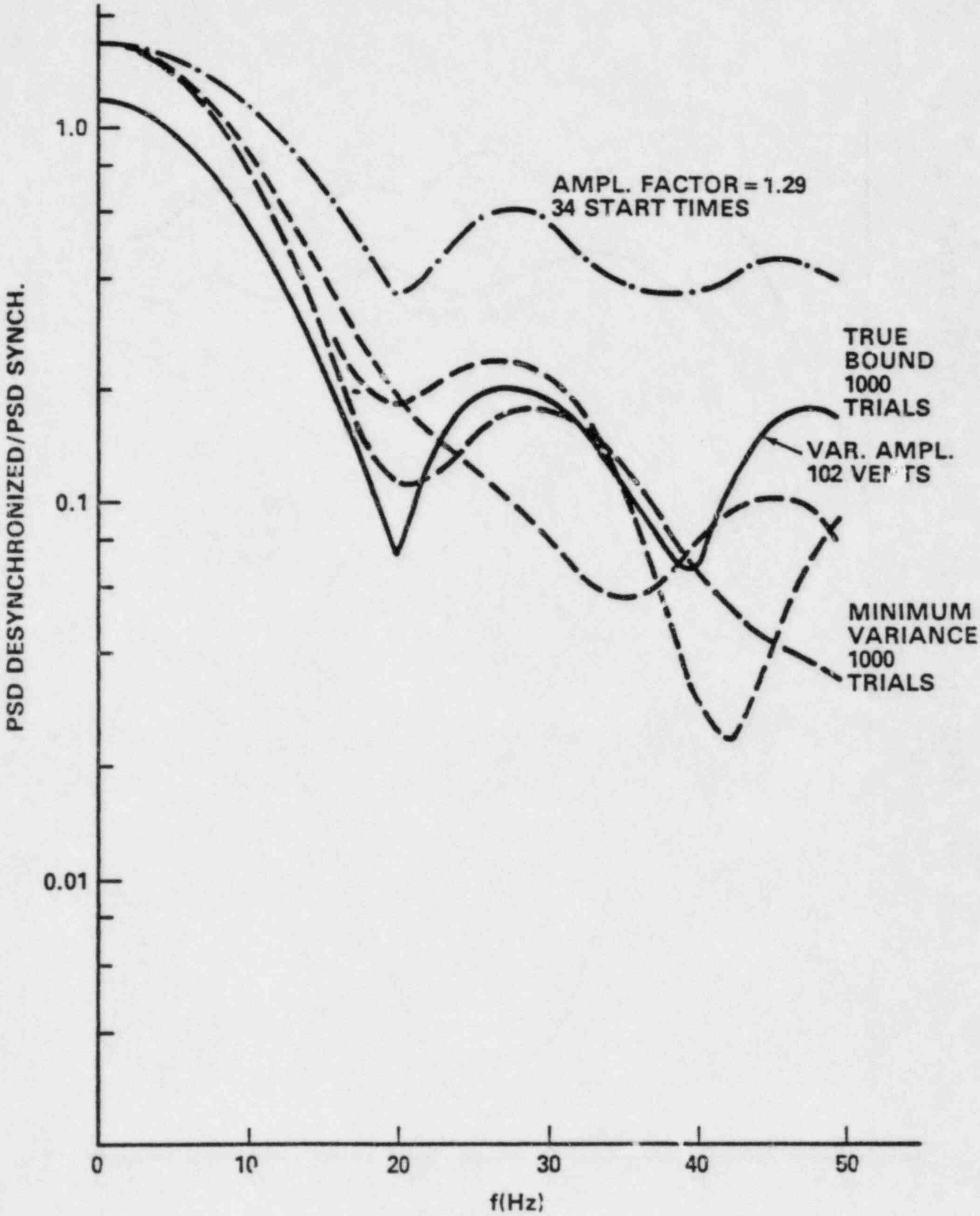

Figure A.2.7 WPPSS symmetric load vertical force A = 1.29

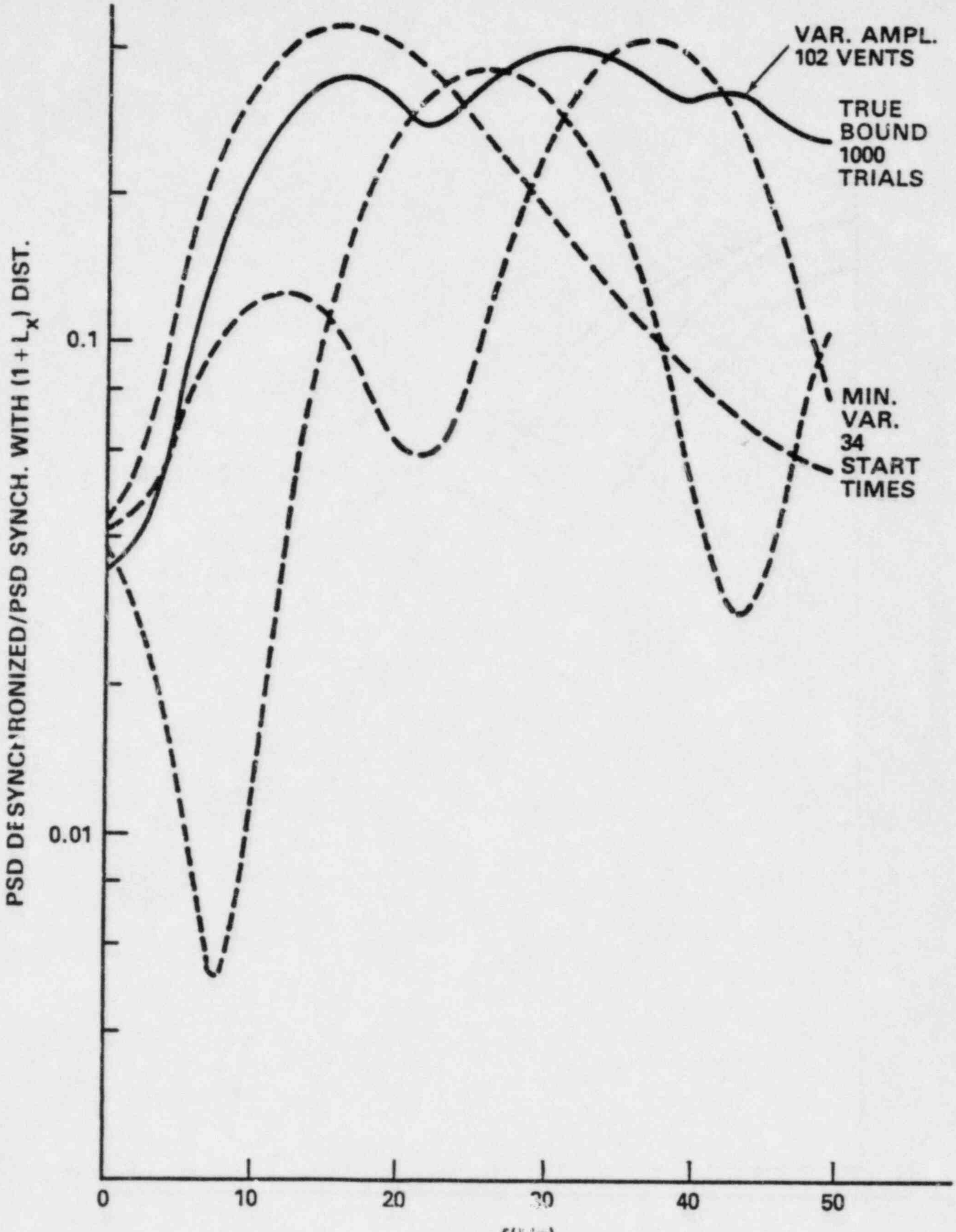

Figure A.2.8 Asymmetric load overturning moment
$\alpha = 0.14$ A = 1.29 A$(1 + \alpha L_x)$

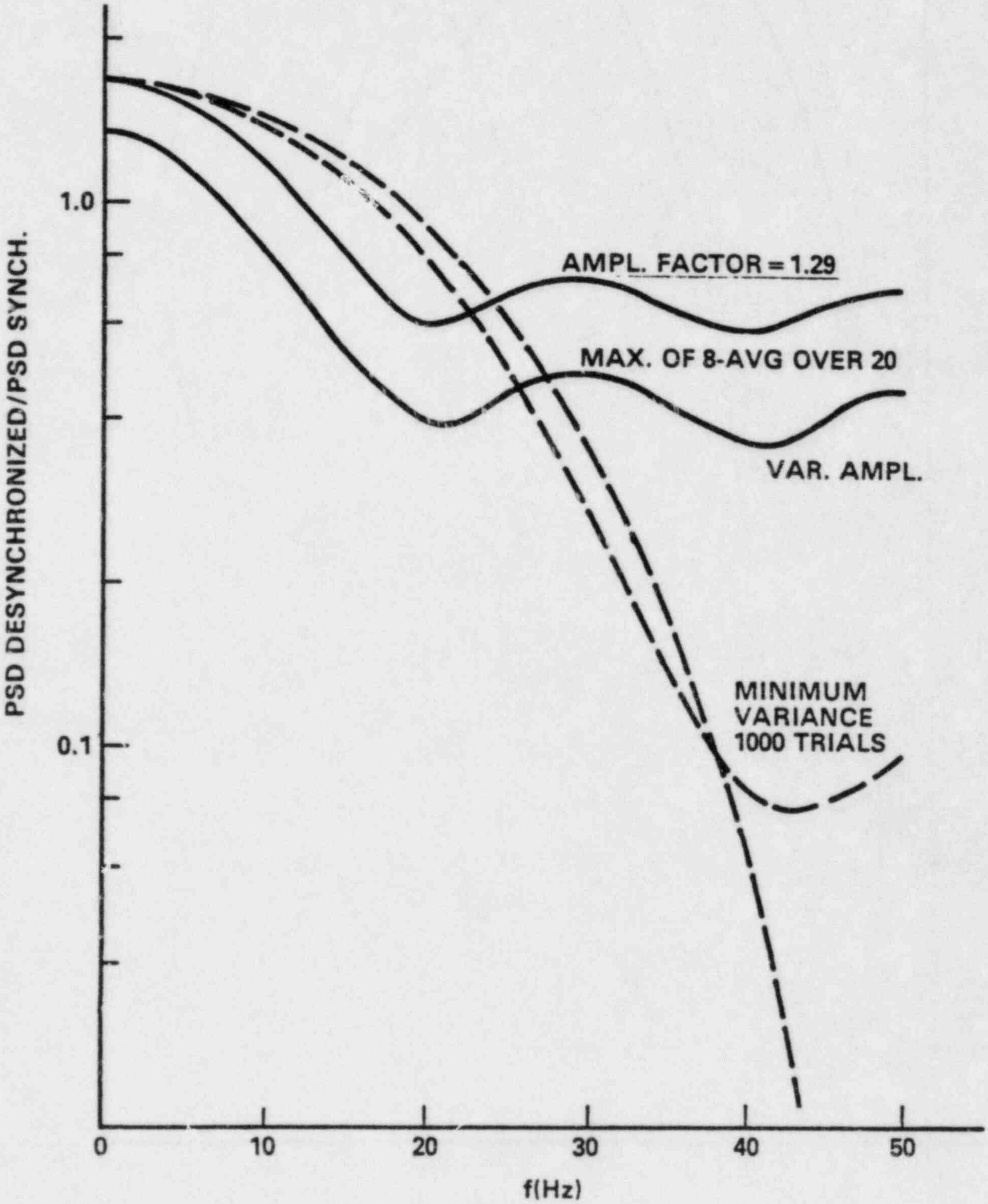

Figure A.2.9 JAERI symmetric load vertical force generic

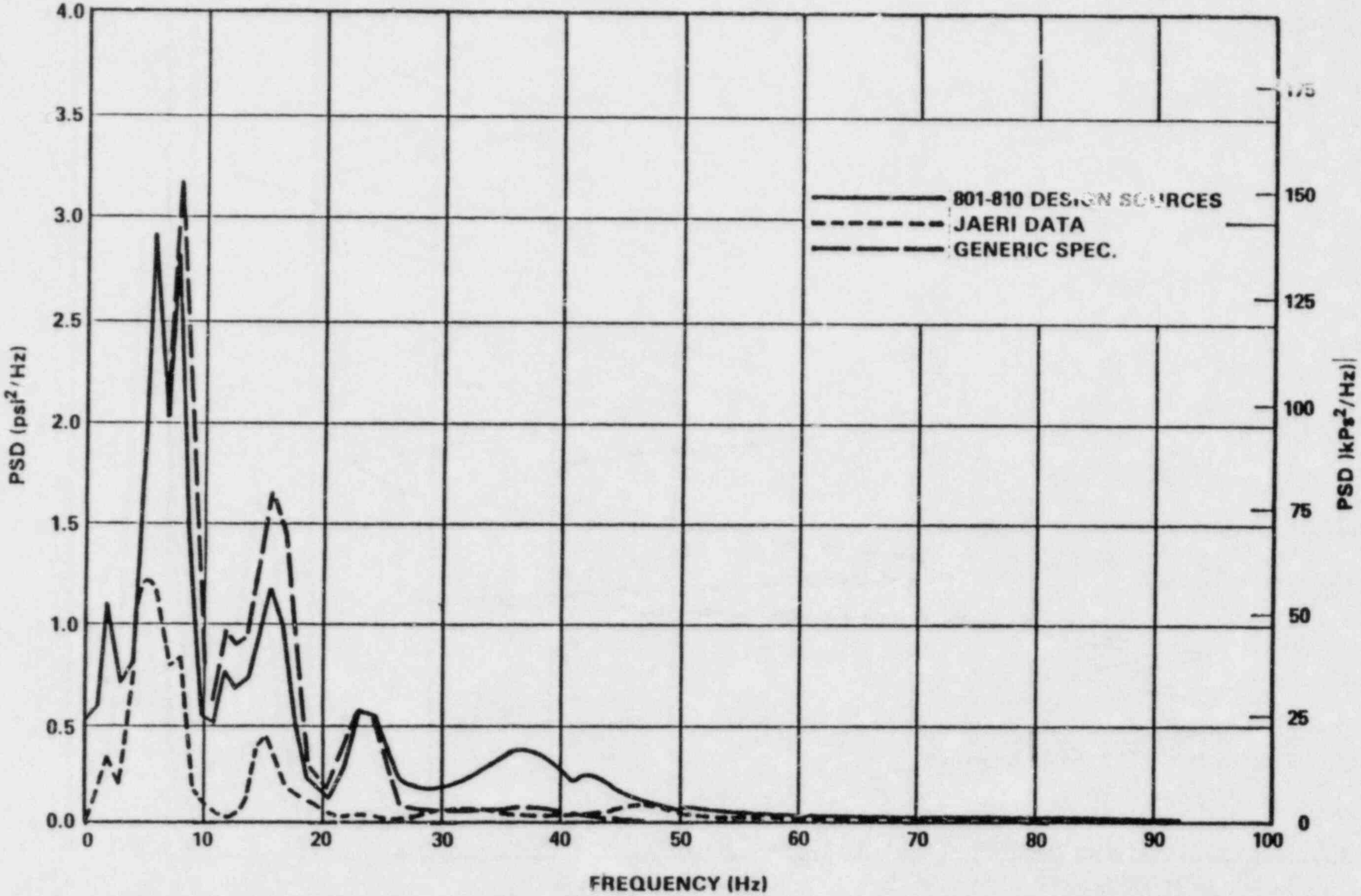

Figure A.2.10 Comparison of generic specification to JAERI Data (1800mm location)

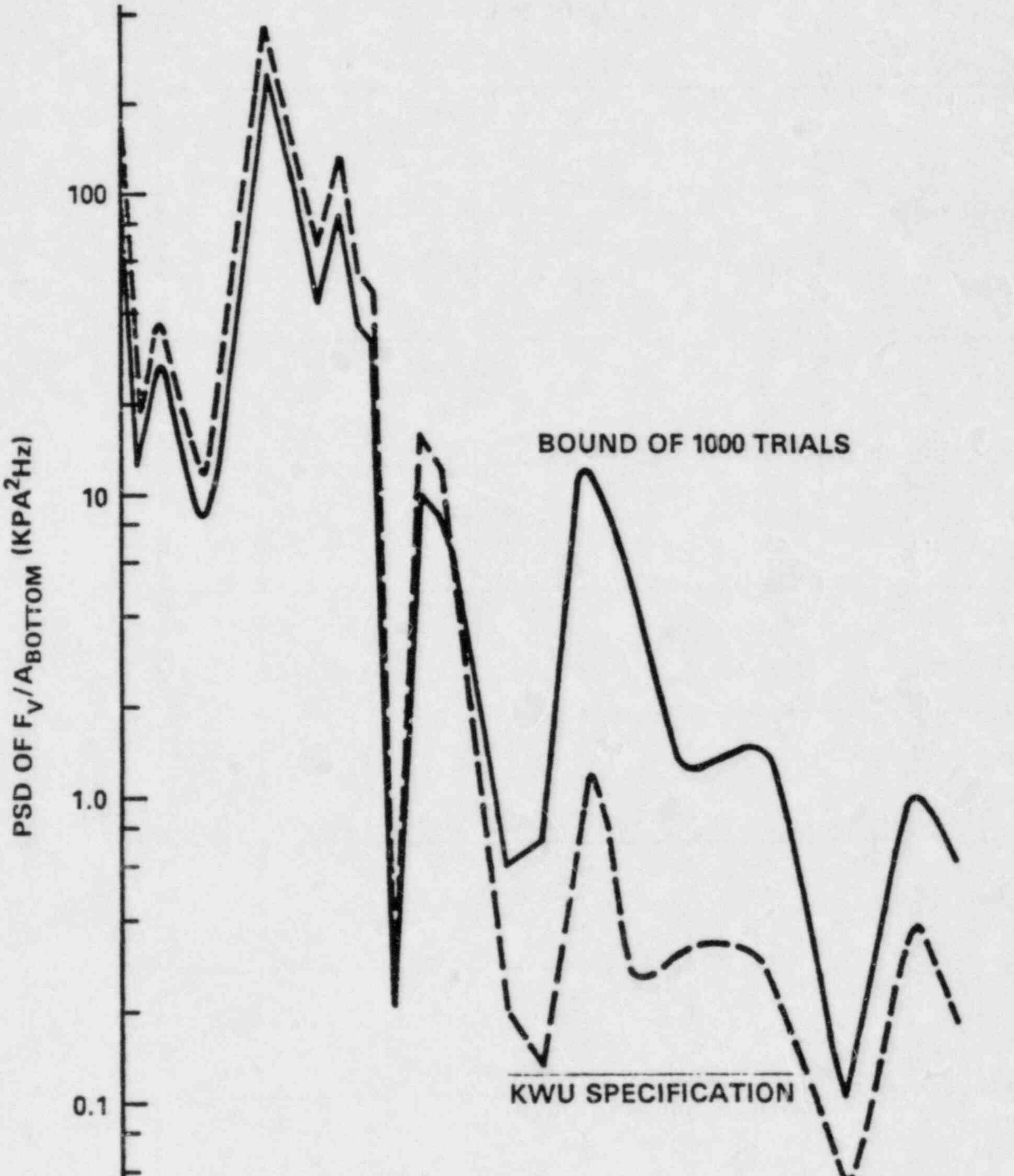

Figure A.2.11 SSES-87 vents symmetric load PTH no. 6-source 306

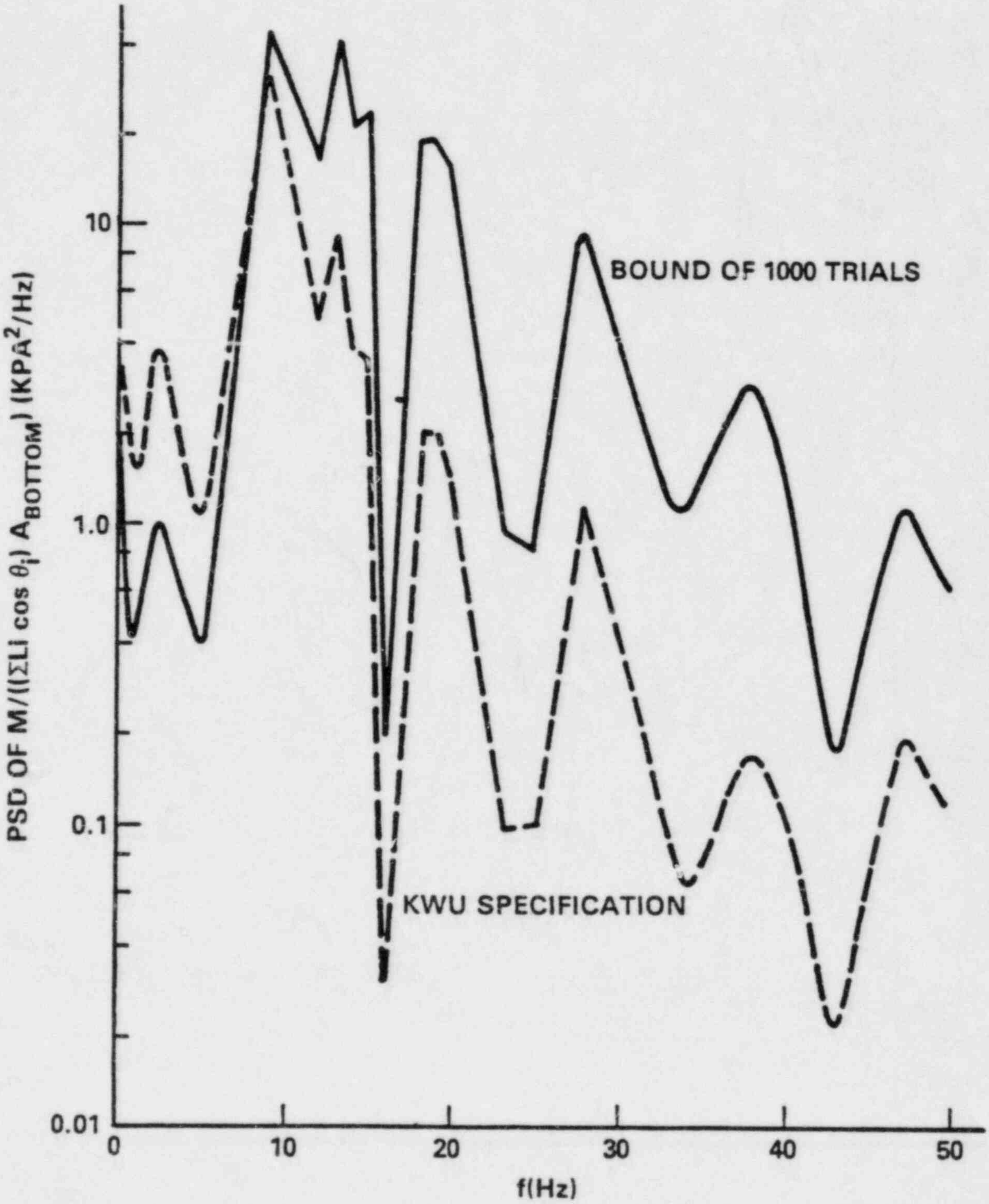

Figure A.2.12 SSES-87 vents asymmetric load PTH no. 6-source 306

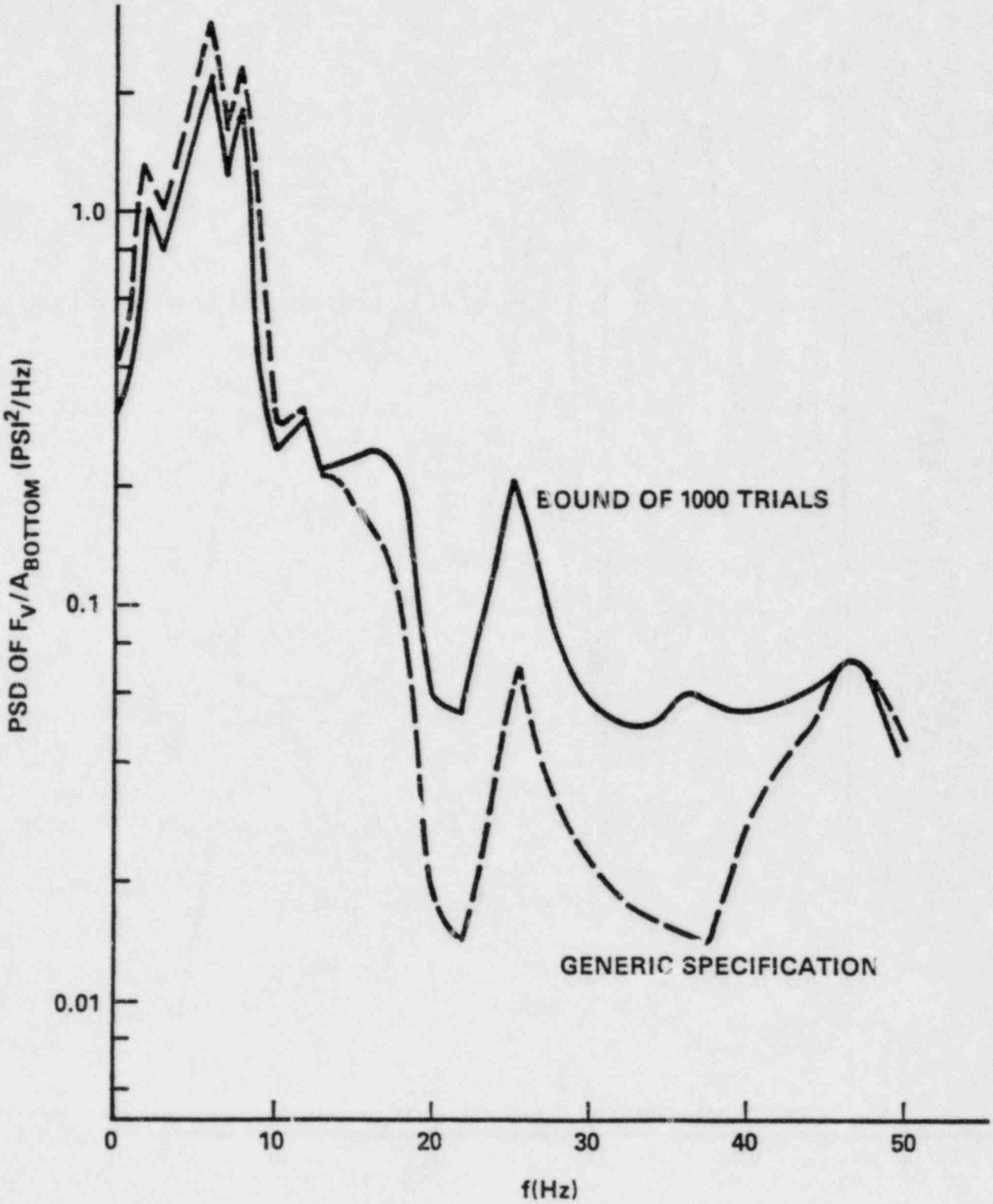

Figure A.2.13 Generic-87 vents symmetric load based on bounding envelope of sources 801-810

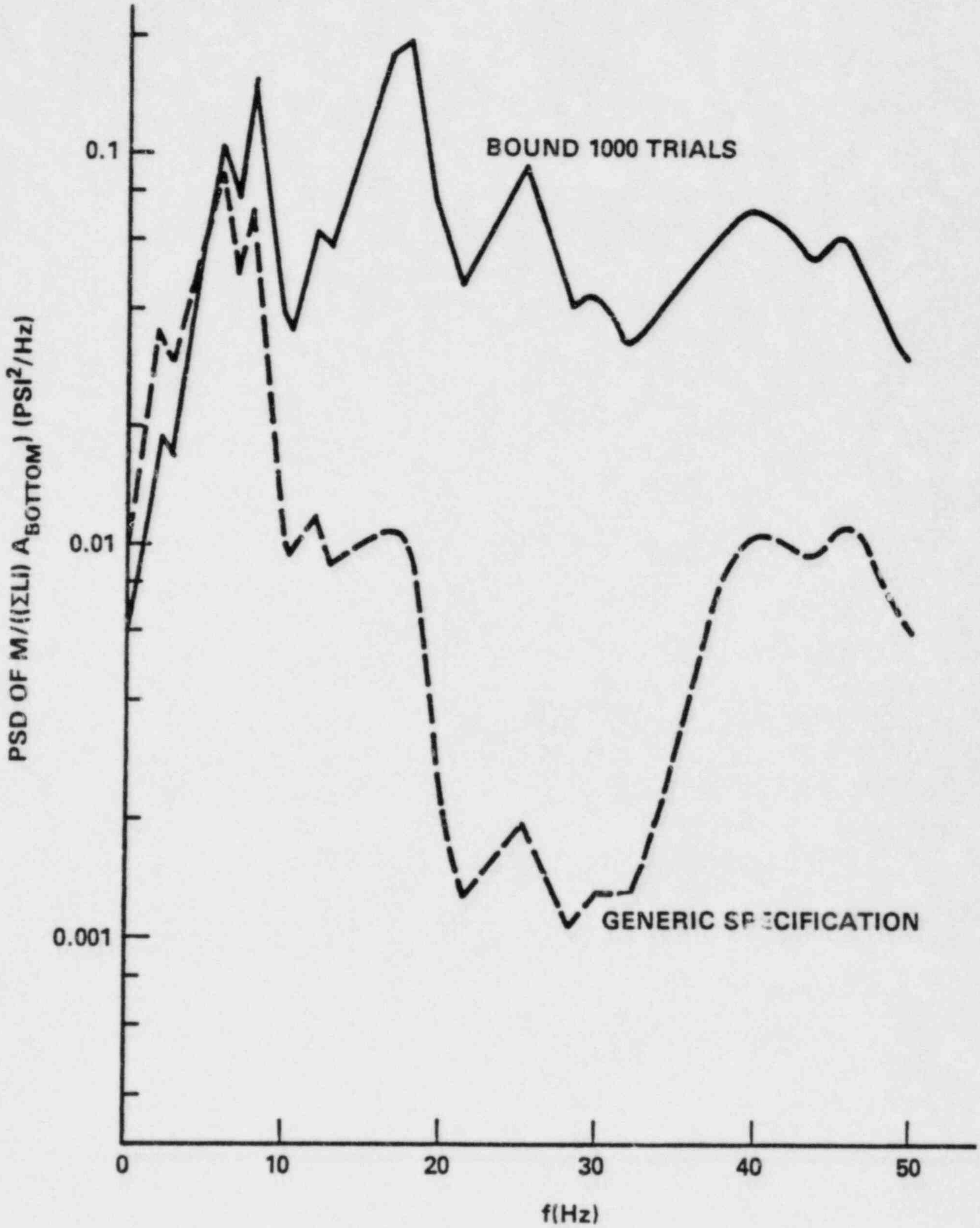

Figure A.2.14 Generic-87 vents asymmetric load envelope of sources 801-810

APPENDIX B

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APPENDIX C

ERRATA TO SAFETY EVALUATION REPORT/
SAFETY EVALUATION REPORT SUPPLEMENT NO. 1

Errata to SER

Page 6-51, Line 27	Delete "electric heating coil."
Page 12-5, Line 24	The penultimate sentence of the paragraph should read: "These units provide continued surveillance for all of the airborne area, process, and effluent monitors."

Errata to SSER 1

Page 3-1, Line 9	After "These additional loads were not" insert "considered in the original design of this."
Page 10-2	Delete the second, third, and fourth paragraphs.

APPENDIX D

INCOMPLETE IMPLEMENTATION OF CONTROL ROOM IMPROVEMENTS

This appendix provides the staff position on those improvement commitments for which implementation is incomplete. The positions listed below are referenced to the HED numbers as they appear in NUREG-0420, SSER No. 1.

1.7	Not yet installed.
1.8	Not yet installed and tested.
2.1	Not yet tested.
3.11	Annunciator audible alarms on adjacent panels do not exhibit sufficient frequency differentiation. No change from original HED.
3.13	Not yet accomplished.
3.14	No change has been made; HED remains.
4.1	Test to be conducted by General Physics; not yet accomplished.
4.9	Not yet accomplished.
4.10	Not yet accomplished.
5.8	Not yet corrected. Meters appear to have been locally modified and the problem compounded.
5.10	Incomplete. Some meters corrected, some not.
6.15	Mimic is complex and needs more arrows showing direction of flow.
7.5	Could not determine what corrective action was taken. Needs explanation.
7.6	Could not determine what corrective action was taken. Needs explanation.
7.7	Annunciators are still across the room from controls. Not corrected.
7.8	No change has been made.
7.9	Color padding has not been accomplished.
8.1	Panel is still inconsistent with remainder of control room and readability is poor.
8.2	Readability still poor; no change made.
8.22	Control room not fully evaluated regarding this discrepancy; HEDs remain.
8.30	Partially corrected, mimic lines should end at black padding and not at individual components.
9.4	No attempt made to make wording on computer displayed alarms the same as, or even similar to, the annunciator tile wording. This problem is particularly serious since Shoreham does not have a "first out" panel and is taking credit for the sequence of events recorder to accomplish the "first out" function.

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(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT *(200 words or less)*

Supplement No. 3 to the Safety Evaluation Report of Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement addresses several items that have come to light since the previous supplement was issued.

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